RECEIVED BY TIC SEP 7 1976

PPPL-1275

PH-R-001

February, 1976

Tokamak
Fusion
Test
Reactor

NOTICE

PORIONS OF THIS REPORT ARE ILLEGIBLE. It has been reproduced from the best available copy to permit the broadest possible availability.

FINAL CONCEPTUAL DESIGN REPORT

Princeton University
PLASMA Physics Laboratory
Princeton, New Jersey

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

Westinghouse Electric Corporation
Fusion Power Systems Department
Pittsburgh, Pennsylvania

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Administration, nor any person acting on behalf of the Administration:

- A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method or process disclosed in this report may not infringe privately owned rights; or
- B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Administration" includes any employee or contractor of the Administration to the extent that such employee or contractor prepares, handles or distributes, or provides access to, any information pursuant to his employment or contract with the Administration.

~~APPLIED TECHNOLOGY~~

Any further distribution by any holder of this document, or of the data therein, to third parties representing foreign interests, foreign governments, foreign companies and foreign subsidiaries, or foreign division of United States companies, should be coordinated with the Director, Division of Thermonuclear Research, United States Energy Research and Development Administration.

Tokamak Fusion Test Reactor

FINAL CONCEPTUAL DESIGN REPORT

Princeton University
PLASMA Physics Laboratory
Princeton, New Jersey

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

Westinghouse Electric Corporation
Fusion Power Systems Department
Pittsburgh, Pennsylvania

CONTRIBUTORS

Princeton Plasma Physics Laboratory

R. Bergman	S. Goldfarb	C. C. Osgood
P. Bonanos	M. B. Gottlieb	M. Pelovitz
G. Bronner	R. Hawryluk	D. Post
R. L. Cassel	G. Hill	P. J. Reardon
U. R. Christiansen	D. Jassby	P. Rutherford
R. S. Christie	F. Jobes	N. Sauthoff
J. C. Citrolo	H. G. Johnson	J. Schmidt
J. R. Clarke	J. B. Joyce	G. V. Sheffield
R. E. Daniels	D. Lee	C. Szerlag
A. L. Deitz	R. Little	H. Towner
J. File	R. Middlebrook	K. E. Wakefield
E. A. Frieman	D. H. Mullaney	M. Yamaguchi
H. Furth	J. G. Murray	K. Young

Westinghouse Electric Corp.

W. C. Brenner	D. Grove	F. A. Reed
W. G. Brussalis	W. J. Havener	R. P. Rose
S. Chattopadhyay	F. M. Heck	D. Shankle
E. A. DeZubay	E. H. Hemmerle	Z. M. Shapiro
G. E. Driesen	E. Iwinski	D. A. Sink
J. E. Faulkner	J. Jedruch	G. Smeltzer
B. J. Fedor	C. K. Kim	R. A. Smith
R. Flaherty	C. S. Klanian	M. Sniderman
W. J. Francy	D. Klein	C. Storey
J. W. French	R. Lampe	F. G. Tauch
H. J. Garber	W. J. Lange	L. E. VanBibber
G. Gibson	A. Y. Lee	T. C. Varljen
R. E. Gold	J. H. McWhirter	J. M. Walsh
D. Green	W. E. Newell	J. E. Zerbe

Important Contributions Were Also Made By:

L. F. Bendinelli	L. Hiraback	J. H. Murphy
P. Blake	G. J. Janosik	J. O. Nichols
C. A. Booker	C. K. Jones	C. L. North
W. A. Bryant	W. J. Jones	R. P. Putkovich
R. T. Byerly	I. Keyfitz	B. Schumacher
J. W. H. Chi	D. S. Kimmel	H. Steinbruegge
D. K. Davies	L. A. Kilgore	R. Sturgess
D. Drawbaugh	J. L. Koetting	M. S. Walker
J. Ellis	J. A. Kyslinger	W. Wallace
J. Fink	W. Lester	J. M. Wells
J. G. Gorman	F. S. Malick	E. C. Whitney
M. W. Hagadorn	E. M. Martin	J. W. Wigert
P. H. Houser	R. W. Miller	J. E. Wolf

PREFACE

The Tokamak Fusion Test Reactor Conceptual Design Study was performed by a joint task force composed of representatives of the Plasma Physics Laboratory of Princeton University and the Fusion Power Systems Division of the Westinghouse Electric Corporation. The period of performance of this study was February, 1974 through February, 1976.

This work was supported by the Division of Controlled Thermonuclear Research of the U.S. Energy Research and Development Administration, Contract E(11-1)-3073, and Subcontract 193, thereto. Reproduction, translation, publication, use and disposal, in whole or in part, by or for the United States Government is permitted.

TOKAMAK FUSION TEST REACTOR

TABLE OF CONTENTS

Section		Page
1.0	INTRODUCTION	1-1
2.0	SCIENTIFIC MOTIVATION	2-1
2.1	Experimental Background	2-1
2.1.1	Attainment of Reactor-Like Plasmas	2-3
2.1.2	Attainment of Reactor-Like Fusion Power Densities	2-7
2.2	Summary of Principal Capabilities of the TFTR Device	2-10
3.0	PLASMA EXPERIMENTS	3-1
3.1	Plasma Physics Aspects	3-1
3.1.1	Discharge Initiation	3-1
3.1.1.1	Initial Magnetic Field Structure	3-1
3.1.1.2	Initial Plasma Current Rise	3-2
3.1.1.3	Control of the Skin Effect	3-5
3.1.1.4	Volt-Second Requirements	3-5
3.1.2	MHD Equilibrium and Stability	3-8
3.1.2.1	Poloidal Equilibrium Field	3-8
3.1.2.2	Stability Against Kink Modes	3-8
3.1.2.3	High - β_p Limitations	3-9
3.1.3	Plasma Energy Transport	3-10
3.1.3.1	Anomalous Transport due to Microinstabilities	3-10
3.1.3.2	Energy Transport by Neutrals and Impurities	3-11
3.1.3.3	Large Orbital Excursions	3-13
3.1.4	Impurity Phenomena	3-15
3.1.4.1	Impurity Influx	3-15
3.1.4.2	Effect of Impurities on Fusion Power	3-17
3.1.4.3	Effect of Impurities on Neutral Beam Penetration	3-17

TABLE OF CONTENTS (Continued)

Section		Page
3.1.5	Auxiliary Heating	3-17
3.1.5.1	Neutral Beam Heating	3-19
3.1.5.2	Ion Cyclotron Heating	3-21
3.1.5.3	Lower Hybrid Heating	3-22
3.1.6	Adiabatic Compression	3-23
3.1.7	Termination of the Discharge	3-26
3.2	Computer Studies of the Operating Modes	3-27
3.2.1	Contents of the Transport Code	3-27
3.2.1.1	Six-Regime Transport Model	3-27
3.2.1.2	Neutral Atom Background	3-29
3.2.1.3	Impurities	3-29
3.2.1.4	Neutral Beam Injection	3-31
3.2.1.5	Adiabatic Compression	3-31
3.2.1.6	Special Conditions for the Present Runs	3-32
3.2.1.7	Computer Diagnostics	3-32
3.2.2	Ohmic Heating	3-32
3.2.3	Injection and Compression	3-35
3.2.3.1	The Weak-Compression Mode	3-35
3.2.3.2	The Strong-Compression Mode	3-40
3.2.3.3	Variation of the Transport Coefficients	3-42
3.2.4	Quasi-Static Injection	3-42
3.2.4.1	Low-Current Discharge	3-45
3.2.4.2	High-Current Discharge	3-45
3.3	Reactor Physics Aspects	3-47
3.3.1	Two-Component Reactor Regime	3-49
3.3.1.1	Deuteron Slowing-Down Process	3-49
3.3.1.2	Energy Multiplication	3-51
3.3.1.3	Fusion Power Density	3-52
3.3.1.4	Energy Clamping	3-53
3.3.1.5	Stability of the Two-Component Plasma	3-55
3.3.1.6	Applications of TCT-Type Reactors	3-56
3.3.2	"Pulsed Ignition" Studies	3-58
3.3.2.1	Alpha Particle Heating	3-58
3.3.2.2	Pulsed-Ignition Tokamak Reactor	3-60

TABLE OF CONTENTS (Continued)

Section		Page
4.0	CONCEPTUAL DESIGN STATUS	4-1
4.1	System Engineering and Design Considerations	4-1
4.1.1	Reference Design Overview	4-1
4.1.2	Reference Experimental Scenarios	4-10
4.1.3	Neutronic and Photonic Design	4-17
4.1.4	Magnetic Field Topology and Design	4-25
4.1.5	EF/OH/Plasma Circuitry	4-32
4.1.6	Thermal and Hydraulic Design	4-38
4.1.7	Device Structural Analysis	4-48
4.1.8	Tritium Handling	4-58
4.1.9	Assembly and Remote Servicing	4-67
4.1.10	Power Supply	4-71
4.1.11	Facility	4-76
4.1.12	Diagnostics	4-87
4.1.13	Neutral Beam Injection System	4-90
4.1.14	Control Systems	4-93
4.1.15	System Studies	4-99
4.2	Tokamak System Design	4-107
4.2.1	Vacuum Vessel Assembly	4-110
4.2.2	Structure and Shielding	4-117
4.2.3	Toroidal Field Coil Assemblies	4-126
4.2.4	Poloidal Field Coil Assemblies	4-131
4.3	Electrical Systems	4-136
4.3.1	Energy Supply and Distribution System	4-137
4.3.2	Toroidal Field Pulsed Energy Conversion System	4-145
4.3.3	Ohmic Heating Field Pulsed Energy Conversion System	4-149
4.3.4	Equilibrium Field Pulsed Energy Conversion System	4-153
4.3.5	Central Instrumentation and Control System	4-156
4.4	Experimental Area Systems Design	4-168
4.4.1	Remote Handling and Maintenance Systems	4-172
4.4.2	Tritium Storage and Delivery System	4-179
4.4.3	Tritium Cleanup System	4-189
4.4.4	Non-Tritium Charge Gas Delivery System	4-194
4.4.5	Machine Area Water Cooling System	4-200
4.4.6	Torus Vacuum Pumping System	4-212

TABLE OF CONTENTS (Continued)

Section		Page
4.5	Neutral Beam Injection System	4-217
4.5.1	Neutral Beam Line	4-217
4.5.2	Neutral Beam Pulsed Energy Conversion System	4-237
4.6	Diagnostic Systems	4-242
4.6.1	Experimental Requirements	4-242
4.6.2	Principal Interactions with Other Systems	4-243
4.7	Civil, Architectural and Structural Systems	4-248
4.7.1	Summary	4-248
4.7.2	Design Description	4-252
5.0	PROGRAM PLAN	5-1
5.1	Organization and Key Personnel	5-1
5.2	Work Breakdown Structure	5-4
5.3	Project Cost Estimate	5-7
5.4	Project Schedule	5-9
5.4.1	Schedule Assumptions	5-9
5.4.2	Critical Path Assembly	5-12
5.4.3	Project Milestones	5-15
5.5	Safety and Environmental Impact	5-15
5.5.1	Safety	5-15
5.5.2	Environmental Impact	5-18
6.0	SUPPORTING PROGRAMS	6-1
6.1	Introduction	6-1
6.2	Research and Development Activities	6-2
6.3	Experimental Research	6-3
6.4	Facility Operation	6-4
6.5	Spare Parts	6-5

LIST OF ILLUSTRATIONS

Figure		Page
2.1-1	The Tokamak Magnetic Field Configuration	2-2
2.1-2	Four Representative Modes of Experimental Operation in the TFTR Device	2-4
2.1-3	Fusion Energy Multiplication Factors	2-7
2.1-4	Schematic Representation of Various Tokamak Reactor Regimes for Plasmas	2-8
2.1-5	Computed Energy Multiplication Factors Q versus Fusion Power Densities for Tokamak Plasmas of Fixed Total Energy Density	2-9
3.1-1	Discharge Initiation	3-2
3.1-2	Computed Discharge Initiation	3-4
3.1-3	Rise of Discharge Current and (Space-Averaged) Plasma Parameters During the Formation Phase of Mode D	3-6
3.1-4	Volt-Seconds-Limited Duration of the ~ 2.5 -MA Flat-Top Regime for Various Average Discharge Temperatures and β_p -Values	3-7
3.1-5	Discharge Equilibria Computed By a 2-Dimensional MHD Code for a Representative High-Current Case (Top), and Weak- Compression Case (Bottom)	3-10
3.1-6	Comparison of Typical Anomalous-Transport Regimes, As Simulated By the Transport Code of Section 3.2, for Various Present-Day and Future Tokamaks	3-12
3.1-7	Schematic Comparison of Characteristic Plasma-Physical Mechanisms in Small Present-Day and Large Future Tokamaks	3-12
3.1-8	Fractional Confinement of 3.5 MeV D-T Alpha Particles Produced By a Source $S(\alpha)$ in the TFTR 1-MA Discharge	3-14
3.1-9	Effect of Plasma Impurities on Neutral Beam Deposition	3-18
3.1-10	Neutral Beam Injection Geometry	3-20
3.1-11	Recent ATC Experimental Results Relevant to TCT Operations	3-24
3.2-1	Schematic of 6-Regime Anomalous Transport Model	3-28
3.2-2	The Mean Ionic Z and the Total Radiation for FE Ions in Coronal Equilibrium	3-30
3.2-3	Ohmic-Heated Discharges with $I = 2.5$ MA and Various Values of Central Density	3-33

LIST OF ILLUSTRATIONS (Continued)

Figure		Page
3.2-4	Weak-Compression Modes with Various Intervals of Neutral-Beam Injection	3-36
3.2-5	Radial Profiles for the Reference Modes	3-38
3.2-6	Strong-Compression Modes with Various Intervals of Neutral-Beam Injection	3-41
3.2-7	Effect of Varying the Thermal and Particle Diffusion Co- efficients of the Trapped-Particle Modes by Factors of 5 Relative to the Reference Calculation for Mode A	3-43
3.2-8	Effect of Varying the Thermal and Particule Diffusion Co- efficients of the Trapped-Particle Modes by Factors of 5 Relative to the Reference Calculation for Mode B	3-44
3.2-9	Low-Current Quasi-Static Injection Modes	3-46
3.2-10	Discharge Modes with 2.5 MA Currents	3-48
3.3-1	Classical Slowing-Down of an Energetic Deuteron in a 5-keV Tritium Plasma	3-50
3.3-2	Energetic-Ion Velocity Distributions (a) Parallel and (b) Per- pendicular to the Magnetic Axis	3-50
3.3-3(A)	Comparison of Fusion Power Density for a Two-Energy Component Plasma with Optimal $\bar{T}$ and a 50% D, 50% T Thermal Plasma	3-54
3.3-3(B)	Reduction in the Maximum Attainable Fusion Power Density P_f and the Corresponding Power Multiplication Q_0 for 200-keV Deuterons	3-54
3.3-4	(A) Optimal Beam Energies for Maximum Energy Multiplica- tions. (B) Energy Multiplaction Factors	3-55
3.3-5	Alpha Energy/Bulk-Plasma Energy in an Ignited D-T Plasma	3-60
3.3-6(A)	After Formation of the Initial Ohmic-Heated Plasma, Neutral Beams Are Injected for Heating to 6-9 KeV	3-62
3.3-6(B)	Fusion Energy Multiplication Q for a Burn that Propagates Outward from Core Plasma of Radius a_0 and Is Terminated When the Burn Front Reaches a Radius a_m	3-62
3.3-6(C)	Fraction of the Final Energy Content Supplied By Injected Neutral Beams	3-62

LIST OF ILLUSTRATIONS (Continued)

Figure		Page
4.1.2-1	Weak Compression (Mode A)	4-12
4.1.2-2	High-Current Mode D	4-15
4.1.3-1	Neutron Flux Within the Test Cell and Through the Concrete Walls in the Equatorial Plane	4-20
4.1.3-2	Gamma Ray Flux Within the Test Cell and Throughout the Concrete Walls in the Equatorial Plane	4-21
4.1.3-3	Dose Rates from a Large 305 Stainless Steel Object Located Just Outside the Device Shield	4-21
4.1.3-4	Dose Rates from Back End of Neutral Beam Injector	4-23
4.1.3-5	Dose Rates from the TFTR Vacuum Vessel	4-23
4.1.4-1	Location of the Turns for the Composite EF/OH Winding Relative to the TF Coils for the TFTR Conceptual Design	4-26
4.1.4-2	Magnetic Field Strength in Horizontal Symmetry Plane	4-27
4.1.4-3	Ripple Contours in a Vertical Plane	4-28
4.1.4-4	Contours of Constant OH Magnetic Field Intensity (T)	4-29
4.1.4-5	EF Index of Curvature Contours	4-30
4.1.5-1	EF/OH/Plasma Circuit	4-32
4.1.5-2	Currents in the EF & OH Coils and Plasma as a Function of Time for 2.5 MA Plasma	4-34
4.1.5-3	EF and OH Instantaneous Power Requirements for 2.5 MA Plasma	4-35
4.1.5-4	Power Requirements for 1.0 MA Plasma During Weak Compression Experiment	4-37
4.1.6-1	TF Coil Design Configuration Analyzed	4-40
4.1.6-2	TF, OH and EF Field Current Profiles for Thermal Analysis	4-42
4.1.6-3	Material Temperature of Hottest Turns	4-42
4.1.6-4	Coolant Temperatures	4-43
4.1.6-5	TF Coil Thermal Operating Limits	4-43
4.1.6-6	Peak Protection Plate Temperature vs. Neutral Beam Flux	4-47
4.1.6-7	Peak Surface Temperature vs. Neutral Beam Flux and Number of Pulses	4-47
4.1.6-8	Maximum Neutral Beam Flux at the Protection Plate	4-49
4.1.7-1	Toroidal Field Coil Slope Versus Coil Angular Location	4-51
4.1.8-1	Schematics of TFTR Test Cell Air Treatment Hookups	4-63
4.1.10-1	Load Sequence — No Compression	4-72
4.1.10-2	Load Sequence — Weak Compression	4-73
4.1.10-3	Load Sequence — Strong Compression	4-74

LIST OF ILLUSTRATIONS (Continued)

Figure		Page
4.1.11-1	TFTR Facility Complex	4-77
4.1.13-1	Neutral Beam Injector Duct.	4-92
4.1.15-1	Average TF Temperature Reached vs. Cycle, Cycle Duration and Flat-Top Time	4-104
4.1.15-2	Comparison of TF Coil Coolant Temperatures	4-105
4.1.15-3	Comparison of Coil Temperatures by Two Cooling Schemes	4-105
4.1.15-4	Comparison of Coolant Temperatures by Two Cooling Schemes	4-106
4.1.15-5	Power System Cost vs. Reactance	4-107
4.1.15-6	Alternative Structural Systems for MG Building Cost Study	4-108
4.2-1	TFTR Device	4-109
4.2.1-1	Vacuum Vessel	4-112
4.2.1-2	Welded Metal Bellows	4-114
4.2.2-1	Device Structure	4-120
4.2.2-2	Device Shielding	4-122
4.2.2-3	Water Cooling Headers	4-123
4.2.2-4	Alternate Shielding Arrangement	4-125
4.2.2-5	Alternate Shielding Arrangement	4-125
4.2.3-1	Toroidal Field Coil Assembly	4-127
4.2.3-2	TF Coil Nose Cross Section	4-128
4.2.4-1	Poloidal Field Coil Assemblies	4-133
4.2.4-2	Band Tensioning Device for Poloidal Field Coils	4-134
4.2.4-3	Multilam Connector	4-134
4.3.1-1	Single Line Diagram	4-141
4.3.1-2	TFTR Motor Generator Power Supply Complex	4-143
4.3.2-1	Reference Design of the TF Pulsed Energy Conversion System (SD-4D)	4-147
4.3.3-1	Reference Design of OH Pulsed Energy Conversion System (SD-4C)	4-151
4.3.4-1	Reference Design of the EF Pulsed Energy Conversion System (SD-4E)	4-154
4.3.5-1	Control Room	4-157
4.3.5-2	Computer and Data Storage Area	4-158
4.3.5-3	Central Control and Monitor System	4-159
4.3.5-4	Central Control and Monitor System (For Gas and Tritium Cleanup Systems)	4-159
4.3.5-5	Central Control and Monitor System (For Torus Vacuum, Machine Area Water Cooling, and Aux. Facility Systems)	4-160

LIST OF ILLUSTRATIONS (Continued)

Figure		Page
4.3.5-6	High Resolution Timer	4-161
4.3.5-7	Safety and Protection System	4-163
4.3.5-8	High-Speed Data Acquisition System	4-165
4.3.5-9	Data Reduction and Display System	4-167
4.3.5-10A	Computer Room	4-170
4.3.5-10B	Control Room	4-171
4.3.5-10C	Control Area	4-171
4.4.1-1	Experimental Area - Tokamak Fusion Test Reactor	4-174
4.4.1-2	Overhead Manipulator and Inspection System	4-176
4.4.1-3	Experimental Area Alternates (Elevation Views)	4-178
4.4.2-1	Schematic of Tritium Storage and Delivery System	4-184
4.4.2-2	Schematic of Interfacing of Tritium Storage and Delivery System with Vacuum Vessel and Vacuum System	4-185
4.4.3-1	Simplified Schematic Tritium Cleanup System	4-192
4.4.3-2	Tritium Cleanup Campaigns — Temporal Variation of Activity Load	4-193
4.4.3-3	Tritium Cleanup Campaign — Operation Time Versus Cleanup Achieved	4-193
4.4.4-1	Schematic of Typical Hookup for Each Array for Non- Tritium Gas Storage and Delivery	4-197
4.4.5-1	Schematic Arrangement of Machine Area Water Cooling System	4-202
4.4.5-2	Field Coil Cooling Loop Demineralization System Piping Schematic	4-206
4.4.6-1	Torus Vacuum Pumping System Showing a Pair of High Vacuum and Backing Systems Plus Tritium Cleanup and Monitoring Systems	4-214
4.5-1	Plan of Four (ORNL) Beam Lines	4-220
4.5-2	Elevation of ORNL Beam Line	4-222
4.5-3	ORNL Ion Source and Accelerator Assembly	4-225
4.5-4	Typical Section Through ORNL Cryosorption Panels	4-226
4.5-5	Typical Target Section of ORNL Calorimeter Assembly	4-227
4.5-6	Complete Two-View of LBL/LLL Beam Line	4-230
4.5-7	Computer Beam Trajectories Through Proposed LBL/LLL Accelerator Structure	4-232

LIST OF ILLUSTRATIONS (Continued)

Figure		Page
4.5-8	Three-View of LBL/LLL Magnet	4-233
4.5-9	Two-View of LBL/LLL Calorimeter	4-236
4.5-10	Simplified Schematic of the Conceptual Design	4-239
4.6-1	Conceptual Layout of Diagnostics in Test Cell (Oct. 75)	4-244
4.6-2	Alternate Scheme for the X-Ray Pulse Height Analysis Sys- tem Showing Tubes Passing Through Coils and Structure	4-246
4.6-3	Speculative Arrangement of Laser, Windows, and Light Collecting System for Thomson Scattering	4-247
4.6-4	Potential Alternate Arrangement of Movable Limiter Blades	4-249
4.7-1	TFTR Complex — Floor Plan at Level-16'	4-253
4.7-2	TFTR Complex — Floor Plan at Level 0'	4-254
4.7-3	TFTR Complex — Floor Plan at Level 32'	4-255
4.7-4	TFTR Complex — Sections A-A and B-B	4-256
4.7-5	TFTR Complex — Sections C-C, D-D and E-E	4-257
4.7-6	TFTR Complex — West and North Elevations	4-258
5.1-1	Princeton Plasma Physics Laboratory Organization	5-2
5.1-2	TFTR Project Organization	5-3
5.4-1	TFTR Level I Master Schedule	5-13

LIST OF TABLES

Table		Page
3.2.2-1	Representative Parameters of 2.5-MA Discharges at Time = 650 MSEC.	3-35
3.2.3-1	Parameters of the Reference Operating Modes	3-39
3.3.1-1	Parameters of TCT-Type Reactors	3-57
3.3.2-1	Plasma Parameters After Neutral-Beam Heating and Major-Radius Compression	3-63
4.1.1-1	TFTR System Design Summary	4-2
4.1.1-2	Summary of TFTR System Design Variations	4-5
4.1.2-1	Plasma Parameters	4-11
4.1.3-1	Doses Delivered Around TFTR Site B6 20 MW, 0.5 SEC D-T Pulses	4-19
4.1.6-1	Cooling Water System Design Summary	4-38
4.1.6-2	TF, OH and EF Coils Thermal Design Summary	4-39
4.1.6-3	Summary of OH and EF Coil Cooling Designs	4-45
4.1.7-1	A Parameter Comparison for the TF Coil Lateral Deflection (Slope) Sensitivity Analysis	4-50
4.1.7-2	A Parameter Comparison for the TF Coil Combined Stress and Load Sensitivity Analysis	4-54
4.1.8-1	Tritium Related Design Options	4-59
4.1.8-2	Tritium Systems Design Summary	4-68
4.1.10-1	Summary Generator Load List	4-75
4.1.11-1	Inventory of TFTR Space Requirements	4-79
4.1.11-2	Design Requirements, TFTR Facility Buildings	4-84
4.1.11-3	Design Requirements, Facility Services	4-85
4.1.12-1	Proposed TFTR Diagnostic Systems	4-88
4.1.14-1	Central Control, Timing and Monitor Requirements	4-95
4.1.15-1	TFTR Electronic Power Loads (MW) (For a One-Second Flat- Top, 2.5 MA Plasma Experiment)	4-100
4.1.15-2	TFTR Electric Energy Loads (MJ)	4-101
4.1.15-3	TFTR Electric Energy Load Variations (MJ) (Additions as Compared to Nominal)	4-101
4.1.15-4	System Temperatures vs. Ballast Tank Volume (1 Sec. Flat- Top, 300-Second Cycle)	4-102
4.1.15-5	Flywheel Droop and Recovery Time vs. Flat-Top Duration	4-102
4.1.15-6	Cost Comparison for Alternate Structural Systems Motor Generation Building	4-110

LIST OF TABLES (Continued)

Table		Page
4.2.3-1	Toroidal Field Coil Specifications	4-130
4.2.4-1	Ohmic Heating Coil Specifications	4-135
4.2.4-2	Equilibrium Field Coil Specifications	4-135
4.3.1-1	Major Auxiliary Electrical Loads	4-140
4.3.1-2	ESDS Equipment Ratings	4-144
4.3.2-1	Toroidal Field Pulsed Energy Conversion System Summary of Parameters	4-148
4.3.3-1	Ohmic Heating Pulsed Energy Conversion System Summary of Parameters	4-151
4.3.4-1	Equilibrium Field Pulsed Energy Conversion System Summary of Parameters	4-154
4.3.5-1	DAS Computer System Summary	4-164
4.3.5-2	DRDS Computer Systems Summary	4-167
4.3.5-3	Summary of Radiation Monitoring System Characteristics	4-168
4.3.5-4	Summary Equipment List for Radiation and Tritium Monitors	4-169
4.3.5-5	List of Equipment to Be Placed in the Central Control Room . .	4-170
4.4.2-1	Tritium Storage and Delivery System Design Summary	4-188
4.4.3-1	Cleanup Systems Principal Features and Design Parameters . .	4-192
4.4.4-1	Non-Tritium Charge Gas and Delivery System Design Summary .	4-198
4.4.5-1	Summary of Principal Specifications of Machine Area Water Cooling System Components	4-208
4.4.5-2	Machine Area Water Cooling System Operating Parameters . . .	4-210
4.4.6-1	Torus Vacuum Pumping System Design Summary	4-215
4.5-1	Operating Data	4-224
5.3-1	TFTR PACE Estimate	5-10
5.3-2	TFTR Buildings and Equipment Cost Summary	5-11
5.4-1	TFTR Project Milestones	5-16

1.0 INTRODUCTION

This document describes the final conceptual design of the Tokamak Fusion Test Reactor (TFTR), the first U.S. magnetic confinement device planned to demonstrate the fusion of deuterium and tritium at reactor level power densities. As such, it represents an essential link between the large hydrogen machines of the tokamak type presently being designed and constructed in the United States (e.g., PT, PDX, Doublet III) and the first experimental fusion power reactor.

Clearly, the production of a significant number of fusion reactions while operating with a deuterium-tritium plasma requires the introduction of new technology in the design, fabrication, and operation of tokamaks in such areas as tritium handling, servicing of activated components and radiological safety. New dimensions are added to the already difficult problems of experimental diagnostics, vacuum technology and neutral-beam technology. A fundamental concern in the planning of the enterprise at this time is the proper choice of objectives and scope to assure that significant contributions will be made, both scientifically and technologically, to the design, construction, and operation of a fusion experimental power reactor.

This report addresses the physics objectives and the engineering goals of the TFTR project. The physics objectives must be considered paramount in the ordering of technical priorities. The basic scientific purpose of the next generation of large tokamaks—the TFTR, the European JET, the Soviet T-20, and the Japanese JT-60—will be that of coming sufficiently close to reactor plasma conditions to predict, with reasonable confidence, the physical size and gross parameters required for tokamak-type power reactors. TFTR represents such a significant extrapolation from present-day tokamaks, in terms of plasma temperature, physical size, neutral-beam auxiliary heating, etc., that its construction would be justified even if it were to be operated only as a hydrogen system, i.e., without D-T fueling. Indeed, the early stages of operations required for establishing the nature, limitations and optimal modes will involve mainly hydrogen discharges. To implement fully this aspect of TFTR research capabilities, strong emphasis must be placed on experimental and operational flexibility. Access must be provided for a full complement of plasma measurements; control must be provided to vary experimental parameters over a wide range. In D-T test-reactor operations, the fundamental goal will be to produce fusion power density levels about 1 watt/cm^3 , using a tritium plasma, into which an energetic neutral beam of deuterium is injected. The additional complication of operations with D-T in the hostile environment produced by the 14-MeV neutrons associated with the production of fusion power makes the TFTR an excellent model for the prototypical study of many technological aspects of a future power reactor. In addition to the scientific objectives, there are important

engineering goals for TFTR; principally, the acquisition of experience with the design, construction, and operation of an integrated "prototypical" fusion power system, including comprehensive system studies to ensure reliability and maintainability.

This Final Conceptual Design Report, dated February, 1976, is the second complete conceptual design report and represents an improved design over that described in the first report, issued in June, 1974. This document contains a more complete development of the technical, cost, and schedule aspects of the project than earlier reports.

Project goals are defined within the boundary conditions of cost and schedule. The present projections are a base hardware and building cost of \$228 million and a completion date of 1981. The TFTR facility will be located on the James Forrestal Campus of Princeton University.

Most of the work contained in this report was performed by the Princeton Plasma Physics Laboratory (PPL) under Princeton University Contract E(11-1)-3073 with the Energy Research and Development Administration (ERDA) and by the Westinghouse Electric Corporation under its contract #192 with the University.

2.0 SCIENTIFIC MOTIVATION

The rapid progress achieved in tokamak research in recent years has provided the scientific and engineering data required for construction of large, next-generation tokamak facilities. The goal of these new devices is to achieve reactor-grade plasmas and significant DT fusion reaction rates. Section 2.1 discusses the experimental background and defines the major scientific objectives of the TFTR project. This section also identifies main problems anticipated in fulfilling the scientific objectives and presents an outline of the experimental strategies for solving these problems in TFTR. Section 2.2 summarizes TFTR device parameters and capabilities.

2.1 EXPERIMENTAL BACKGROUND

In the period 1967-1969 the T-3 tokamak device at the I. V. Kurchatov Institute in Moscow demonstrated the attainment of peak electron temperature $T_e \sim 1000$ eV, ion temperatures $T_i \sim 400$ eV, and densities $n \sim 5 \cdot 10^{13} \text{ cm}^{-3}$, with $n\tau_E$ -values (product of density and energy confinement time) of order $5 \cdot 10^{11} \text{ cm}^{-3} \text{ sec}$. These combined results represented a dramatic improvement relative to previous accomplishments in controlled fusion research. The favorable scaling laws observed for the dependence of $n\tau_E$ on plasma temperature, density and size confronted the fusion research community for the first time with a concrete prospect of attaining reactor-like plasma temperatures $T_e, T_i \sim 5 - 10$ keV and reactor-like confinement parameters $n\tau_E \sim 10^{13} - 10^{14} \text{ cm}^{-3} \text{ sec}$, in plasma experiments of economically realistic size.

Subsequent tokamak confinement experiments on the ST device at the Princeton Plasma Physics Laboratory, the ORMAK at the Oak Ridge National Laboratory, the T-4 at the Kurchatov Institute, and the TFR at the Centre d' Etudes Nucléaires at Fontenay-aux-Roses, as well as specialized experiments on many smaller facilities, served to confirm the basic results of the T-3 experiment. The tokamak parameter range has been extended to somewhat higher peak values ($T_e \sim 3$ keV, $T_i \sim 1$ keV, $n \sim 10^{14} \text{ cm}^{-3}$, $n\tau_E \sim 2 \cdot 10^{12} \text{ cm}^{-3} \text{ sec}$), and a much better understanding has been gained of plasma stability and transport processes. Predictions (Section 3.2) based on present theory are considerably more cautious than the simple empirical extrapolations of the T-3 results. Even the "pessimistic" present-day predictions, however, point to the attainment of reactor-like plasma confinement in devices of the magnitude of the TFTR, and to the ultimate feasibility of tokamak fusion power reactors. Figure 2.1-1 shows a simplified schematic of the tokamak confinement configuration.

The tokamak plasma benefits from an inherent ohmic heating process: the same plasma current I that is needed to generate the poloidal confining B_p (Figure 2.1-1) serves to

Figure 2.1-1. The Tokamak Magnetic Field Configuration.

A strong toroidal magnetic field component plus a weak poloidal field component, generated by the plasma current produce helical magnetic field lines covering nested toroidal flux surfaces. A vertical component of the poloidal field is generated by external coils; the force that this field component exerts on the plasma current provides the plasma with major-radius equilibrium. (PPL 753898)

heat the plasma electrons, typically to the range of a few keV. The plasma ions are then heated by collisional thermal equilibration with the electrons. Reliance on ohmic heating alone appears almost certainly inadequate for the achievement of tokamak reactor conditions. During the past several years, therefore, the application of auxiliary tokamak heating techniques has been studied intensively. A technology of extremely high-powered neutral atom beams has been developed at the Oak Ridge National Laboratory and at the Lawrence Berkeley Laboratory; this achievement has made possible the demonstration of direct ion heating in present-day tokamaks, and the specification of realistic neutral-beam heating systems for tokamaks of the magnitude of the TFTR. The effectiveness of several high-frequency wave-heating methods has also been tested experimentally, and this remains a significant supplementary approach. A third new development in tokamak heating has been the demonstration of adiabatic compression (by plasma displacement in major radius) on the ATC device at Princeton. In recent ATC experiments, initial ohmic-heated plasma parameters $T_i \sim 200$ eV, $n \sim 2 \cdot 10^{13} \text{ cm}^{-3}$ were raised to $T_i \sim 400$ eV, $n \sim 4 \cdot 10^{13} \text{ cm}^{-3}$ by injection of a 230 kW, 15 keV deuteron beam, and to $T_i \sim 1200$ eV, $n \sim 2 \cdot 10^{14} \text{ cm}^{-3}$ by subsequent adiabatic compression. No plasma-related upper limit to this heating process has been encountered thus far; we expect that the ATC techniques will extrapolate straightforwardly to the achievement of reactor-like plasmas in tokamaks of TFTR size.

2.1.1 ATTAINMENT OF REACTOR-LIKE PLASMAS

The design of the TFTR device has been guided by four principal interrelated objectives:

- a. The attainment of $T_e, T_i \sim 5 - 10$ keV in substantially pure hydrogenic plasmas.
- b. The maximization of plasma density within the constraint of grossly stable confinement.
- c. The realization of plasma energy confinement times corresponding to $n \tau_E \sim 10^{13} \text{cm}^{-3} \text{sec}$, and the technical capability of attaining $n \tau_E \sim 10^{14} \text{cm}^{-3} \text{sec}$ under favorable circumstances.
- d. The generation of reactor-like fusion power densities under conditions of approximate energetic break-even, in DT operation.

Figure 2.1-2 presents the basic device parameters envisaged for TFTR, and illustrates several reference modes of plasma operation. The appropriateness of this design to the attainment of objectives 1-3 is reviewed in the present section. Attainment of the fourth objective — which follows from the first three — is discussed further in Section 2.1.3.

One major concern is the stability of the plasma. The tokamak configuration must, first of all, be given a high degree of gross stability against magnetohydrodynamic (MHD) disturbances. Thus, the plasma current strength must be limited by an appropriately high MHD safety factor, $q = 2\pi/\iota = aB_t/RB_p = a^2B_t/2Rl$ where B_t is the toroidal magnetic field, (see Figure 2.1-1), and that the plasma density then limited by $\beta_p = 8\pi n(T_e + T_i)/B_p^2$ ($a < R/a$). A second major concern is the quality of energy confinement in the presence of various small-scale plasma transport mechanisms. To reach a specified $n\tau_E$ -value, it is desirable to maximize the plasma current I and density n . Within the given MHD constraints, these parameters are raised at the expense of increases of device size and toroidal magnetic field strength ($I \propto RB_t$; $n \propto B_t^2$). An alternative approach to raising these parameters is to stretch the tokamak cross section into various vertically elongated shapes. The third major concern is the avoidance of a substantial migration of impurity ions into the hot central plasma region. The outflow of plasma energy to the vacuum wall — particularly in the form of energetic plasma ions or charge-exchange neutrals — is expected to sputter surface contaminants, as well as wall atoms. The resultant impurity influx can be reduced, at least transiently, by positioning the hot plasma region away from direct wall contact. For long-pulse operation, an appropriate alternative approach to the tokamak impurity problem is the development of a magnetic divertor system.

These three major problem areas of the tokamak — gross stability, energy confinement, and plasma purity — are addressed directly by the TFTR design. In order to permit confinement of plasma densities of $n \sim 10^{14} \text{cm}^{-3}$, the toroidal magnetic field has been raised to the maximum technologically convenient value of 9.5T at the coil bore, which

Figure 2.1-2. Four Representative Modes of Experimental Operation in the TFTR Device. Programmed compression in major radius can be used to control the initial skin effect, energize the bulk plasma particles and injected ions, and remove the hot plasma core from direct wall contact. (PPL 753877)

corresponds to 5-6T at the typical plasma locations. The ratio of major to minor radius of the vacuum vessel (2.4), has been made as small as possible consistent with the mechanical accommodation of the toroidal field (TF) coils and the transformer primary (OH) windings, which induce the plasma current. As a result, a plasma limiter radius $a = 85$ cm is accommodated at the 5.2T location, permitting a substantial safety factor $q = 7.5$ to be used for a plasma current of $I = 1$ MA. The TFTR device is also capable of operating at higher plasma currents, ranging up to 2.5 MA, which correspond to safety factors ranging down to $q = 3.0$.

The ultimate $n\tau_E$ -values that will be achieved in a tokamak of these capabilities cannot be predicted with accuracy at the present time. The currently beginning T-10 and PLT experiments (located at the Kurchatov Institute and Princeton Plasma Physics Laboratory, respectively) will provide an initial test of high-temperature transport theory. A precisely relevant confinement parameter regime, however, will be encountered only by the TFTR device itself: the detailed measurement of the corresponding transport rates is one of the principal objectives of the experiment. Prevalent estimates of the $n\tau_E$ -values likely to be reached for reactor-like plasmas in tokamaks of TFTR size fall into the range 10^{13} to 10^{14} cm⁻³sec. The lower end of this range corresponds to operation at the conservative 1-MA level, and to predictions of the pessimistic transport code employed in the TFTR design study. The upper end corresponds to high-current operation and to more optimistic assumptions about high-temperature transport.

The principal heating method for TFTR is neutral-beam injection, selected on the basis of its well-proven effectiveness, and because of its capability of maximizing the fusion power density (cf. Section 2.1.3). The heating power provided for the TFTR is based on the assumption of a 1-MA discharge with $n \sim 10^{14}$ cm⁻³ and $n\tau_E \sim 10^{13}$ cm⁻³sec, which calls for the relatively high injection capability of $\gtrsim 20$ MW in order to approach the limiting β_p -value of $R/a \sim 3$. An energy of $\gtrsim 120$ keV is selected for the primary deuteron beam, in order to give adequate penetration into the plasma core. These beam parameters are also suitable for the higher-current operation, provided that the plasma density is lowered (to $\sim 5 \cdot 10^{13}$ cm⁻³ in the 2.5 - MA case) to facilitate beam penetration and to permit the larger plasma volumes of the higher-current discharges to be heated to the desired temperature range with the available beam power. The 1-MA current level is satisfactory for the confinement of 120-keV deuteron orbits, but represents a lower bound in this respect. The 120-keV energy represents a lower bound for purposes of the DT reactor experiments discussed in Section 2.1.2.

The primary objectives of the TFTR experiment call for the achievement of reactor-like plasmas, but not for the maintenance of these conditions during times that correspond to large multiples of the $n\tau_E$ -value. Accordingly it is possible to make use of the inherent time delay of the impurity influx in large tokamak devices. In the case of transient neutral-beam heating of 1-MA discharges, the hot plasma core can ultimately be removed from direct wall contact, and impurity problems remain minimal. In the case of hot plasmas that fill the limiter aperture, the relatively pessimistic assumptions about impurity influx made by the transport code discussed in Section 3.2 lead ultimately to catastrophic impurity levels, but even in these modes of operation the plasma typically passes through regimes of reactor-like temperature.

Since the basic objectives of the TFTR experiment with respect to temperature, confinement, and purity can be met straightforwardly by the standard tokamak plasma configuration, the TFTR poloidal field system has not been designed to stretch the minor cross section of the plasma or to provide a divertor. These promising refinements of the tokamak configuration will be tested, respectively, by the Doublet III device at General Atomic Company and the PDX at Princeton. The TFTR will use the standard type of poloidal equilibrium field (EF); that is to say, a principally vertical magnetic field, with a mild outwardly convex curvature, adjustable within the range where both horizontal and vertical axisymmetric displacements are positively stable. A special feature of the TFTR plan is development of new hot-plasma handling techniques that exploit the standard tokamak poloidal field configuration in order to optimize various plasma parameters. Four basic operating modes are illustrated in Figure 2.1-2:

- a. The objective of the weak-compression mode is the realization of transient high-purity, high-temperature conditions. In order to control the plasma current density profile, avoiding skin effect or excessive central peaking, a suitably programmed vertical magnetic field (EF) is used to shift the discharge equilibrium position away from an outer limiter as the discharge current rises. The plasma is then heated by neutral-beam injection and is finally compressed in major radius (by a factor $C \sim 1.2$) through a further increase of the vertical field. The impurity control provided by removing the plasma from direct wall contact can be maximized by filling the intervening region with a cold, dense plasma. The hot-plasma core and the energetic-ion component are energized by the radial compression; electron and ion temperatures in excess of 5 keV are easily reached, and continue to rise after compression, as the energetic-ion component thermalizes. Detailed plasma parameter evolutions for this and the other operating modes will be illustrated in Section 3.2.
- b. The strong compression case is designed to achieve transient regimes with maximal plasma temperatures and densities. The operating cycle begins in the same way as for Mode A, but the radial compression factor is larger ($C \sim 1.5$), leading to compressed-plasma temperatures above 10 keV. The onset of impurity contamination, however, is more rapid than in Mode A.
- c. The transient-heating Modes A and B can be supplemented by quasi-static heating studies, if the plasma is neutral-beam injected for an extended period following compression. Quasi-static injection into a weakly compressed plasma is illustrated in Figure 2.1-2.
- d. The high-current mode is designed to take advantage of favorable conditions of MHD-stability and impurity influx to reach very large $n\tau_E$ -values and high-temperature regimes of extended duration, maintained by neutral-beam heating. Temperatures in the range 5-10 keV are reached for plasmas with $5 \cdot 10^{13} \text{cm}^{-3}$ density, but impurity contamination eventually stops effective neutral-beam penetration.

The transient Modes A and B, and intermediate cases, are also referred to as Mode I. The quasi-static Modes C and D, and intermediate cases, are referred to as Mode II.

The final plasma locations in Figure 2.1-2 are off-centered with respect to the vacuum vessel, in order to minimize the spatial inhomogeneity of the toroidal field. As a result, the toroidal "coil ripple" is below $\pm 5 \cdot 10^{-3}$ at the plasma edge and below $\pm 5 \cdot 10^{-5}$ at the center of the hot-plasma region.

2.1.2 ATTAINMENT OF REACTOR-LIKE FUSION POWER DENSITIES

In neutral-beam heating, the injected particles pass freely through the confining magnetic field, then are ionized by plasma particles and trapped within the plasma volume, where they slow down gradually, imparting their energy to the bulk plasma by Coulomb scattering. During this slowing-down process, the injected ions also have exceptionally large cross sections for undergoing fusion reactions with the bulk plasma ions. A remarkable consequence is that, in a tritium plasma of $T_e \gtrsim 5$ keV, maintained by injection of a deuteron beam in the energy range $W_0 \gtrsim 120$ keV, the fusion power that results from the reaction of the "hot" and "cold" ion components exceeds the injection power. As shown in Figure 2.1-3, the magnitude of the energy multiplication factor Q attained with this type of two-component reaction can be made several times greater than unity, depending mainly on the temperature of the bulk plasma electrons, which control the slowing-down rate of the injected deuterons.³

Figure 2.1-3. Fusion Energy Multiplication Factors.

Q = Number of DT reactions $\times 17.6$ MeV/ W_0 for tritium plasmas of various temperatures $T_e = T_i$ and deuteron injection energies W_0 . Enhancements of Q due to finite triton temperature and deuteron-energy diffusion are taken into account. Reactions of deuterons at energies that have fallen below $2T_i$ are not included. (PPL 753884)

The $n\tau_E$ -value required for energy break-even in a two-component torus (TCT) reactor is identical with the requirement that neutral-beam heating succeed in maintaining the plasma electron temperature at $T_e \gtrsim 5$ keV. The TCT $n\tau_E$ -value is therefore closely related to the slowing-down time, τ_s , of the energetic deuterons; in the range $W_0 \gtrsim 120$ keV, this is typically characterized by $n\tau_s \sim 10^{13} \text{ cm}^{-3} \text{ sec}$. The confinement time for the energetic-ion component evidently cannot be allowed to fall much below τ_s , since otherwise the injected-ion energy is lost before it heats the plasma or can be regenerated as fusion energy. As to the bulk-plasma $n\tau_E$ -value, it is readily seen that, if τ_E is much shorter than τ_s , the energy stored in the injected particle component would have to be much greater than the total bulk-plasma energy in order to maintain the plasma temperature. Consequently, the appropriate bulk-plasma $n\tau_E$ -value for neutral-beam heating — as well as for implementation of the TCT regime — is $n\tau_E \gtrsim 10^{13} \text{ cm}^{-3} \text{ sec}$.

The relationship between the TCT and some other toroidal reactor regimes is illustrated qualitatively in Figure 2.1-4, for the representative case of plasmas in the range $T_e, T_i \gtrsim 6$ keV, heated by deuteron beams of energy $W_0 \gtrsim 120$ keV. In every case, the use of a fusion-reaction-generating beam to introduce the heating power is of some benefit in enhancing the system Q -value.⁴ For $n\tau_E$ values around $10^{13} \text{ cm}^{-3} \text{ sec}$, this advantage is so overriding that the highest Q -values are obtained with pure tritium plasmas, which contribute hardly any bulk-plasma fusion power. In the $n\tau_E$ -range $3 \cdot 10^{13}$ to $3 \cdot 10^{14} \text{ cm}^{-3} \text{ sec}$, bulk-plasma reactions can begin to effect significant increases of the Q -value, and the

Figure 2.1-4. Schematic representation of various tokamak reactor regimes for plasmas in the temperature range $T_e, T_i \gtrsim 6$ keV, heated by beams of $W_0 \gtrsim 120$ keV. The DT mixture is assumed to be optimized for maximum Q at each $n\tau_E$ -value. The energy density ratio $\bar{\Gamma}$ of superthermal ions (injected ions plus fusion-generated alpha particles) relative to bulk plasma particles is indicated. (PPL 753889)

optimal plasma composition then shifts towards 50:50 DT. At the upper end of this range, the alpha particles of the DT-reaction take over the plasma-heating function, so that injected-ion reactions become insignificant and the system Q -value becomes very large. The ratio $\bar{\Gamma}$ of the energy densities of the "hot" and "cold" ion components is typically 0.1–0.5 (including the alpha particle contribution) in the high- $n\tau_E$ range, and rises to 0.5–1.0 in the TCT range. As Figure 2.1-4 indicates, it is possible to maintain the condition $Q \sim 1$ down to even lower bulk-plasma $n\tau_E$ -values, by means of further increases of $\bar{\Gamma}$. In these very-high- $\bar{\Gamma}$ regimes, it then becomes advantageous to rely principally on the fusion reactions between counter-streaming energetic ion components.⁵

While the Q -value is an essential figure of merit for fusion reactor schemes, the achievable fusion power density is also important — and can even be of prime importance in certain reactor applications. In Figure 2.1-5, the relation between these two quantities has been computed for a range of plasmas with the same energy density, but with different temperatures and $n\tau_E$ -values, heated by a 200-keV deuteron beam.⁶ The DT-mixture of the bulk plasma has been optimized at each operating point to give maximum Q . As indicated by Figure 2.1-5, the typical TCT regime, around $n\tau_E = 10^{13} \text{ cm}^{-3} \text{ s}$, can provide higher fusion power densities than are attainable in the high- $n\tau_E$, high- Q cases, even at arbitrarily high bulk-plasma temperatures.

Figure 2.1-5. Computed energy multiplication factors Q versus fusion power densities for tokamak plasmas of fixed total energy density 0.34 J cm^{-3} , including the energetic ion component, and various $n\tau_E$ -values and temperatures $T_e = T_i$. The DT mixture is optimized at each operating point for maximum Q . (PPL 753890)

As a candidate for a tokamak fusion test reactor (TFTR) experiment, the TCT is clearly a favored contender; both because of its relatively modest $n\tau_E$ - requirements and because of its ability to produce fully "realistic" fusion power densities, corresponding to (or even exceeding) the levels at which economic tokamak fusion power reactors are expected to operate. Computer studies carried out for the illustrative operating regimes of Figure 2.1-2 give Q-values that are roughly of order unity, and fusion power outputs of some tens of megawatts with total yields of several megajoules. At the present stage of tokamak technology the duty cycle of "DT burns" is, of course, quite low; the time-averaged fusion power output of the TFTR is only a few kilowatts. These considerations define the nature of near-term tokamak reactor experiments. It will be possible to provide a high degree of realism in regard to instantaneous fusion effects, such as alpha-particle phenomena in the plasma; but there will be only a weak capability for producing time-integrated effects, such as neutron damage of wall materials. The TFTR experimental program has been designed accordingly with an emphasis on the diagnosis of fusion plasma effects and a policy of minimizing, rather than maximizing, the time-integrated wall activation, so as to facilitate experimental access. The DT plasma phenomena to be studied include the alpha-particle orbit confinement in the plasma, the resultant alpha contribution of several megawatts to the plasma heating (well above the expected Ohmic contribution), and the possible velocity-space and electrostatic instabilities resulting from the alpha-particle population.

Aside from the contribution to the "physics of burning plasmas," the TFTR will serve as a direct prototype for a type of toroidal reactor where high power density, rather than high Q-value, is the primary figure of merit (cf. Figure 2.1-5). The most immediate of these applications would be to a higher-duty-cycle tokamak engineering test reactor, intended to produce intense neutron irradiation on sample wall materials and blankets.⁷ Extensive studies are currently under way on the economic application of TCT-type reactors for the breeding of the fissile isotopes ^{239}Pu and ^{233}U in fertile blankets (depleted uranium or thorium),⁸ the generation of power in subcritical fusion-fission hybrid reactors,⁸ and the transmutation of the long-lived actinide waste products of fission reactors.⁹

2.2 SUMMARY OF PRINCIPAL CAPABILITIES OF THE TFTR DEVICE

a. Basic Device Parameters

The standard limiter aperture is $a = 85$ cm, at $R = 248$ cm, where $B_t = 5.2\text{T}$. These numbers correspond to an aspect ratio $R/a = 2.9$, and to MHD safety factors in the range $7.5 - 3.0$ for the available range of plasma currents $I = 1.0 - 2.5$ MA. The standard discharge duration is 1 s; this can be extended to 4 s, with $B_t = 4.5\text{T}$.

b. Special Features

- The applied poloidal field (EF) has a uniform curvature index, adjustable in the range 0.0 to 0.4. The EF can effect major-radius compressions

up to $C = 1.5$ for 1-MA plasmas (corresponding to adiabatic multiplication factors of 1.7 for the temperature, 2.25 for the density, and as much as 2.25 for the energy of injected ions). The poloidal field will have an initial multipole null for discharge start-up, but no provision for cross-sectional shaping or divertor action.

- The plasma current profile will be controlled by a programmed plasma displacement away from a fixed limiter during initial current rise.
- Auxiliary heating is provided in the form of neutral beams containing 20 MW of 120 keV deuterons, accompanied by 12 to 15 MW of half-energy deuterons. The beam injection angle is variable, and the injection time ranges up to 0.5 s.
- The discharge can be removed from limiter contact and enveloped in a cold-plasma protective blanket generated by a pulsed-gas input.

c. Expected Plasma Parameters

Typical central plasma parameters are computed to be $T_e, T_i = 5 - 10$ keV and $n = 10^{14} \text{ cm}^{-3}$. The corresponding β_p -values lie in the range 1-3, and the $n\tau_E$ -values exceed $10^{13} \text{ cm}^{-3} \text{ sec}$.

In typical operation with tritium plasmas and deuterium beams, fusion power densities of $\sim 1 \text{ watt cm}^{-3}$ are obtained, with Q -values in the range 0.4-1.4 and total energy releases of 1-4 MJ. The fusion power released in alpha particles ranges up to 5 MW; in discharges having $I \geq 1$ MA, over 80% of the alphas can be confined within the limiter aperture.

d. Experimental Capabilities

- The neutral beam power is sufficient to attain β_p -values of order R/a in 1-MA discharges, thus permitting study of limiting MHD instabilities and "bootstrap" diffusion effects. The radial profile of the q -value can be modified both by the current start-up technique and by control of the neutral-beam deposition profile.
- The temperatures calculated for high-purity hydrogenic plasmas are sufficient to raise most of the plasma profile into the predicted trapped-ion-mode regime, which is the most relevant regime for future tokamak reactors. The special stabilizing effects predicted in this transport regime for plasma zones where the density gradient is reversed relative to the temperature gradient can be studied by the cold-plasma-blanket technique.
- The impurity influx can be retarded, and theoretically even reversed, by the cold plasma blanket.
- Well-diagnosed experiments can be carried out on intensely reactive DD and DT plasmas; the effects of reactor-like alpha particle populations can be investigated.

REFERENCES

1. L.A. Artsimovich, Nucl. Fusion 12 (1972) 215.
2. H.P. Furth, Nucl. Fusion 15 (1975) 487.
3. J.M. Dawson, H.P. Furth, F.H. Tenney, Phys. Rev. Letters 26 (1971) 1156.
4. H.P. Furth, D.L. Jassby, Phys. Rev. Letters 32 (1974) 1176.
5. R.M. Kulsrud, D.L. Jassby, Princeton Plasma Physics Lab. Report MATT-1114 (1975), to be published in Nature; J.G. Cordey, W.G.F. Core, Nucl. Fusion 15 (1975) 710; D.L. Jassby, Nucl. Fusion 16 (1976), No. 1.
6. D.L. Jassby, Nucl. Fusion 15 (1975) 453.
7. R.W. Conn, D.L. Jassby, Princeton Plasma Physics Lab. Report MATT-1155 (1975), to be published in Proc. Int. Conf. on Radiation Test Facilities for the CTR Surface and Materials Program (Argonne, 1975), P.J. Persiani, ed.
8. Proc. DCTR Fusion-Fission Energy Systems Review Meeting (Germantown, MD 1974) U.S.E.R.D.A. Report ERDA-4 (1975), S.L. Bogart, ed.
9. R.P. Rose, et al, Westinghouse Electric Corp. Reports WFPS-TME-003, -024 (1975).

3.0 PLASMA EXPERIMENTS

The typical experimental operating modes of the TFTR device have been discussed in Section 2.1 and illustrated in Figure 2.1-2. Section 3.1 summarizes the plasma physics considerations that bear on the design of these operating modes. Section 3.2 presents computer modeling studies of plasma behavior in representative operating modes. Various aspects of fusion power production that are particularly related to the TFTR experiments are summarized in Section 3.3. The technical implementation of the standard operating modes will be discussed in Section 4.1; engineering "reference parameters" are summarized in Section 4.1.2.

3.1 PLASMA PHYSICS ASPECTS

3.1.1 DISCHARGE INITIATION

3.1.1.1 Initial Magnetic Field Structure

In the present-day tokamaks, plasma equilibrium is generally lacking during the initial phases of the discharge. Photographic evidence shows that the tokamak discharge usually begins with a "hot spot" near the edge of the limiter aperture. Large tokamaks will need to exert more careful control over the phenomena of discharge initiation. In particular, it appears advantageous to begin with a moderate-sized discharge (Figure 3.1-1) and gradually increase the minor radius and the particle content as the discharge current rises. The location of the initial breakdown within the limiter aperture can be influenced strongly by the method of preionization. (Appropriately placed major-circumference loops can be used for the induction of local ~ 100 kHz ohmic discharges, or one can have an input of microwave power that is resonant with the electron cyclotron frequency at a given major radius). In addition, it is desirable to provide an initial poloidal magnetic field structure that will avoid sweeping the discharge against a limiter, due to lack of MHD equilibrium, as the plasma current begins to grow. Extensive studies of the latter problem have already been carried out at PPL in connection with the planning of the PDX experiment.¹

The design of the TFTR OH-coil system has been governed by the desire to avoid irregular poloidal fields within the limiter aperture at the time of discharge initiation. Instead, it is planned to override all "stray" initial poloidal fields by means of a single hexapole null, centered at the desired initial discharge position. For this purpose, a separate "multiple winding" had been considered, but it appears at present that adjustment of the OH winding itself will be sufficient to provide the desired magnetic field structure. As in present-day tokamaks, adjustable initial radial and vertical magnetic fields will be available to "null out" small systematic errors in the magnetic field pattern.

Figure 3.1-1. Discharge initiation. Initial current flows at the breakdown null. The limiter aperture is programmed to increase as the current rises in order to shape the poloidal field profile within the plasma. New layers of plasma are added in the outer region by ionization of a controlled hydrogen gas feed. (PPL 753903)

The advantage of the initial hexapole null is that it will permit magnetic field lines to be closed within the "proprietary flux" of the discharge already at low current levels (unlike an initial nonzero vertical field or a quadrupole null), and will provide major-radius equilibrium from the first instants of the discharge up to the point in time where the EF system can provide accurate major-radius control. With a large limiter aperture, such as that of the TFTR, this approach should minimize the initial impurity deposit. On the other hand, the acceleration of runaway electrons to very high energies (up to the 100-MeV range) is favored by larger tokamaks and by properly closed initial magnetic fields. If this phenomenon were to prove troublesome, special countermeasures will have to be taken to "skim" the runaways before they have gained high energies. Once a large-sized, high-current tokamak discharge is established, the evolution of runaways should actually be a lesser problem than in present-day tokamaks, since for a given β_p -value the ratio of the directed velocity of the current carriers to their thermal velocity diminishes with rising line density

$$\left[v_d / v_{th} \propto (n a^2)^{-1/2} \right].$$

3.1.1.2 Initial Plasma Current Rise

The volt-seconds requirement of a large, hot tokamak discharge of the TFTR type is mostly inductive: If the plasma heating is at all successful, the circumferential voltage along the plasma axis soon drops below a volt, so that many seconds of discharge duration would be needed to equal the inductive volt-seconds investment, which is typically of order 10 Vs for a 2.5 MA plasma. There is, however, a serious problem connected with the magnitude of resistive volt-seconds losses during the initial phase

of discharge formation. If the power input during the low-current phase is insufficient to surmount ionization, radiation, and other losses, there may be an excessive initial "dwell time" at high plasma resistance (100 V applied to the discharge consumes 1 Vsec for every 10 msec of dwell), or the discharge current may even reach a low maximum and then decay again. This problem of initial discharge fizzle is more severe in larger tokamaks, because the central ohmic heating power goes as the square of the central current density, which is limited by the MHD safety consideration $q < 1$, and therefore scales as $J \rightarrow B_t/R$. Initial plasma impurity admixture also has a highly adverse effect (because of radiation cooling), but in this respect large tokamaks may have advantages, if they are designed for minimization of initial impurities. Finally, the timing of the hydrogen gas influx for plasma formation is critically important: low initial density favors rapid growth of T_e and improvement in discharge conductivity. (Excessively low initial density must, of course, be avoided from the point of view of runaway generation).

These remarks are illustrated by Figure 3.1-2: Case A represents an unsuccessful discharge-initiation attempt, where the plasma current begins to decline after the first 5 msec. Case B represents a successful initiation, with a resistive volt-seconds consumption of only 0.3 V/sec during the first 15 msec, and a rapidly rising current, which is mainly governed thereafter by the inductive circuit impedance. In both cases, the limiter aperture is 20 cm (the plasma being positioned near the outer vertical limiter, as in Figure 3.1-1). A transformer voltage of 150 V is applied for the first 5 msec, then reduced to 50 V. An initial (preionized) plasma density of 10^{11}cm^{-3} is assumed; the results are found to be insensitive to this value. In addition, the, hopefully, somewhat pessimistic assumption of 3% oxygen admixture is made throughout the discharge evolution. The difference in operating conditions between Cases A and B consists only in the program of neutral gas feed. In Case A the sum of neutrals and plasma electrons was ramped from $2 \cdot 10^{13} \text{cm}^{-3}$ to $3 \cdot 10^{13} \text{cm}^{-3}$ in 15 msec; in Case B the ramping was from $1 \cdot 10^{13} \text{cm}^{-3}$ to $3 \cdot 10^{13} \text{cm}^{-3}$. These illustrative cases serve to show that discharge-initiation problems can indeed be overcome by appropriate programming of the neutral gas feed. They also suggest that availability of a somewhat higher OH voltage than 150 V for the first 10 msec would be a prudent design feature. Engineering studies now underway are aimed at the implementation of breakdown voltages up to 300 V (twice the present reference value) in future TFTR design iterations.

In the discharge-initiation studies of Figure 3.1-2, a simple zero-dimensional computer code² was used. The same cases were also studied in greater detail by means of the full one-dimensional transport code of Section 3.2. In the latter case it is possible to follow phenomena with radial dependences, such as skin effect and localization of the various ionization states of oxygen.³ The overall results are very similar to those of the zero-dimensional code, and have served to document its validity.

Figure 3.1-2. Computer discharge initiation, with 150 V applied by OH primary up to $t = 5$ msec, and 50 V thereafter. Top: total density of neutrals and electrons is ramped from 2 to $3 \cdot 10^{13} \text{ cm}^{-3}$ in 15 msec; initiation fails. Bottom: Particle density is ramped from 1 to $3 \cdot 10^{13} \text{ cm}^{-3}$; initiation succeeds. Voltage shown is the resistive voltage drop along the magnetic axis. (PPL 753904)

3.1.1.3 Control of the Skin Effect

In present-day discharges there is evidence of enhanced decay of skin currents — which may be due to resistive MHD instabilities in the skin layer, or due to micro-instabilities associated with the shear-flow of the electrons that carry the skin current. In larger tokamak discharges, it will be desirable to control the buildup of the current distribution so as to avoid the onset of MHD modes. Even if the skin effect turns out to be damped spontaneously by microinstabilities, the accompanying heat outflow to the limiter may be excessive; it seems prudent, therefore, to provide means for avoiding the entire skin-relaxation phenomenon.

A natural method for avoiding skin effect is illustrated in Figure 3.1-1; this is a variant of the expanding-limiter concept originally proposed by B. Kadomtsev. An initial discharge similar in size and current to present-day tokamaks is formed (as in examples of Figure 3.1-2).

While the discharge current rises further, the equilibrium point is displaced inward in major radius and a new layer of plasma is formed between the initial plasma core and the outer vertical limiter. The radial distribution of the plasma current density and q -factor can be controlled through appropriate time dependences of the EF and the neutral gas feed. The evolution of this initial phase of the discharge has been calculated with the zero-dimensional code² (cf. Section 3.1.1.4). The subsequent neutral-beam heating phases are best followed by the one-dimensional code of Section 3.2. Further TCT computer studies are planned, in which all phases of the discharge will be followed by both the zero and one-dimensional codes.

3.1.1.4 Volt-Second Requirements

The TFTR mode requiring the largest volt-seconds capability is the high-current mode (D) of Figure 2.1-2. The prediction of our zero-dimensional code for this case are shown in Figure 3.1-3. The initial breakdown phase at constant limiter radius is terminated when the current has risen to the point where $q = 3$ (at the limiter); the limiter aperture is then increased rapidly, until the current reaches 1.1 MA in 140 msec, and more slowly, while the current rises to 2.2 MA. A small compression then shifts the discharge to its final position and current of 2.5 MA. (Alternatively, the compression could be carried out before the plasma current has reached the final value.) The resistive volt-seconds consumption in this process is 1.15 Vsec; the total consumption is 13.7 Vsec. This requirement on the OH transformer, as well as the requirement on the EF circuit, are well within the capabilities of the present systems. Volt-seconds limitations are encountered only in the form of a time limitation on the duration of the 2.5-MA discharge. In Figure 3.1-4, the maximum attainable duration τ of the 2.5-MA current plateau is plotted against the average plasma electron temperature. Longer times can be obtained for the high- β_p cases, since the contributions of the EF circuit to the total volt-seconds is then greater.

(A)

(B)

Figure 3.1-3. (A) Rise of discharge current and (space-averaged) plasma parameters during the formation phase of Mode D. The final peak density corresponding to a rounded radial profile would be $\sim 6 \cdot 10^{13} \text{ cm}^{-3}$. The voltage V is the resistive voltage drop. (B) Voltages and currents of the OH and EF circuits for the current build-up illustrated in (A). [PPL 763124 (top) and PPL 763123 (bottom)]

Figure 3.1-4. Volt-seconds-limited duration of the ~ 2.5 -MA flat-top regime for various average discharge temperatures and β_p -values. Average densities corresponding to β_p and $T_e = T_i$ are shown. Total OH and EF volt seconds are given by 10.6 Vsec , and $3.11 + 1.45 \beta_p \text{ Vsec}$. Current initiation consumes 0.8 Vsec resistively. An effective Z of 1.4 is assumed; for higher Z_{eff} -values, the T_e -scale is reduced as $Z_{\text{eff}}^{2/3}$. (PPL 753883)

The durations shown in Figure 3.1-4 satisfy the requirements of the reference TFTR operating modes as presently envisaged. Considerable thought has been given, however, to the usefulness of an iron transformer core as a means of further raising the volt-seconds capability of the TFTR. The principal effect can be obtained by filling the interior of the small- R section of the OH primary windings with a cylindrical iron core: the magnetization of this core creates an incremental $\sim 2 \text{ Vsec}$ of transformer flux "free of charge," for present OH circuit parameters. As a result, the discharge current could be made to rise more rapidly to the 2.5 MA level, and could be held there for longer times, accompanied by lower power dissipation in the OH primary windings. A secondary advantage, obtained by adding outer return-flux "legs" to complete the transformer yoke, would be the reduction of the poloidal fringe fields to which auxiliary experimental apparatus (such as the neutral beam lines) is exposed at the periphery of the TFTR device. The principal disadvantage, especially in the case of a transformer yoke with outer legs, arises from the local saturation of the transformer iron and from its interaction with the EF system: the relatively uncontrolled and nonsymmetric poloidal fringe fields that result within the limiter aperture tend to interfere with plans for optimization of the initial breakdown configuration (cf. Section 3.1.1.1) and to introduce additional uncertainties in regard to the confinement of particle orbits (cf. Section 3.1.3.3). In view of these conflicting considerations, an obvious strategy is to avoid the use of an iron core when operating with modest discharge currents — since the core is then not needed and merely serves to introduce complications — but to provide for its optional insertion when extensive studies of high-current discharges with long

durations are to be undertaken. Accordingly, the present TFTR reference design does not include any iron core, but makes provision for the optional insertion of a cylindrical central core at some later time. Magnetic shielding for auxiliary apparatus will be provided by specially dedicated local iron structures, rather than by means of massive transformer legs designed to channel the main OH flux.

3.1.2 MHD EQUILIBRIUM AND STABILITY

3.1.2.1 Poloidal Equilibrium Field

The predominantly vertical EF magnetic field of the TFTR is designed to give an equilibrium that is positively stable against axisymmetric displacements. The radius of curvature R_c of the EF lies in the range $0.4 > R/R_c > 0$, thus meeting the requirements for horizontal stability ($1.5 > R/R_c$) and vertical stability ($R/R_c > 0$). A curvature index of ~ 0.4 is useful in facilitating compression, but increases the sensitivity of the plasma major radius to errors in EF amplitude. A curvature index around zero reduces the maximum EF energy required for static high-current operation, but increases the sensitivity of the vertical plasma position to stray field components directed along R . In addition to the intrinsic stability of the TCT against axisymmetric plasma displacements, both active and passive feedback systems will be used to minimize transient excursions from the intended discharge locations.

3.1.2.2 Stability Against Kink Modes

The current channels of present-day tokamaks have a tendency to contract, raising the central temperature and current density and lowering the central q -value. This process continues typically until the safety factor q drops below unity; the fundamental helical kink mode ($n = 1, m = 1$) then begins to grow and precipitates a plasma disruption that reexpands the temperature and current density profiles. The central safety factor is clamped in the vicinity of unity by this spontaneous feedback control mechanism of the tokamak discharge. The price that is paid for this rather primitive form of control is the effective destruction of plasma confinement within the central zone of the discharge. If the limiter q -value is relatively low (i.e., $q \lesssim 3$), the affected interior zone tends to be fairly large; in addition, the higher-harmonic (resistive) kink modes ($n = 1, m = 2, 3, \dots$) become important and may precipitate the disruptive termination of the entire discharge. If the limiter q -value is relatively high ($3 \lesssim q \lesssim 8$), the MHD stability is perceptibly improved, as is the gross energy confinement time of the discharge.

Larger, hotter tokamak discharges are expected to have a number of advantages in respect to the control of kink modes. The time constant for the contraction of the current channel is calculated to become relatively long compared with E , since it depends primarily on the plasma conductivity, which is expected to remain close to classical values (except possibly in the special skin-current situation discussed in Section 3.1.1.3). In the one-dimensional computer studies of Section 3.2, central

q-values below unity are found to be avoidable during the entire time of interest, if the poloidal field distribution initially set up within the plasma has a central q-value slightly above unity. It seems likely that further control can be exerted by proper choice of the neutral beam deposition profile: in a beam-heated plasma, the temperature profile will not tend to shrink along with the current-density profile as in present-day ohmic-heated discharges. Thus the fundamental kink mode and its accompanying disruptions may be avoided altogether.

As to the higher-harmonic kink modes, there are theoretical predictions that these resistive MHD instabilities may become stabilized at higher plasma temperatures.⁴ On the other hand, one worries that, as τ_E increases, particle loss mechanisms due to low-level MHD activity may become important in limiting the plasma energy confinement. In order to minimize risks of this type, the TFTR is designed to achieve reactor-grade plasmas while operating with limiter q-values as high as 7.5 (cf. Figure 2.1-2, Cases A and C). On the other hand, the capability of operating down to $q = 3$ has also been incorporated (cf. Figure 2.1-2, Cases B and D), in order to take advantage of the possibility of strong radial compressions and high-current discharges in the event that MHD-stability should turn out not to be a sensitive problem.

3.1.2.3 High- β_p Limitations

A number of MHD equilibrium and stability considerations conspire to limit the confinable plasma pressure to $\beta_p \lesssim R/a$. The heating capabilities of the TFTR are sufficient to reach this limit for 1-MA discharges. (Roughly R/a megajoules of plasma energy content are needed to reach $\beta_p \sim R/a$. In the present context, the definitions of β_p is, of course, extended so as to include the unthermalized injected ions.) In the case of the 2.5-MA discharge, where a 6.25-times greater energy content would be needed to reach the same β_p -values, the computer models typically give $\beta_p \lesssim 1$ for available TFTR heating power. Some illustrative results of two-dimensional MHD calculations⁵ for the TFTR high-current and low-current operating modes are shown in Figure 3.1-5.

According to neoclassical theory, a milder type of high- β_p limitation will be encountered before the MHD limit is reached: For $\beta_p > (R/a)^{1/2}$, the poloidal magnetic flux will begin to be pushed outward by the plasma pressure, resulting ultimately in an undesirable "bootstrap" skin-current configuration. (Neoclassical processes are not dominant in most aspects of tokamak phenomenology, so one might question whether the bootstrap effect will actually be seen; a simple analysis of the anomalous transport case, however, suggests that the bootstrap effect should persist and may be even more powerful in this case.)⁶ The concern about such a development is mitigated by factors similar to those already discussed in connection with the contraction of the discharge current (Section 3.1.2.2): In large, hot tokamaks, the time required for the poloidal field to be driven out by the plasma pressure becomes much longer than τ_E . Furthermore,

Figure 3.1-5. Discharge equilibria computed by a 2-dimensional MHD code for a representative high-current case (top) and weak-compression case (bottom). (PPL 752069)

since we are here concerned simultaneously about the dangers of excessive contraction and excessive expansion of the current channel, it seems likely that there will turn out to be an intermediate β_p -value, adjustable through the heating power input, which will steer a safe course between $q(0) < 1$ and the bootstrap effect, even in steady state.

3.1.3 PLASMA ENERGY TRANSPORT

3.1.3.1 Anomalous Transport due to Microinstabilities

Plasma particles diffuse across the confining magnetic fields by Coulomb collisions. This "classical" process for transporting plasma mass and heat sets a lower bound on the plasma energy loss rate. The theory of plasma microinstabilities predicts "anomalous" transport rates that can be substantially greater than the classical ones. Classical loss rates decrease with rising plasma temperature and increase with rising density, whereas the most important anomalous loss rates (based on trapped-particle instabilities) decrease with rising density and increase with rising temperature. In present-day experiments, anomalous transport effects are clearly already at work, but the resultant gross plasma energy transport typically differs from the classical expectation by less than an order of magnitude. For tokamak reactor plasmas, the divergence between classical and anomalous transport predictions is far greater. (Plasma densities will be about the same as now, but plasma temperatures will be substantially higher.)

In planning the TFTR device, it has seemed prudent to assume anomalous, rather than classical, transport rates. The computer model of Section 3.2 is based on the mainstream concepts of present-day microinstability theory: drift waves, trapped-electron modes and trapped-ion modes. The microinstability regimes that will be encountered at the relatively cold outer edge of the TFTR discharge resemble those in present-day tokamaks; accordingly, predictions for the TFTR edge plasma are likely to prove fairly accurate. The higher-temperature, trapped-particle-mode regimes predicted to occur over the main part of the TFTR plasma profile are not yet documented by present-day experiments, so that these predictions could turn out to be either substantially optimistic or pessimistic. A comparison of characteristic regimes in present devices and in TFTR is given in Figure 3.1-6. The effect on the TFTR plasma of increasing or decreasing the theoretical trapped-particle-mode transport coefficients by a factor of 5 is examined in Section 3.2.3. While the impact on peak plasma temperatures and fusion yields is appreciable, the basic objectives of the TFTR project evidently can be met over the entire range of transport rates considered.

The TFTR experiment will not only play the role of providing a definitive test of the magnitude of anomalous transport in reactor-grade plasmas; it will also provide the opportunity to test some proposed means for suppressing this transport. As illustrated schematically in Figure 3.1-7, (and documented in detail in our computer transport studies), a controlled input of neutral hydrogen gas can be used to raise the edge density of a larger-sized tokamak plasma, without penetrating into the hot interior region or causing its radial extent to shrink appreciably. In the region of opposite temperature and density gradients, which lies between the axis and the density maximum, the anomalous transport associated with trapped-particle modes is predicted to be greatly reduced.⁷ The transport due to ordinary drift instabilities would be enhanced in the reversed-gradient region, but the overall effect is expected to be favorable.⁸ In the computer simulation of Section 3.2, this possible improvement in confinement has yet to be taken into account. If confirmed by further analysis and experiment, the gradient-inversion method may go a long way toward resolving the problem of anomalous transport in transient reactor plasma regimes.

3.1.3.2 Energy Transport by Neutrals and Impurities

The penetration depth of neutral hydrogenic atoms into present-day tokamak plasmas is comparable to the plasma radius. In TFTR-size plasma, neutral penetration will be limited to a small fraction of the minor radius. The edge neutral gas density in TFTR may be quite high — especially in the case of a pulsed gas input — but the neutral density at the plasma center is below 10^7 cm^{-3} . (In the case of neutral beam heating, there can be a significant contribution to the central neutral density due to that fraction of injected particles which is stopped by charge-exchange rather than by ionization; however, for injection energies $W_0 \gtrsim 120 \text{ keV}$, this contribution is not excessive.) As a result, ion energy loss by charge-exchange tends not to be important in the central region, and to become highly important only in cooling the plasma edge.

TRAPPED - PARTICLE - MODE REGIMES

Figure 3.1-6. Comparison of typical anomalous-transport regimes, as simulated by the transport code of Section 3.2, for various present-day and future tokamaks. TE1 and TE2 are for the collisional and collisionless trapped-electron-mode regimes. T1 is the trapped-ion-mode regime. Densities refer to central values. For ATC, a precompression regime is shown; for PLT, a 4-MW neutral-beam-heated mode; for TFTR, a strong-compression mode. (PPL 753093)

SMALL PRESENT TOKAMAKS

LARGE FUTURE TOKAMAKS

Figure 3.1-7. Schematic comparison of characteristic plasma-physical mechanisms in small present-day and large future tokamaks. In the latter case, gas influx can reverse ∇n without causing discharge shrinkage and onset of MHD instability. The reciprocal MHD safety factor $q^{-1} = l/2\pi$ is shown. (PPL 752068)

Nonhydrogenic atoms generally have an even shallower penetration depth into the plasma. Moderately heavy impurities, such as oxygen, are important in cooling the plasma edge by line radiation, but are no longer energetically important if they reach the hot plasma interior, since they become highly stripped so that their radiation power is much reduced. Heavy impurities, such as iron, remain partly stripped and continue to radiate effectively from the interior of all but the hottest TFTR plasmas. This radiation by heavy impurities from the hot plasma core constitutes a dangerous cooling mechanism, potentially comparable in importance to the plasma energy loss by anomalous transport. The impurity radiation from the edge region, on the other hand, is harmless — and in fact provides a convenient way of distributing the heat outflow from the central plasma region over a large wall area in a manner that minimizes sputtering of wall materials. Unfortunately, even the edge-plasma impurities give rise to undesirable side-effects (cf. Section 3.1.4); they also tend to diffuse into the hot plasma interior and dilute the reacting hydrogenic plasma. The minimization of every kind of impurity ion admixture is therefore a desirable objective in reactor-plasma experiments. As discussed in Section 3.2, in TFTR this objective can be achieved even under fairly pessimistic assumptions, at least in the weak compression case of Figure 2.1-2, which is specially designed to provide pure plasmas of reactor-like temperatures.

3.1.3.3 Large Orbital Excursions

In an ideal axisymmetric tokamak configuration, the charged-particle orbits have relatively small radial excursions. As Figure 3.1-8 illustrates, even the 3.5-MeV alpha particles of the DT reaction can be confined fairly well at the 1-MA level in a TFTR-type discharge with a current profile that is centrally peaked within the limiter aperture. The worst orbital excursion of the 120-keV deuterons is five times smaller, and the typical excursions of 120-keV deuterons injected tangential to the major circumference are about 10% of the limiter radius.

Symmetry-spoiling defects in the tokamak magnetic field structure, however, can lead to very large orbital excursions, which may intercept the limiter or the vacuum-vessel wall. A well known example arises from the modulation of the toroidal field strength along the major circumference by the discreteness of the TF coils. (This discreteness is unavoidable, since the widest possible access must be provided for the neutral-beam injection lines. Cf. Section 4.5) Particles that are mirror-trapped between TF coils can drift out vertically, due to the major-radius gradient of the toroidal field. The principal obstacle to this type of loss is Coulomb scattering, which can readily remove an escaping particle from its shallow magnetic trap between TF coils, and restore it to a well-confined orbit. Highly energetic particles, such as alphas, beam-injected deuterons, or runaway electrons, however, can escape before appreciable scattering; and even the bulk-plasma ions of reactor-grade plasmas can show significant enhancements of their transport rates. It is with these considerations in mind that the centers of the typical hot-plasma locations in Figure 2.1-2 were displaced toward small major

Figure 3.1-8. Fractional confinement of 3.5 MeV D-T alpha particles produced by a source $S(\alpha)$ in the TFTR 1-MA discharge. Various current density profiles J are considered, with middle case corresponding roughly to the typical situation. (PPL 744596)

radii, where the TF ripple is virtually negligible. On the other hand, the ripple in the region of initial plasma formation becomes as large as 5% at the limiter. Such values of the TF ripple are fairly typical of present-day tokamaks, but may prove excessive for TFTR plasma temperatures; if so, the precompression position of the plasma may have to be shifted slightly inward. (Alternatively, the TF ripple could actually turn out to be useful as a kind of nonmaterial limiter, to remove and distribute the hot-particle outflow from the plasma edge.) A method for reducing TF ripple by special correction windings was recently proposed, which would allow the field ripple to be reduced even when fewer TF coils are used, thus providing better access for neutral-beam injection.⁹

A less well-known orbit-spoiling effect arises from nonaxisymmetry of the poloidal magnetic field (e. g., due to eddy currents in the TF coils, or due to an iron transformer yoke with outer legs). While these poloidal field variations are not sufficiently strong to influence the particle trapping, they can easily displace the particle orbits in such a way that there results a net drift in the toroidal field gradient. A proper calculation of this phenomenon must begin by analyzing the (stable) plasma MHD perturbations that are forced on the plasma column by inhomogeneities in the poloidal field, and then proceed to calculate particle orbits within the perturbed magnetic-field structure.¹⁰ Rough initial estimates for the TFTR show that the associated particle losses should not be important, but further analyses are needed to clarify this topic.

Plasma particles can also escape along magnetic field lines, in case the confining magnetic field structure is imperfectly closed. In an ideal axisymmetric tokamak, this cannot happen, but in the presence of symmetry-spoiling effects of sufficient magnitude the magnetic field line structure can become slightly ergodic, so that a particle traveling along a magnetic field with high velocity may escape before scattering. In present-day experiments, this type of loss has been encountered mainly in the form of outward leakage of energetic runaway electrons. The principal causes of the loss have been low-level MHD instabilities and externally applied poloidal field asymmetries. This type of particle-orbit excursion can actually be helpful in "skimming out" runaway electron components. On the other hand, it could also lead to an undesirable enhancement of electron thermal conduction to the limiter.

3.1.4 IMPURITY PHENOMENA

3.1.4.1 Impurity Influx

Light impurities such as oxygen and carbon are generally present as loosely bound surface contaminants on the vacuum vessel wall. Their abundance can be reduced by pre-baking of vacuum vessel components, as well as by baking the vacuum vessel intermittently during the operation, typically at temperatures of 200-500°C. These light impurities are readily sputtered from the vacuum wall by the particle outflow of a tokamak discharge; they then enter the plasma and become ionized. This mechanism suggests a second highly effective means for removing surface contaminants: In "discharge cleaning," many low-level tokamak discharges are struck with a high repetition rate, and some fraction of the plasma-impurity mixture is turned to neutral gas and pumped out after each pulse. (The cleanup time can be shortened considerably by means of divertors or gettered surfaces that bury the entire plasma content after each discharge.) The discharge-cleaning approach has the attraction of cleaning preferentially the exposed surfaces, which are the principal sources of impurity evolution during regular tokamak operation; its disadvantage is that it fails to clean the recessed impurity reservoirs, which may then serve to recoat the exposed surfaces after the end of the discharge cleaning operation.

Some present-day tokamaks rely on vacuum-vessel baking only, while others rely on discharge cleaning only, and still others use some combination of the two. There has been no clear experimental demonstration of the advantage of any one of these methods over the others. The present TFTR design calls for discharge cleaning only, but engineering studies are being carried out for the additional provision of regular baking up to 200-250°C. (The TFTR vacuum-vessel components are, of course, prebaked at high temperature.) The provision of baking for TFTR would be of incidental usefulness in minimizing the expenditure of tritium while switching over from discharges in some other working gas.

The pumping speed for TFTR is $10 \text{ m}^3/\text{s}$, and the volume to be pumped is 64 m^3 ; so the pumping time constant is $\sim 6 \text{ sec}$. By means of discharge cleaning, and probably also baking, it is planned to reduce the total rate of gas evolution from the walls to $1.3 \times 10^{-5} \text{ Pa m}^3/\text{s}$ corresponding to a base pressure of $1.3 \times 10^{-6} \text{ Pa}$. (An appreciable fraction of this base pressure will actually be hydrogen.) Under these conditions, the equilibrium rate of gas evolution during a one-second discharge would create an impurity admixture of less than one part in 10^6 . Evidently the only significant source of impurity influx during a tokamak discharge is the greatly enhanced evolution of contaminants from those surfaces that are exposed to direct plasma particle bombardment. The base pressure of $1.3 \times 10^{-6} \text{ Pa}$ is important only as a measure of the cleanliness of the vacuum wall surface.

The impurities of present-day-tokamaks consist mainly of light ions (1-5% oxygen is typical), ingested into the plasma early in the discharge formation process. Heavy impurities seldom constitute more than 0.1% of the plasma ion population. There is no evidence that the plasma mixture becomes increasingly impure with time, even in the course of discharges lasting an order of magnitude longer than the plasma energy confinement time.¹¹ Large future tokamaks with reactor-grade plasmas may actually be less contaminated with light impurities, as a result of more careful initiation procedures — or at any rate because the volume-to-surface ratio will be more favorable. On the other hand, the charge-exchange neutrals produced by plasma ions of multi-keV energies will cause substantial sputtering of the metallic vacuum wall atoms, and the outflow of hot plasma ions to the limiter could sputter the limiter atoms. Thus one would predict a much more dangerous role for heavy impurities in future tokamak plasmas, such as those in TFTR.

In the computer studies of Section 3.2, the treatment of plasma impurities has been specialized to detailed consideration of sputtered iron atoms. The pessimistic assumption is made that the entire influx of iron remains permanently in the TFTR plasma. (This represents an upper limit on the time-averaged influx, which in practice is likely to be reduced somewhat — or even greatly — by an outflow of impurity ions along with the hydrogen plasma.) The motion of the iron ions within the plasma is governed in part by classical diffusion terms that tend to cause impurity concentration at the hydrogen density maximum, and in part by anomalous diffusion terms that tend to mix the impurities uniformly with the hydrogen plasma. The continuous accumulation of iron ions in the plasma sooner or later causes a catastrophic decline of the plasma temperature. It appears, however, that this event can be postponed beyond the time interval of principal experimental interest in all four representative modes of Figure 2.1-2. In the weak compression case, the impurity flux into the hot central core can be stopped altogether by means of a density maximum in the cold outer plasma (cf. Figure 3.1-7): the classical diffusion terms act in this case to concentrate the impurities in the outer plasma, rather than at the center. Development of this impurity-shielding technique is of considerable interest for initial reactor-plasma experiments and perhaps also for pulsed-ignition reactor modes

(cf. Section 3.3.2). In a steady-state tokamak reactor, of course, the density profile cannot be peaked outward, and so one will have to look to more power techniques of impurity control, such as the incorporation of tokamak divertors.

3.1.4.2 Effect of Impurities on Fusion Power

The role of impurities in cooling the tokamak plasma has been discussed already in Section 3.1.3.2. A second major effect is to dilute the population of ions that are useful for the production of fusion power. For a given plasma electron density n , the dilution of the triton population in TFTR will go as $n_I Z_I / n$, where n_I is the density of impurity ions and Z_I is their charge. The fractional reduction in energy multiplication factor Q in the TCT reactor mode is also roughly $n_I Z_I / n$. In spite of the Z_I -dependence of this effect, light impurities are as serious a hazard as heavy ones, since they tend to sputter more abundantly.

The dilution of the tritium plasma by impurity ions is taken into account in the studies of Section 3.2, and is partly responsible for the decline of fusion power at late times in the discharges.

3.1.4.3 Effect of Impurities on Neutral Beam Penetration

Even impurities that are localized at the plasma edge can have a serious influence on the central plasma temperature; they can stop a large fraction of the injected neutral beams by impact ionization.¹² While the cross sections for this process of beam deposition are not well-known experimentally, their predicted magnitude for the 100-150 keV range indicates a substantial — or even total — alteration in the TFTR beam deposition profile for the expected magnitudes of impurity contamination (cf Figure 3.1-9). In our computer studies this effect is found to be aggravated by a feedback mechanism whereby the impact-ionized beam particles heat the edge plasma preferentially and thus increase the impurity influx. Impurity control by pulsed-gas densification of the outer plasma is obviously of no help here, and would further reduce the beam penetration. As a result, the period of useful quasi-static neutral beam heating is found to be limited in our studies to several times τ_E . A more optimistic model would help to extend this time, but for purposes of allowing true steady-state injection into a reactor plasma the presence of a divertor is likely to prove necessary.

3.1.5. AUXILIARY HEATING

Since the attainment of reactor-like temperatures $T_e \sim T_i = 5\text{-}10\text{ keV}$ is an essential objective of the TFTR experiment, great emphasis has been placed in the Conceptual Design on the provision of a realistic plasma-heating capability. Primary dependence on ohmic heating does not appear to be a prudent course for plasmas in the density range of interest (cf. Section 3.2.2). MHD-limiting β_p -values —

BEAM-DEPOSITION INSTABILITY

Figure 3.1-9. Effect of plasma impurities on neutral beam deposition. Top: Schematic profiles of beam-deposited energetic ion density for various effective Z -values. Bottom: growth of an outer impurity barrier, due to the sequential processes of local plasma ion heating, charge exchange, sputtering of wall material, and increased impurity content in the edge plasma, which impact-ionizes the neutral beam. (PPL 753912)

or even β_p -values of order unity — could not be reached in this way. In view of the relatively low current, powerful auxiliary heating may be needed just to establish long-lived, low-resistance discharges with desirable current profiles.

At the present time, the auxiliary heating method that is best proven experimentally, and the theory of which is most readily extrapolatable to TFTR plasma parameters, is neutral-beam heating. This approach has the merit also of permitting very large fusion power densities to be reached with high probability (cf. Figure 2.1-5). Accordingly, the TFTR Conceptual Design includes extensive and detailed plans for high-powered neutral beam heating (cf. Section 4.5). The fraction of the total TFTR facility cost assigned to the heating capability is about 20%—greater than that of the device itself. The importance, and high cost, of an effective auxiliary heating method has also played a key role in the optimization of the TFTR parameters. In ohmic-heated tokamaks high currents have the twin advantages of providing better

confinement (if MHD-stable) and higher power input. On the other hand, in auxiliary-heated tokamaks the input power is fixed by cost constraints, so that the heating power density actually diminishes for higher-current discharges, because of the accompanying increase in plasma volume (cf. Figure 2.1-2). As a result, the optimization of tokamak parameters for the attainment of high plasma temperature and density in TFTR-sized tokamaks implies a somewhat stronger bias in favor of moderate plasma currents and high toroidal fields (i.e., small plasma volumes) than would be appropriate for present-day ohmic-heated devices.

While the present TFTR Conceptual Design is specialized to the use of neutral-beam heating, it is recognized that auxiliary heating by various high-frequency methods also has a great potential usefulness, both for the TFTR experiment and for future reactors.

In particular, high-frequency heating would avoid diluting the tokamak plasma ions with the ion species of the heating beam, and might also have advantageous plasma-penetration features. From the technological point of view, efficient high-frequency heating may well turn out to be easier to provide than efficient neutral-beam injection. High-frequency heating would also solve the problem of the outflux of plasma neutrons through the shielding penetrations that house beam-injection duct. Accordingly, the present TFTR device and building plans have kept in mind the preservation of the option of high-powered high-frequency heating. The prime candidates appear at present to be ion cyclotron heating (at ~ 50 MHz) and lower-hybrid heating (at ~ 2000 MHz). The nature and rate of implementation of these methods of TFTR will be guided by the rate of experimental success on PLT, PDX, and other large tokamak devices of the pre-TFTR generation.

3.1.5.1 Neutral Beam Heating

Tangential neutral beam injection (cf. Figure 3.1-10) is preferred over perpendicular injection for several reasons: the latter is more likely to excite microinstabilities (cf. Section 3.3.1); ions with large transverse energies born near the plasma edge often have banana orbits that intercept the limiter, especially in the presence of toroidal-field ripple (cf. Section 3.1.3.3); parallel-injected ions can be energized more effectively by compression; finally, for parallel injection the path length within the plasma core is greater relative to the path length through the cold edge region. Fast ions impart momentum to the bulk plasma in the process of thermalizing. The toroidal momentum is damped if B_t has about 1% ripple, but in a device such as TFTR, where the central plasma ripple is appreciably smaller, one must depend on oppositely-directed beams to minimize the net momentum. Neutral beams injected tangentially give rise to fast ions that have "passing" orbits located on surfaces centered around a stagnation axis (cf. Figure 3.1-10), somewhat displaced from the magnetic axis (about 5 cm in TFTR). The orbits of co-injected ions (injected parallel to the ohmic current) are shifted toward larger R -values, while the orbits of counter-injected ions (injected antiparallel) are shifted toward smaller R .

Figure 3.1-10. Neutral beam injection geometry. (a) Top view, showing the standard beam trajectory. (b) Side view, showing the limiter-intercepting outer boundary of co-injected ion orbits. Ions at the stagnation axis remain on the midplane. The actual outward shift of the stagnation axis relative to the magnetic axis is ~ 5 cm for typical TFTR parameters. Counter injected ion orbits are similar, but their shift is inward in R . (PPL 743445)

The selected neutral-beam energy must be suitable for optimal plasma penetration, which means minimization of the number of particles that are lost to the wall during plasma entry, or that reemerge from the far side of the plasma. For typical TFTR plasmas, the optimal penetration energy is in the range 100 to 150 keV, with the former corresponding to precompression states and the latter to high-current discharges. A second principal factor in the selection of a beam energy is the rapid drop-off in neutralization efficiency with rising energy: 100-keV deuterons are neutralized with $\sim 52\%$ efficiency, but 150-keV ions with only 31%. From the point of view of the fusion-power application of the injected beam, ~ 120 keV is the optimal instantaneous energy for maximizing the DT reaction rate relative to the collisional slowing-down rate (cf. Section 3.3.1), but injection at 150-keV gives a somewhat better time-averaged Q -value (Figure 2.1-3). Taking these various factors into account, and weighting them with the slightly greater likelihood of having somewhat lower-energy injection systems available in time to meet the TFTR project schedule, has led to the specification of 120-keV injection for initial operation. The optional provision of an electrical system capable of 150-keV injection is under study. In the present state of the ion-source art, the 120-keV neutral beam power

will actually be "contaminated" with an added 60-75% of fractional-energy neutral power, resulting from molecular ions which are neutralized more efficiently than the primary beam. (The 150-keV would contain an added 80-100% of fractional-energy power.) Advances in design and wall-material selection for the ion source may serve to give better control of the beam composition.

The computer studies of Section 3.2, which take into account realistic beam deposition profiles, half-energy components, etc., have led to the conclusion that about 20 MW of 120-keV beam should be provided; this would be accompanied by 12-15 MW of fractional-energy beam. The lower-energy component is actually useful during preheating, but aggravates the problem of impurity influx in the quasi-static heating phase (cf. Section 3.4.3). A maximum injection period of 500 msec has been specified, since this seems ample to cover the intervals of principal experimental interest in our computer studies. Extension of injection to longer times may be indicated in the event that impurity problems prove to be less severe than we have assumed.

In standard reactor-studies operation, an energetic deuteron beam will be injected. Use of triton beam would be attractive for preheating, but would complicate the tritium-handling problem, and is not currently planned. If hydrogen beams are used in initial operation, to avoid neutron production, their neutralization efficiency at 120 keV will be only 13%. Helium beams would be advantageous, but would complicate the cyro-pumping in the neutral beam lines. The special technical problems of the beam lines, pumps, and dumps are reviewed in Section 4.5. It appears that four separate beam lines (two each, for co- and counter-injection) should be sufficient; but access is being provided for the option of two additional lines. Specification of the permissible gas influx from the neutral beam lines into the main vacuum chamber is based on a "discharge physics" consideration. The cold-plasma-blanket effects that are to be created in Mode A of Figure 2.1-2, and avoided in Mode C, depend on the addition of a pulsed gas input comparable to the plasma particle content. If these effects are to be studied controllably, the spontaneous gas influx from the beam lines should be held to about 10% of the plasma particle content.

3.1.5.2 Ion Cyclotron Heating

Ion-cyclotron resonance heating (ICRH) with a "magnetic beach" has been used on various stellarators to produce bulk ion temperatures as high as 600 eV without apparent enhancement of plasma losses. In the tokamak configuration, the geometry for wave launching and absorption is considerably different; one makes use of the natural gradient of the B_t -field. Experiments on the ST tokamak at PPL have produced ~ 100 eV ion temperature rises, but have also indicated a limitation on the permissible energy input due to wall-impurity evolution. The problem appears to have been caused largely by wall-intercepting energetic ions, including resonant impurity ions, and is expected to diminish in higher-current tokamaks. One key consideration here is the availability, in larger tokamaks, of a spectrum of resonant plasma modes that is much

more suitable for volume heating than the spectrum in small tokamaks, which favors surface heating. In addition, the orbits of resonantly heated energetic ions will be much better confined in higher-current discharges. A modest advance in the suitability of experimental conditions will be realized in the ATC device, where ICRH experiments are planned. A decisive demonstration of ICRH, under conditions that are closely relevant to TFTR, will await the implementation of a multi-megawatt 50-MHz experiment on PLT.

The most promising ICRH approach for TFTR appears to be the excitation of fast waves at twice the plasma ion cyclotron frequency. The exact frequency to be specified for the heating system will depend on whether deuterons or tritons are to be resonant, and on whether the heating is to be carried out before or after compression; the frequencies of principal interest are in the neighborhood of 50 MHz. The heating power input would be comparable to the neutral beam power input specified in the present TFTR Conceptual Design. The degree to which the same power supplies should be shared by ICRH is still under consideration. The detailed design of the RF coupling structures will depend on experimental results in PLT, but it appears that the required port sizes are smaller than those for neutral beam injection. (Both power input systems could be accommodated simultaneously on TFTR, if desired.)

Aside from the use of ICRH to help preheat the plasma for beam-injected, two-component reactor experiments, one can also envisage the study of regimes where ICRH is used to raise the plasma ion temperature preferentially, and the principal DT yield results from bulk-ion reactions.¹³ As has been pointed out in Ref. 14, energy multiplication factors of the same order as those in the two-component regime, or greater, can be achieved even if T_e lags considerably below T_i (e.g., $T_e \sim 7\text{keV}$, $T_i \sim 20\text{keV}$). The required electron $n\tau_E$ -values are somewhat greater than in the two-component regime, but still well within the range of plausible expectation. This form of fusion-power generation would not give quite as high power densities as the two-component approach. In relation to the simulation of the ultimate tokamak reactor regimes, the ICRH approach would have the merit of emphasizing bulk-ion reactions. On the other hand, the heating of the bulk ions by way of beam-heated electrons ($T_e \gtrsim T_i$) in the two-component regime is more nearly representative of the ultimate alpha-particle heating situation at thermonuclear ignition.

3.1.5.3 Lower Hybrid Heating

A lower hybrid heating (LHH) system on the TFTR could be used to heat either ions or electrons, depending on the nature of the waveguide coupling system. A combination of theory and experimental tests on the (linear) H-1 device and the ATC tokamak at PPL indicates that the wave coupling and penetration problems have been solved. If efficient heating of the central plasma core can also be demonstrated in experiments currently underway on ATC, there will be good reason to expect LHH to work satisfactorily on TFTR. Further experimental tests, in more directly relevant plasma regimes, are planned for the PLT and PDX devices.

The appropriate input frequency range for TFTR heating will be around the ion plasma frequency (~ 2000 MHz). The RF coupling system as currently conceived would consist of two rows of waveguides feeding into the outside equatorial ports. Each row would consist of three pairs of waveguides with adjustable phase differences between each member of a pair, to control the wave penetration. In regard to the LHH power level, sharing of available power supplies, and suitability of present TFTR ports for power input to the plasma, the situation is roughly the same as for ICRH.

The LHH power could be used to help preheat the TFTR plasma, and may have the advantage of giving greater preference to the heating of electrons than is the case for neutral-beam heating. Alternatively, LHH could be used to enhance the energetic-ion tail and so establish a two-component plasma on its own; the disadvantage here would be less accurate control over the ion-energy distribution and the energetic-ion species. Finally, LHH could be used as an alternative to ICRH in heating the bulk plasma ions.

3.1.6 ADIABATIC COMPRESSION

The tokamak equilibrium (cf. Figure 2.1-1) can be shifted to smaller major radius R by increasing the strength of the externally applied vertical magnetic field component (EF). In this process, the toroidal field at the plasma location increases as $\beta_t \propto R^{-1}$, and the plasma minor radius must shrink as $a \propto R^{1/2}$, in order to conserve the toroidal magnetic flux within the plasma cross section. The plasma volume current increases as $I \propto R^{-1}$, while the safety factor q remains the same. If the "compression ratio" is defined as the ratio of initial to final major radius $R_0/R_1 = C$, then the plasma density increase is given by $n_1/n_0 = C^2$, and the temperature increase for adiabatic compression is $T_1/T_0 = C^{4/3}$. Here it is assumed that the plasma particle and energy confinement times are both considerably longer than the compression time τ_C , and that the self-collision times of the plasma particles are sufficiently short to keep the plasma isotropic and Maxwellian. For the injected ions, the latter assumption clearly does not apply. For the planned tangential injection, the velocity distribution of the energetic ions will be strongly peaked in the directions parallel and antiparallel to the magnetic field, and their energy will be raised more effectively ($W_1/W_0 \propto C^2$) than that of the bulk-plasma particles.

Compression experiments on the ATC device at PPL have verified all of these predictions — with the exception that the ATC electron temperature is generally found to increase less strongly than $C^{4/3}$, corresponding to a non-negligible electron-energy loss during the compression time of ~ 2 msec. The maximum compression factor in ATC was $C = 2.3$, so that the density and temperature increases observed, and the final plasma parameters achieved were substantial. Some recent data, obtained by combining ohmic and neutral-beam preheating in the precompression phase is shown in Figure 3.1-11A.

(A)

(B)

Figure 3.1-11. Recent ATC experimental results relevant to TCT operations.

(A) A 70-kA discharge is preheated by various combinations of ohmic heating and neutral-beam injection, and then compressed by $C \sim 2.3$ at $t = 30$ msec. The peak density reached, with ohmic heating only, is 10^{14} cm⁻³; with 230 kW beam injection, it is $2 \cdot 10^{14}$ cm⁻³. (B) A 10-kA discharge is initiated against an outer vertical limiter and then removed from limiter contact by a slight inward compression ($C \sim 1.1$). The plasma core profiles remain well detached, as shown, for the 10-20 msec discharge duration, while a cold low-density "halo" forms between the plasma core and the limiter. The plasma current rises gradually to a maximum of 30 kA. (A: PPL 753841; B: PPL 753845)

While adiabatic compression is at present the best-documented and most powerful way of energizing a tokamak plasma, it has the inherent disadvantage of not depositing heat irreversibly; adiabatic compression is merely a way of transforming the discharge parameters. For example, one can preheat the plasma slowly at low power density, by some irreversible method, and then transform it rapidly to high temperatures and densities by compression. The price one pays for this capability is that the final plasma volume is considerably smaller than the total magnetic-field volume that must be provided (cf. Figure 2.1-2). The degree of usefulness of the compression depends on the nature of the dominant energy loss process and on the particulars of the plasma preheating method. In ohmic-heated, radiation-cooled discharges, or when only a limited neutral-beam heating power is available, a strong compression capability can have great advantages. In the case of large tokamaks with (presumably) adequate neutral-beam heating, such as the TFTR, the provision for a compression factor comparable to that in ATC would cause a disproportionately large increase in the cost of the device and associated power supplies. It was therefore decided not to specialize the TFTR as strongly in the direction of adiabatic compression as was done in the case of the ATC device, but simply to arrange for the maximum available compression capability within a basic limiter aperture and magnetic field structure appropriate for a normal tokamak device. In this way, the dimensions shown in Figure 2.1-2 were selected: the maximum compression capability for 1-MA discharges is $C = 1.5$; higher currents are less compressible, and compression of a 2.5-MA discharge has room only for $C = 1.1$.

As can be seen in Sections 3.1.1, 3.1.4, 3.1.7, and others, the radial compression capability of the TFTR fulfills a variety of needs. It is used to initiate and terminate the discharge. It serves to position the hot plasma core away from limiter contact. (The latter application of compression has also been studied on ATC and is illustrated in Figure 3.1-11B.) The main compression ($C=1.2$ if "weak" or 1.5 if "strong") serves to increase the plasma density by factors of 1.44 to 2.25, the bulk plasma temperature by 1.27 to 1.72, and the super-thermal-ion energy by 1.44 to 2.25 for parallel-velocity ions, or 1.27 to 1.72 for an isotropic ion population. The peak energy of the 120-keV injected ions can thus be raised to 173 keV for weak compression and to 270 keV for strong compression. In the strong compression case the amplification of the plasma-heating effectiveness of beam power injected prior to compression is substantial; on the other hand, higher-current cases can have longer useful beam-injection interval. As a result the total plasma energy obtained in the computer studies of Section 3.2 is quite comparable for the strong-compression case and for the 2.5-MA quasi-static injection case, as is the total fusion yield. The plasma β -value is somewhat higher in the strong compression case, and the p -value and the fusion power density are substantially higher.

The shortest time during which the TFTR will be able to carry out its compression process will be 20-30 msec, limited primarily by the capabilities of the EF power system and by eddy-currents in the TF coils and their support structure. Since the

energy confinement time of both electrons and ions is generally well in excess of 100 msec, the compression-heating process will be fairly adiabatic. Similarly, since the slowing-down time of the energetic ions is of order 100 msec, their energy content can be raised well above the precompression value. Alternatively, the compression can be protected on purpose to maintain the injected deuterons at the ~ 120 keV level while they react with the tritium bulk plasma. In this way the system Q-value can be maximized (cf. Section 3.3.1).

3.1.7 TERMINATION OF THE DISCHARGE

Present-day tokamak discharges are permitted to terminate in an uncontrolled manner: sometimes by gradual decline of the current, sometimes by a disruptive instability, and commonly by a combination of the two. In TFTR-sized plasmas, typically carrying several megajoules of plasma heat energy, plus several megajoules in the form of a poloidal magnetic field and superthermal ions, provisions for the harmless deposition of the energy content in the form of limiter and wall heating begins to be a major technical consideration. The vacuum wall must evidently be built to withstand even a violent instability of the tokamak discharge, which might entail a local deposition of the plasma heat in ~ 100 μ sec times. On the other hand, the standard operating modes should be designed to spread the heat load controllably in time and space.

The termination scheme envisaged for TFTR is precisely the reverse of the initiation scheme shown in Figure 3.1-1. First the plasma is moved outward in major radius, thus reducing its thermal and poloidal-field energy. (The energetic ions are assumed to have thermalized before the termination procedure is started.) Next the OH voltage is reversed, driving down the plasma current and removing the poloidal flux embedded in the plasma "layer by layer," while the discharge edge is pushed into the outer vertical limiter. The limiter heat load in this phase can be reduced by pulsing a heavy gas (e.g., argon) into the plasma edge region, which will help to radiate the plasma heat outflow over a much wider wall area. This technique should be useful in TFTR-size plasmas, since the edge can be cooling without causing the central current channel to shrink. As the plasma size is reduced, however, the impurity-radiation-cooling approach is in danger of bringing on premature disruptive instabilities. (Another possibility would be to utilize a strong TF ripple at the large-R edge of the vacuum vessel to serve as a kind of magnetic limiter, causing the most energetic plasma particles to be lost by vertical drift to appropriately placed collector plates.) After the current has been driven down in this way to the level of a few hundred kA, the discharge can be allowed to terminate in the manner of present-day tokamaks.

3.2 COMPUTER STUDIES OF THE OPERATING MODES

During the past five years, a one-dimensional, time-dependent code has been developed to model the radial transport of plasma particles, plasma heat, neutral atoms, and impurity ions in tokamak discharges. The code is also able to handle the injection, deposition, and slowing-down of neutral-beam particles, the recycling or controlled injection of cold neutral gas, and the adiabatic compression of the plasma. It takes into account atomic processes, such as ionization, recombination, and radiation, as well as nuclear processes, such as the DT reaction. Preliminary versions of the transport code have been reported from time to time;¹⁵ a more definitive write-up is currently in progress.¹⁶

In many respects, the transport code can be based on well-documented and accurate physics input data — for example, in the case of atomic phenomena. Unfortunately, the least well known part of the physics input has to do with the plasma heat and particle transport coefficients themselves. In this respect the code output cannot be considered as providing reliable predictions; the situation is rather that the comparison of code predictions with actual experimental results serves as our most powerful means of testing the theory of transport in hot plasmas. Initial versions of the code were based on neoclassical and pseudoclassical transport coefficients and provided only very rough — and generally very optimistic — predictions of the observed temperatures and confinement times. The present refined version of the code has succeeded in predicting detailed profiles and other particulars of ongoing experiments with fair accuracy. It has shown a mild average bias on the side of optimism — whether for basic reasons, or because of particular experimental defects, is not clear as yet. The code is in a state of continual testing and development. At present ATC and TFR data are being digested, and PLT data are being predicted. Following initiation of PLT experiments, a comparison with the PLT predictions will permit further code refinements, which should then provide substantially reliable guidance for our preparations for TFTR experimental operation.

3.2.1 CONTENTS OF THE TRANSPORT CODE

3.2.1.1 Six-Regime Transport Model

The code employs local transport coefficients that are derived from the present theory of anomalous microinstability transport (cf. Section 3.1.3.1). Six different regimes are treated (cf. Figure 3.2-1); in order of decreasing plasma collisionality they are:

- (a) Collisional drift waves. (Three regimes: Bohm, Kadomtsev, pseudoclassical.)
- (b) Trapped-electron modes. (Two regimes.)
- (c) Trapped-ion modes. (One regime.)

The code, in effect, selects the regime that gives the greatest transport at any given time and radial position, taking into account the local plasma parameters, including the gradients of temperature, density and rotational transform (i.e., shear). To facilitate the regime-selection process, the selection is based on simple algebraic profiles that are fitted to the actual computed profiles at each time step.

The code also computes the classical transport coefficients and substitutes them for the anomalous ones when the classical transport is greater. (This happens only for the ion heat flow.) Finally, the code notes the appearance of MHD-unstable regions, where the safety factor q drops below unity, and specifies the relatively large Bohm value for transport over any such region.

Figure 3.2-1. Schematic of 6-Regime Anomalous Transport Model

Bohm: $D_B = cT_e/16eB$;

Kadomtsev: $D_K = 2\nu_c^{1/3}(1 + T_i/T_e)^{1/3}(r_n\theta)^{-2/3} \times (\Omega_i\Omega_e)^{1/3}(cT_e/eB)^{4/3}$;

Pseudoclassical: $D_p = 2\nu_e\rho_e^2\theta^{-2}(1 + T_i/T_e)$;

Trapped Electron Modes: $D = 3(r_n r_T)^{-1}(r/R)^{3/2}(cT/eB)^2\nu_e/(\nu_e^2 + \nu_o^2)$,
with a cutoff if $\nu_e R/r > 3\omega_{he}$;

Trapped-Ion Modes: $D = 2(\nu_e r_n^2)^{-1} \times (r/R)^{5/2}(cT/eB)^2(1 + T_e/T_i)^{-2}$
with a cutoff if $\nu_e \nu_i^{1/2}(R/r)^{3/2} > \omega_{bi}^{3/2}$.

Thermal transport in the drift-wave regime is set equal to D ; in the trapped-particle regimes, it is of order 3-5 times D .

Notation:

$$r_n^{-1} = d(\ln n)/dr; \quad r_T^{-1} = d(\ln T_e)/dr; \quad \theta = rr_n B^{-1} \times |d(B_p/r)/dr|;$$

$$\nu_o = (5r/Rf_n)(T_e/m_i)^{1/2}\theta^{1/2}; \quad \nu_e, \nu_i = \text{electron and ion collision frequencies};$$

$$\omega_{be,i} = (r/R)^{1/2}(2T_{e,i}/m_{e,i})^{1/2}/qR, \text{ trapped electron and ion bounce frequencies};$$

$$\Omega_e, \Omega_i = \text{electron and ion gyrofrequencies};$$

$$\rho_e = \text{electron gyroradius}.$$

(PPL 752114)

3.2.1.2 Neutral Atom Background

The model includes an influx of "cold" neutral gas at the plasma boundary, as well as ten generations of "hot" neutrals arising from the cold gas by successive charge exchanges. The energy prescribed for the cold gas does not seem to be an important variable: the present computations use 10 eV. The amount of cold-gas influx varies with time during the calculation, and is usually determined by the requirement that it should balance exactly the outflux of ions and charge-exchange neutrals (100% recycling). In the experiments there is always a small steady evolution of gas from the wall — or a larger one from neutral-beam injection lines — independent of the plasma recycling process; furthermore, programmed pulsing of additional gas may be specified on purpose, to help control the edge-plasma conditions. Accordingly, our computations can include a prescribed additional component of cold gas influx.

3.2.1.3 Impurities

The code contains a detailed treatment of heavy impurities, represented by the iron model. A small initial iron admixture can be specified; beyond that, the code calculates the appropriate rate of iron influx into the plasma by taking note of the spectrum of outgoing charge-exchange neutrals and inferring the sputtering rate. We have assumed a sputtering coefficient

$$S = S_m^4 \left[(E - E_c) E_m \right]^{3/2} / \left[1 + 3 (E/E_m)^2 \right]$$

where:

$$E_c = 23 \text{ eV}, E_m = 6 \text{ keV}, S_m = 2.1 \cdot 10^{-2},$$

for tritium on stainless steel. All injected iron atoms are retained in the plasma — which is probably pessimistic. On the other hand, limiter sputtering by the edge outflow of plasma ions is neglected. The iron ions are transported inward at the appropriate classical rate (which depends on the collisionality of the regime); superimposed is an anomalous diffusion (spreading) which is the same as for the normal plasma particles at the same location. The iron ions are stripped through successive stages of ionization, and radiate according to the local coronal model (cf. Figure 3.2-2). (The reduction in the ratio of tritium ion density to electron density, due to the electrons stripped from the iron ions, is taken into account in calculating the DT reaction rate.)

The code simultaneously treats light impurities according to a simplified model. An initial deposit of oxygen is assumed; beyond that the edge radiation and ionization cooling due to recycling of both hydrogen and oxygen is simulated by assuming an energy loss of 600 eV per hydrogen ionization event. This number has been calibrated in separate calculations using an oxygen rate-equation code. Both light and heavy impurities contribute to the effective ionic charge Z_{eff} , which is used in computing all transport coefficients.

Figure 3.2-2. The mean ionic Z and the total radiation for Fe ions in coronal equilibrium (generalized to include dielectronic recombination and auto-ionization), as a function of T_e . The detailed computations have been done specifically for $n_e = 10^{14} \text{ cm}^{-3}$ and $n_{Fe} = 10^{12} \text{ cm}^{-3}$, but the radiation would be expected to be linear in both densities, within a reasonable range around these values. The root-mean-square value of the ionic Z , which is also needed for the transport, is barely distinguishable from the mean Z . These results have recently been obtained for us at LASL by the authors of Ref. 19. (PPL 753208)

3.2.1.4 Neutral Beam Injection

Provision is made for simultaneous injection of a full-energy and a half-energy beam. The profiles of their radial deposition (on the basis of electronic ionization, charge-exchange, and impact ionization by regular plasma ions or impurities) are calculated at each time step, taking into account the specified injection angle and beam size. The injection powers at full and half energy, and the period of injection are also specified. In the calculations, the injection period is divided into ten equal intervals of time, and each of the two beam populations is divided into ten components according to the time of injection. The beam density is built up in each time interval with the appropriate radial distribution and according to the injection rate; however, all ions injected at the same radius within a certain time interval are subsequently assumed to have the same energy, which is determined by the classical (Fokker-Planck) slowing-down process. Energy diffusion of the beam ions is thus neglected, although it would be expected to increase the nuclear energy yield by about 10%. The energy lost in slowing down is, of course, transferred to the background ions and electrons according to the classical formulae. In this model, the "banana orbit" width of the beam ions is neglected, as is their radial diffusion; thus, pitch-angle scattering of the beam ions is irrelevant. The beam ion density is attenuated by charge exchanges with the background neutral gas, and the high-energy neutrals so produced are assumed to be lost entirely. Nuclear reactions between the beam deuterons and the plasma tritons are included, with the resultant alpha-particle energy assumed to be deposited instantly into the electrons. When the energy of a beam ion has been reduced to the background-ion temperature, the beam ion is entirely neglected; thus, the deuteron density in the background tritium plasma does not build up, and there are no nuclear reactions among the background plasma particles. The finite triton temperature is also neglected in calculating the beam-plasma reaction rate. Alternatively, one can treat the case where the background plasma is a deuterium-tritium mixture of fixed proportions; in this case nuclear reactions among the background plasma particles are included but, again, the background deuteron density does not build up.

3.2.1.5 Adiabatic Compression

The code incorporates the facility for making an instantaneous major-radius compression by an arbitrary factor C . The plasma density, temperature, and poloidal field profiles are compressed according to the adiabatic laws (cf. Section 3.1.6), and the discharge subsequently evolves according to the transport processes in the new configuration; the neutral gas is, of course, not compressed, but readjusts to the new conditions. The beam ions are compressed along with the background plasma: their energy is increased by the factor C^2 . More accurate computations of the energetic-ion velocity-space behavior and the compression process have been carried out in time-dependent cases having zero space dimensions but two dimensions in velocity space.^{17,18} The present transport code is in the process of being checked and upgraded in regard to the treatment of velocity-space-related effects.

3.2.1.6 Special Conditions for the Present Runs

The computations in the following sections make use of the simplifying assumption that there is no skin effect: the current is at full strength from the beginning of the calculation; the initial current density is parabolic as a function of radius. This artificial model is used precisely because an initially time-rising current would give such a strong skin effect as to interfere with subsequent calculations — unless one assumes either a set of anomalous transport factors to suppress the skin effect spontaneously, or else a time-dependent limiter aperture to suppress it by external control. The latter situation is, in fact, planned for TFTR (cf. Section 3.1.1), but detailed one-dimensional modeling studies have yet to be carried out for the initial discharge formation process.

The initial temperature profiles are assumed to be roughly parabolic, with a central value of 200 eV for both T_i and T_e , and a boundary value that is held at 10 eV throughout the calculation. The density profile is similar, with an initially specified central value. The present calculations are all for tritium plasmas and deuteron beam injection. We assume an initial oxygen contamination giving $Z_{\text{eff}} = 2$ (i.e., $\sim 1.5\%$), and an initial iron contamination of 0.1% of the central density.

3.2.1.7 Computer Diagnostics

The code prints out, at every specified time step, radial profiles of T_e , T_i , n , B_p , J , q^{-1} , Z_{eff} , the discharge electric field E , the impurity density n_{Fe} , the density and mean energy of each of the twenty injected-ion groups, the density of cold neutrals and the density and temperature of each of the ten charge-exchange neutral generations, as well as the power flowing through each radial position in the form of ion and electron particle flow, heat conduction, charge-exchange, and radiation. The code also indicates the prevailing transport regime at each radius. It keeps track of the total particle number and the energy in electrons, ions, beam particles, and magnetic fields, and computes the β_p -value and the particle and energy confinement times, the DT-yield, and numerous incidental parameters. For the sake of brevity, only a few of the diagnostics are displayed and discussed in the following sections.

3.2.2 OHMIC HEATING

In order to appreciate the role of neutral-beam heating in TFTR-size tokamaks, it is useful to begin with a study of purely ohmic-heated discharges. Results for the 2.5-MA mode of Figure 2.1-2 are illustrated in Figure 3.2-3. The characteristic result is that the ohmic-heated discharge either develops the normal centrally peaked T_e -profile, with a central temperature in the ~ 4 -keV range, or the central part of the T_e -profile collapses to very low temperatures, leaving a somewhat

Figure 3.2.3. Ohmic-heated discharges with $I = 2.5 \text{ MA}$ and various values of central density. Profiles at bottom are for a single discharge at $t = 650$ and 1000 msec. The radial scale is normalized to a limiter radius of 85 cm . (PPL 753936)

hotter edge region, in the ~ 1 -keV range. (These two types of discharge have a striking resemblance to "ORMAK Type A" and to the more pathological forms of "ORMAK Type B.")²⁰ The determining factor in bringing on the undesirable second type of discharge is an increase in the magnitude of the impurity radiation power relative to the ohmic heating power. In the illustrative cases of Figure 3.2-3, the discharge with central density $5 \cdot 10^{13} \text{ cm}^{-3}$ is well behaved at $t = 650$ msec, while the discharge with $7.5 \cdot 10^{13} \text{ cm}^{-3}$ central density begins to show a radiation-cooling "notch" in a region half-way out to the limiter, where the combination of plasma and impurity densities is most conducive to radiation cooling. The same phenomenon is already at a more advanced stage in the discharge with 10^{14} cm^{-3} central density: at $t = 1000$ msec, the T_e -notch has swallowed up the entire central part of the profile. The basic reason for the suddenness of the collapse is clear from Figure 3.2-2: when T_e is in the low-keV range, the radiation power increases with falling T_e , so that a kind of thermal instability can take place. Comparing the collapsing discharges with present-day experiments, one notes that the problem is due to the weakness of the ohmic-heating power, rather than to the need to sustain unusually high energy densities.

The plasma thermal energy density in the ATC experiment of Figure 3.1-11 is actually greater than that for any of the ohmic-heated cases of the present section.

Some characteristic parameters for these discharges are given in Table 3.2.2-1. In the normal case ($5 \cdot 10^{13} \text{ cm}^{-3}$ density), only some 30% of the input power goes into radiation. In the pathological case (10^{14} cm^{-3} density), all of the input power and more — goes into radiation, since the plasma is actually getting colder at late times.

Ohmic heating alone could clearly produce some very interesting high-temperature plasmas in TFTR, provided that one could be sure of long-pulse operation, with a discharge profile of the normal type. In our computer studies, this can be done by operating at sufficiently low plasma density — or by making more optimistic assumptions in the treatment of the impurity influx. One must worry, however, that in the actual TCT experiments unfavorable new factors may arise (e. g., plasma loss by edge MHD instabilities), which may shift the power balance in favor of the pathological discharge profile. In that case, an effective countermeasure will be to reinforce the 2-5 MW of ohmic heating power with the available 35 MW of neutral beam power (cf. Table 3.2.2-1). Even though the 10^{14} cm^{-3} central-density case is relatively unfavorable to neutral-beam penetration (the edge impurity density rising, in our calculations, to $n_{Fe} \sim 10^{12} \text{ cm}^{-3}$), a normal temperature profile with ~ 4 keV central maximum is maintained at a time when the purely ohmic-heated temperature profile would already have collapsed. Better results are obtained in lower-current or lower-density regimes that are specifically optimized for neutral-beam heating; it appears, however, that the fractional impurity content in even the cleanest of the neutral-beam heated regimes is somewhat higher than that in the cleanest ohmic-heated case. This is the price one pays — at least within the present treatment of impurity influx — for raising the ion temperature.

TABLE 3.2.2-1 REPRESENTATIVE PARAMETERS OF 2.5-MA
DISCHARGES AT TIME = 650 MSEC.

	Ohmic Heating Only			20 MW 120 keV 15 MW 60 keV
	Standard Impurities		No Iron	Standard Impurities
$n(0)^* (10^{13} \text{cm}^{-3})$	5	10	10	10
$T_e(0) (\text{keV})$	4.0	0.8	2.5	4.1
$T_i(0) (\text{keV})$	3.1	0.8	2.3	4.2
$n_{\text{Fe}}(0) (10^{11} \text{cm}^{-3})$	0.8	1.9	-	2.1
$n_{\text{Fe Max}} (10^{11} \text{cm}^{-3})$	1.0	3.6	-	9.1
$q^{-1}(0)$	1.0	1.0	1.0	1.0
β_p^{**}	0.15	0.07	0.22	0.48
V (volts)	1.3	2.2	1.0	0.7
$P_{\text{Rad}} (\text{MW})$	1.2	7.0	0.3	32.5
$\tau_E (\text{sec})$	0.3	0.1	0.6	0.1

*Refers to the value at $r = 0$.

**Includes superthermal ions.

3.2.3 INJECTION AND COMPRESSION

This section considers 1-MA tokamak discharges that are preheated by injection of 20 MW of deuterons at 120 keV, along with 15 MW at 60 keV, during intervals of 75-500 msec, and are then compressed adiabatically. (The half-energy beam used in this computer model represents a rough approximation of the real situation, where part of the power is in 1/3-energy neutrals — making things worse — but where there tends to be somewhat lower total power in fractional-energy neutrals — making things better, on the whole.

3.2.3.1 The Weak Compression Mode

If one wishes to attain transient high-temperature plasmas while minimizing impurity influx, it is logical to pass through the neutral-beam injection stage as soon as possible, then remove the discharge from limiter contact and surround it with a cold plasma "protection blanket" (cf. Section 3.1.4). Figure 3.2-4 shows the effects of neutral-beam preheating for various durations (75, 100, 150, 300, and 500 msec),

Figure 3.2-4. Weak-Compression Modes with Various Intervals of Neutral-Beam Injection; 75, 100, 150, 300 and 500 msec. (PPL 753880)

after which the plasma is weakly compressed, as in the illustration of Mode A in Figure 2.1-2. ($C = 1.25$ is used here for the compression factor.) As expected, there is a tendency for T_e and T_i to increase with the length of the injection period, but finally to saturate and decline again when the injection period exceeds $\sim \tau_E$. In the latter case, the impurity influx increases rapidly, and counteracts the beneficial effects of greater beam energy input. The portion of the DT reaction energy that is released during the interval of principal interest for reactor plasma studies (i.e., following compression) is greatest for plasmas that are preheated for 100-150 msec periods. The best combination of post-compression parameters (maximum T_e and T_i with minimum n_{Fe}) is obtained for the 100 msec injection period. This case has therefore been adopted as the reference embodiment of the weak compression mode (A) in Table 3.2.3.1, and in Figure 3.2-5, which shows the plasma density and temperature profiles at the particular time (200 msec) to which the tabulated parameters refer.

A pulsed gas input is applied (by raising the boundary neutral density to $6 \cdot 10^{10} \text{ cm}^{-3}$) during the period 155 to 180 msec, just after compression; the edge region then fills with cold, dense plasma, and the particle content of the discharge is ultimately doubled. In Figure 3.2-5, the intended outer density peak has begun to emerge; shortly afterwards, $n(r)$ assumes the shape of the right-hand illustration in Figure 3.1-7. The central impurity density quoted for the weak compression case in Table 3.2.3-1 is fairly substantial: This impurity level is due to a large influx into the discharge during the preheating period, when the hot plasma core is in close proximity to the wall. Following compression, the impurity content actually declines, so that the central impurity density is lower at 300 msec than at 200 msec.

The temperature profile shown for Mode A in Figure 3.2-5 has most of its gradient in the trapped-ion mode regime. (The sharp "corners" in this and the other profiles should not be taken too literally, since they are related to the mechanics of the 6-regime fitting process.) The flat top in the T_e -profile is not due to MHD-instability: the inverse central safety factor $q^{-1}(0) = 1(0)/2\pi$ remains slightly below unit for the entire time of interest.

The total DT reaction yield for the discharge is 1.6 MJ. For purposes of calculating a Q-value during some appropriate interval, the guideline (defined in Section 3.3.1.2) that the interval considered must be representative of a possible steady-state process was applied. This means that the plasma thermal energy must be at least as high at the end of the interval as at the beginning. In the case at hand, there is 1.0 MJ of energy left in the deuterons that were injected at 120 keV, just after compression. There is also 0.4 MJ left in the deuterons that were injected at 60 keV, which do not participate appreciably in the reaction process. For an ideally simple Q-determination, the injection system should be designed to avoid producing — or to remove — these half-energy deuterons; however, the importance of an exact Q-determination is not sufficient to warrant this extra technological complexity at the present stage of neutral-beam development. The initial presence of the low-energy

Figure 3.2-5. Radial profiles for the reference modes: (a) Weak-compression mode at 200 msec ($a = 85$ cm); (b) Strong-compression mode at 200 msec ($a = 45$ cm); (c) Low-current quasi-static mode at 400 msec ($a = 85$ cm); (d) High-current quasi-static mode at 400 msec ($a = 85$ cm). (PPL 753962)

TABLE 3.2.3-1. PARAMETERS OF THE REFERENCE OPERATING MODES

Mode:	Injection Power: 20 MW at 120 keV, 15 MW at 60 keV			
	Transient Regimes		Quasi-Static Regimes	
	Weak Compression C = 1.25	Strong Compression C = 1.5	Compression Prior to Injection	
	(A)	(B)	(C)	(D)
1 (MA)	1.0	1.0	1.0	2.5
Tabulation Time (Sec)	0.2	0.2	0.4	0.4
$n(0)$ (10^{13}cm^{-3})	8	11	6	5
$T_e(0)$ (keV)	8.6	14.1	7.0	9.9
$T_i(0)$ (keV)	6.4	12.4	5.4	11.2
$n_{\text{Fe}}(0)$ (10^{11}cm^{-3})	2.7	3.7	4.8	0.8
$n_{\text{Fe max}}(10^{11}\text{cm}^{-3})$	2.7	8.1	6.8	5.1
$q^{-1}(0)$	0.9	0.9	1.0	1.0
β_p	2.5	3.9	1.8	0.6
V(volts)	0.1	0.7	0.3	0.9
$\tau_E(\text{sec})$	0.2	0.2	0.1	0.4
	Total Yield	Total Yield	Up to 0.4 sec	Up to 0.7 sec
DT Yield (MJ)	1.6	4.0	2.0	3.6
Q	0.9	1.4	0.4	0.7

dueterons is compensated simply by requiring that the reaction interval considered for the Q-measurement end at a time when there is at least 0.4 MJ of energy left in the total injected-ion population. The reaction interval so defined extends from 154 to 240 msec; the fusion yield in this interval is 0.9 MJ. Since 1.0 MJ of high-energy ions is consumed in the process of producing this yield, we have $Q = 0.9$. The plasma thermal energy rises from 1.7 to 2.1 MJ during the reaction interval, more than satisfying the relevant condition.

3.2.3.2 The Strong-Compression Mode

If one wishes to break through to the highest possible plasma temperatures and densities, it is obviously plausible to increase the adiabatic compression factor. In order to realize the maximum geometrically possible factor of $C = 1.5$, the hot plasma must be positioned close to the wall both during the preheat stage and after compression (cf. Figure 2.1-2, Mode B). As a result, there is a continuous rise of the discharge impurity content. Very large temperatures, densities and fusion yields can be reached, however, before the impurity content becomes excessive.

Figure 3.2-6 illustrates the effect of various preheating pulse durations on the strong compression mode. The precompression plasma is somewhat different from that in Figure 3.2.4 (lower density and lower current), since the same final-stage parameters of $n(0) \sim 10^{14} \text{cm}^{-3}$, $I = 1 \text{ MA}$ are desired in both cases. The optimum injection interval, however, again turns out to be 100 msec. The parameters reached for this case at $t = 200 \text{ msec}$, i.e., 50 msec after compression, are shown in Table 3.2.3-1, Mode B. The plasma energy density is indeed found to be high — about five times that in the present ATC experiments. The β_p -value, which includes superthermal ions, is almost 4; this, however, is an artifact of the present computer model: For a compression carried out on a realistic time scale of $\sim 30 \text{ msec}$, rather than instantaneously, the superthermal ions would not be energized so strongly (cf. Section 3.3.1.4), and the peak β_p -value would be about 3.

The impurity density is greater than in the weak compression mode, and is rapidly rising; the transient $n\tau_E$ -value, during the reaction interval of interest, however, is large: $n\tau_E \gtrsim 2 \cdot 10^{13} \text{cm}^{-3} \text{ sec}$. The Q-value is calculated as in the preceding section: the reaction interval extends from 154 msec to 280 msec, during which time a high-energy ion group of 1.9 MJ energy content and a low-energy ion group of 0.7 MJ content are reduced to a total low-energy ion content of 0.7 MJ. During this interval a fusion yield of 2.7 MJ is produced, so that we obtain $Q = 1.4$. The plasma thermal energy rises in the same interval from 2.0 to 2.9 MJ. The T_e -profile corresponding to Mode B at $t = 200 \text{ msec}$ is shown in Figure 3.2-5. It is similar to that of Mode A, but lacks the protective cold-plasma blanket.

Figure 3.2-6. Strong-Compression Modes with Various Intervals of Neutral-Beam Injection: 75, 100, 150, 300, and 500 msec. (PPL 753881)

The most practical mode of operation in the actual conduct of the TFTR experiment is likely to be intermediate between Modes A and B. The optimal compression factor will be selected on the basis of the relative importance of impurity protection, as against compressional heating power. The ability to adjust the experimental operation in the appropriate way represents one of the important elements of flexibility of the TFTR device.

3.2.3.3 Variation of the Transport Coefficients

Perhaps the least reliable physics ingredient in the present computations is the magnitude of the plasma transport assumed in the highly collisionless (trapped-particle) regimes. In order to convey some idea of the sensitivity of the TCT projections to the assumptions about this transport, the discharges of Modes A and B have been recalculated using five times larger and five times smaller numerical coefficients for the transport coefficients of the trapped-particle instabilities. The results for all three versions are illustrated in Figures 3.2-7 and 3.2-8. The most optimistic assumption leads to temperatures that are almost twice as great as the least optimistic, along with impurity levels that are no worse. On the other hand, even the most pessimistic outcome yields plasma parameters that still satisfy the basic TFTR objectives, and produces respectable DT fusion yields of 1-3 MJ.

3.2.4 QUASI-STATIC INJECTION

The operating modes of Section 3.2.3 make use of transient effects to guard against impurity influx and MHD instability, thus promoting the attainment of reactor-like plasmas. The techniques involved are directly relevant to the concept of a pulsed tokamak reactor (Section 3.3.3); the main line of planning for tokamak reactors at present is, however, oriented toward quasi-static or even steady-state operation. In this context, it is of great interest to provide the TFTR device with the capability of exploring quasi-static regimes, lasting for a number of energy confinement times. If serious difficulties develop in this type of operation - as would be predicted by the present computer studies - they can be explored, with a view to developing appropriate remedies. If conditions turn out to be better than now expected; it will be desirable to take advantage of the situation to generate and study quasistatic reactor-grade plasmas. In that case, it may even prove possible to reach values much higher than $10^{13} \text{ cm}^{-3}\text{sec}$ and ultimately to extend the duration of the high temperature regime into the range of several seconds. For initial planning purposes, however, the prospects of the quasi-static TFTR regimes have been calculated on the basis of the same "pessimistic" assumptions that were used in the preceding sections.

The TFTR neutral-beam system is able to inject into either the precompression or postcompression plasma location (cf. Section 4.5). The following calculations assume injection into the post compression location, with pulse durations of 500 msec, using the reference injection system with 20 MW of 120-keV and 15 MW of 60 keV deuterons, unless otherwise specified.

Figure 3.2-7. Effect of varying the thermal and particle diffusion coefficients of the trapped-particle modes by factors of 5 relative to the reference calculation for Mode A. (PPL 753876)

Figure 3.2-8. Effect of varying the thermal and particle diffusion coefficients of the trapped-particle modes by factors of 5 relative to the reference calculation for Mode B. (PPL 753882)

3.2.4.1 Low-Current Discharge

An initial ohmic-heated plasma like that in Mode A is weakly compressed ($C = 1.2$) at $t = 50$ msec; subsequently, the neutral-beam injection is begun. The results for the reference case are shown in Figure 3.2-9 (top). The plasma parameters at $t = 400$ msec are tabulated in Table 3.2.3-1 for Mode C, and corresponding radial profiles are given in Figure 3.2-5(C). In Mode C, the creation of a dense outer plasma layer would interfere with injection; accordingly, no special pulsed-gas input is used after compression. Instead, the initial compressed plasma (peak density $\sim 10^{14}\text{cm}^{-3}$), spreads out to fill the limiter aperture and eventually recycles at the plasma edge. At $t = 400$ msec, the resultant $n(r)$ profile begins to resemble that for Mode A, but the magnitude of the density is substantially lower. At this time, the injection process has also begun to suffer seriously from the impurity-barrier problem illustrated in Figure 3.1-9; the calculation was terminated shortly thereafter. The DT yield up to $t = 400$ msec is 2 MJ, and Q -value calculated at the most favorable time ($t = 200\text{--}300$ msec) from the ratio of fusion power to injection power is only $Q = 0.4$, even if one neglects the presence of the half-energy beam, which is stopped at the plasma edge.

Figure 3.2-9 also considers the case of a beam with 12 MW at 150 keV and 12 MW at 75 keV, which would be obtainable from the TFTR neutral-beam power supply if 150-keV operation were introduced as an experimental option. The advantage of increased beam penetration is offset by a reduction in total beam power, which leads to a reduction in the initial rate of temperature rise; nonetheless there is a mild improvement in every respect.

In particular, the greater-time-integrated fusion probability of 150-keV deuterons leads to a slightly larger peak fusion power, in spite of the reduced injection power. To prolong the useful injection period significantly beyond that of the reference Mode C, however, would require a much more drastic change in the neutral beam parameters — or in the physics assumptions of our code.

3.2.4.2 High-Current Discharge

The most interesting prospect for quasi-static injection is the 2.5-MA discharge of Figure 2.1-2, Mode D. If MHD stability turns out to be acceptable for the corresponding limiter q -value of 3, then even within the assumptions of the present code an interesting plasma regime can be produced. In order to attain high temperatures with the available neutral-beam power, the central plasma density is best lowered to $\sim 5 \cdot 10^{13}\text{cm}^{-3}$; the lower density also facilitates penetration of the beam through the plasma edge region. As was shown in Section 3.2.2, a central plasma density of $5 \cdot 10^{13}\text{cm}^{-3}$ lends itself to the maintenance of a long-lived discharge with a normal tokamak profile, even if only ohmic heating power is available. Without neutral-beam injection, one reaches $T_e \sim 4$ keV at $t = 650$ msec; and the temperature continues to rise somewhat while the density falls. (In the present calculations Mode D has been initiated in a simplified way, by locating the full-current discharge at the final position. A more detailed initiation procedure is followed by our zero-dimensional code, and has been given in Figure 3.1-3.)

Figure 3.2-9. Low-current quasi-static injection modes: (Top) Injection of a 120-keV beam as in Mode C; (Bottom) Injection of a 150-keV beam. (PPL 75878)

If neutral-beam injection is carried out in the interval $t = 150\text{--}650$ msec, peak temperatures around 10 keV can be reached for both electrons and ions. Representative data for this discharge are given in Table 3.2.3-1 (D), in Figure 3.2-5 (D), and in Figure 3.2-10 (bottom left). After $t = 500$ msec, the growth of the outer impurity barrier puts an end to the period of useful injection, but a total DT yield of 3.6 MJ is reached. The Q-value computed at the most favorable time ($t = 300\text{--}400$ msec) is 0.7, if one neglects the presence of the nonpenetrating half-energy beam. The confinement time τ_E in this phase is exceptionally long (0.4 sec), and the $n\tau_E$ -value is about the same as in Mode B ($n\tau_E \sim 2 \cdot 10^{13} \text{ cm}^{-3} \text{ sec}$). Again the relative performance of the 150 keV injection system has been examined; the effect on the discharge, as shown in Figure 3.2-10 (bottom right), is very slight, but the computed Q-value is raised above unit, since the injection power has been lowered. The appearance of the edge impurity barrier is again evident. Further computer studies will be carried out to test the effect of variations in the treatment of the impurity influx.

3.3 REACTOR PHYSICS ASPECTS

Initial experimental operation of the TFTR device will utilize "hydrogen plasmas;" if the neutral-beam injectors are to be used effectively, this means plasmas that are mostly hydrogen will be heated by deuteron beams. Even in this basic plasma physics phase of the experiment, there will be an appreciable level of fusion events (about $10^{16}/\text{sec}$) from the reactions

The dominant contribution to this yield will come from the head-on collisions of injected deuterons — as in the CIT reactor scheme described in Ref. 21. If deuterium target plasmas are used instead of hydrogen, the total yield is increased only by a factor of three or so. The elaborate shielding provisions made for experimental operation in the DT-reactor phase will thus begin to be useful already in the "hydrogen" phase. The principal objective in this initial experimental period will be to create reactor-like plasmas and investigate the physics of stability, heating, and energy confinement in the reactor plasma regime. In view of the operating experience that will have been gained meanwhile on precursor experiments, such as PLT, it is possible that satisfactory reactor-like plasma regimes will be achieved and analyzed in TFTR within the first one to two years of experimental operation. At this point, it should be possible to predict with accuracy the fusion yields that will be obtained when using the reaction

The TFTR device will then be ready to enter its Tokamak Fusion Test Reactor phase.

Figure 3.2-10. Discharge modes with 2.5 MA currents: (Top) Ohmic heating only; (Bottom left) Injection of a 120-keV beam, as in Mode D; (Bottom Right) Injection of a 150-keV beam. (PPL 753879)

The DT-phase of experimental operation will be initiated with small tritium admixtures in deuterium bulk-plasmas. Following the demonstration of successful operation of the tritium-handling and neutron-diagnostics systems, the tritium plasma fraction will be increased, and the maximum-fusion-power regime will be realized. The reactor experience of the TFTR will provide a key input into subsequent tokamak fusion reactor projects. It is most specifically relevant to two lines of experimental reactor development that are likely to prove useful in the period before full-scale steady-state tokamak reactors, such as those of Ref. 22 and 23, are economically feasible: (1) The two-component technique can be used to produce relatively high fusion power in a modest-sized tokamak installation.^{24,25} (2) The pulsed-heating technique can be used to achieve transient ignition and Q-values substantially greater than unity, even before the technical problems of steady-state operation have been solved.²⁶ The features that make these two concepts attractive as near-term tokamak power reactor experiments correspond directly to the simplifying features of the TFTR device that have made it a practical near-term project: (1) A two-component reactor can operate with relatively low $n\tau_E$ -values. (2) A pulsed-ignition reactor can make use of transient techniques to control the current distribution and the plasma composition. Various aspects of tokamak reactor physics relating to the two-component and pulsed-ignition approaches are reviewed in the following two sections.

3.3.1 TWO-COMPONENT REACTOR REGIME

3.3.1.1 Deuteron Slowing-Down Process

As illustrated in Figure 3.3-1, a 180 keV deuteron slowing down in a tritium plasma of $T_e = 5$ keV gives up its energy at first mostly by collisions with the electrons, but later mostly by collisions with the ions. In the latter phase, the deuteron direction of motion may also change appreciably by scattering. The slowing-down time τ_s is characterized by $n\tau_s \sim 10^{13}\text{cm}^{-3}\text{sec}$. The presence of impurity ions, with higher charge and mass, does not have a strong effect on the deuteron slowing-down rate, but contributes strongly to scattering of the directed motion when the corresponding Z_{eff} is large.

The steady-state distribution function of beam-injected deuterons slowing down in this manner takes the form

$$f_b(v) \sim (v_c^3 + v^3)^{-1} \text{ for } v \leq v_{bo},$$

where

v_{bo} is the injection velocity and we define

$$v_c^3 = 3\pi^{1/2} T_e^{3/2} / (2m_e)^{1/2} m_i.$$

This distribution is monotonically decreasing and isotropic, like the ion thermal distribution, but with a much enhanced high-energy component. The steady-state distribution function for slowing-down of parallel-injected beams (as in the TFTR device) is illustrated in Figure 3.3-2. It is anisotropic and mildly nonmonotonic.

Figure 3.3-1. Classical slowing-down of an energetic deuteron in a 5-keV tritium plasma. (The $1n\Lambda$ factors are calculated at $n = 3 \times 10^{13} \text{cm}^{-3}$.) The energy multiplication factor is $Q = \text{Integrated fusion probability} \times 17.6 \text{ MeV}/W_0$. (PPL 743432)

Figure 3.3-2. Energetic-ion velocity distributions (a) parallel and (b) perpendicular to the magnetic axis. The direction of injection is 10° to the magnetic axis. v_{bo} denotes the injection velocity. (PPL 753980)

The fractional energy $\bar{\Gamma} = (\text{Superthermal Ion Energy})/(\text{Plasma Thermal Energy})$ residing in a steady-state population that is maintaining the plasma temperature follows from the condition that the input power $(\text{Injection Energy})/\tau_s$ should equal the plasma-loss rate $(\text{Plasma Thermal Energy})/\tau_E$; thus the equation $\bar{\Gamma} \sim \tau_s/2\tau_E$. For TCT parameters $\bar{\Gamma} \sim 1/2$, is expected; this is the typical situation in the computer studies of Section 3.2.

3.3.1.2 Energy Multiplication

In the example of Figure 3.3-1, the injected deuteron has a sufficiently large time-integrated fusion probability ($\sim 10^{-2}$) so that the energy multiplication factor Q exceeds unity, where one defines:

$$Q = (\text{Expected No. of Fusion Events}) \times 17.6 \text{ MeV}/(\text{Injection Energy}). \quad (1)$$

From slowing-down curves such as the one in Figure 3.3-1, a table of energy multiplication factors can be constructed. In a finite-temperature bulk plasma, there is, however, considerable random variation in the slowing-down rate of individual deuterons; taking into account this effect, as well as the finite velocity of the reacting bulk-plasma tritons, ^{17,27} leads to the Q -curves in Figure 2.1-3. The results turn out to be somewhat more favorable than those given by the simpler approximation, which was the one used in the older TCT literature.

In comparing the theoretical Q -value [Eq. (1)] with the results of an ideal steady-state experiment, one can write

$$Q = \frac{(\text{Reaction Rate}) \times 17.6 \text{ MeV}}{\text{Power Input}} \quad (2)$$

Actual near-term reactor experiments will necessarily be transient, and so it is logical from the physics point of view to apply the definition (2) to a selected interval during which the plasma and beam parameters are approximately constant. The experimental interval serves as a physically representative slice of a possible steady-state reactor regime, i. e., one of sufficiently long duration so that the energy invested in setting up the regime can be neglected. In quoting illustrative Q -values for the quasi-static Modes C and D of Section 3.2.4, the definition (2) was used. The further stipulation was made that power injected in half-energy deuterons and deposited outside the hot-plasma region should not be taken into account. (Further experiments with more nearly monoenergetic neutral beams will be able to avoid this complication.)

For the case of pulsed-reactor experiments, such as Modes A and B of Section 3.2.3, Eq. (1) is written in the form

$$Q = \frac{\int_{t_0}^{t_0 + \tau_R} (\text{Reaction Rate}) dt \times 17.6 \text{ MeV}}{\text{Superthermal Ion Energy Content at } t = t_0} \quad (3)$$

The reaction interval τ_R can be regarded as a representative cycle of a possible cyclic reactor regime where there is a sufficient number of sequential reaction cycles so that the energy invested up to time t_0 in preheating the plasma can be neglected. In order for this interpretation to be valid, the bulk-plasma thermal energy content must be at least as high at $t_0 + \tau_R$ as it was at t_0 . In the illustration of Section 3.2.3, a further complication is introduced by the presence of deuterons injected at half-energy, which make little contribution to the fusion yield, but help to maintain the plasma temperature. To allow for their initial presence, it is stipulated that at time $t_0 + \tau_R$ there should be as much superthermal energy content left in the total injected-ion population as there was initially in the half-energy component. The Q -value of Eq. (3) is then evaluated using only the energy content at $t = t_0$ of the deuterons that had been injected at full energy. (This procedure gives somewhat lower Q -values than would be obtained if all the ions could actually be injected at full energy.)

While the realization of Q -values "of order unity" is one of the objectives of the TFTR project, the precise numerical achievement and experimental documentation of $Q > 1$ has not been given major emphasis. From the reactor-engineering point of view, the definition of Q employed in Eqs. (1-3) is a highly idealized one, since factors such as the efficiency of beam injection and of energy recovery are not taken into account. The technical complications involved in attempting to produce truly monoenergetic neutral beams (or, for that matter, to obtain measurements of the total fusion yield to an accuracy better than $\pm 30\%$) appear too great at the present time to warrant a diversion of the TFTR effort from its primary objectives.

3.3.1.3 Fusion Power Density

The total plasma pressure is limited by

$$\beta_p = 8\pi \langle n(T_e + T_i) + (2/3)n_b W_b \rangle / B_p^2 (a) < R/a,$$

where n_b and W_b are the density and energy of the beam-injected ions. In ordinary thermonuclear reactors where $n_b W_b$ is negligible, the maximum fusion power density in the presence of the β_p -constraint is obtained at $T_e \sim T_i \sim 15 \text{ keV}$. Higher-temperature plasmas have higher reactivities,

but their density must be lowered disproportionately. A two-component ion population has the advantage of achieving high-cross-section relative DT energies without requiring a high bulk-plasma thermal pressure. The resultant advantage in power density of the TCT regime has been illustrated in Figure 2.1-5. Detailed calculations of TCT power optimization have been carried out in Ref. 28. A direct comparison between the fusion power of a thermonuclear 50:50 DT-plasma and an optimal TCT regime of the same total plasma pressure is given in Figure 3.3-3(A): the TCT power from a 5-keV bulk plasma is shown to exceed by a factor of 5 the ordinary thermonuclear power density at any plasma temperature. At still lower TCT temperatures, still higher power densities can be achieved, but these regimes are of reduced practical interest because the Q-value deteriorates. In regard to sensitivity to plasma impurity contamination, Figure 3.3-3(B) shows that the fusion power density of the TCT can be affected only by extremely large impurity admixtures — provided, of course, that beam penetration takes place.

In computer studies of Section 3.2, peak fusion power densities of about 1 Wcm^{-3} are obtained in the quasi-static Modes C and D; the highest values are given by the strong-compression Mode B, which reaches $\sim 5 \text{ Wcm}^{-3}$. Whereas the predicted maximum power density for an ordinary thermonuclear regime would be quite sensitive to the degree of pessimism of the plasma-physics assumptions made, the range of power densities will be achieved in the TFTR can be expected to be substantially the same as the predicted values.

3.3.1.4 Energy Clamping

The deuteron energy that produces the maximum fusion reaction rate for a given power input is in the range 105 to 145 keV (depending on T_e). In standard TCT operation, deuterons are injected at energies well above the optimum value and pass through it as they slow down (cf. Figure 3.3-1). If the deuterons could be injected directly at the optimum energy and "clamped" there by means of a post-injection energy input, the system Q-value could be roughly doubled. This concept is discussed in Refs. 29 and 30; the results are illustrated in Figure 3.3-4.

The energy clamping idea could be implemented in a variety of ways. The most straightforward is to carry out the Mode B compression on the expanded time scale of the deuteron slowing-down time $\tau_s \sim 100 \text{ msec}$, so as to maintain the injected deuteron energy at $\sim 120 \text{ keV}$ instead of boosting it into the $\gtrsim 200\text{-keV}$ range.²⁹⁻³¹ A closely connected plasma-handling technique with reactor implications is adiabatic decompression, which reconverts the energy of the injected ions, the bulk plasma, and the alpha particles, directly into electric energy of the EF circuit. These "exotic" techniques can be studied on the TFTR device, simply by taking advantage of available system capabilities.

Figure 3.3-3(A). Comparison of fusion power density for a two-energy-component plasma with optimal $\bar{\Gamma}$ and a 50% D, 50% T thermal plasma ($\bar{\Gamma} = 0$). In each case, total plasma pressure = 0.655 J/cm^3 , corresponding to $B_t = 6 \text{ T}$, $\beta_p = R/a = 3.5$, $q = 2.5$. Alpha-particle effects are neglected. (PPL 743654)

Figure 3.3-3(B). Reduction in the maximum attainable fusion power density P_f and the corresponding power multiplication Q_b for 200-keV deuterons injected into a 6-keV triton target plasma, containing a single impurity species of charge Z_I with concentration Z_{eff} . P_{f0} and Q_{b0} are values for $Z_{eff} = 1$. (PPL 743839)

Figure 3.3-4. (a) Optimal beam energies for maximum energy multiplication: W , in standard TCT operation; W_c , with energy clamping. (b) Energy multiplication factors: $(Q)_{\max}$ in standard operation; $(Q_c)_{\max}$ with energy clamping. Deuteron beams and pure tritium target plasmas are assumed. (PPL 743740)

3.3.1.5 Stability of the Two-Component Plasma

In view of the high fractional superthermal ion energy density $\bar{\Gamma} \gtrsim 0.5$ of the TCT regime, it is natural to inquire whether new types of instability might not make their appearance. This critically important topic has been investigated intensively during the past two years by theorists at a number of plasma research institutions.^{17,32-36} The conclusion is that steady-state slowing-down velocity distributions in TCT regimes with parameters of practical interest are generally too weakly anisotropic or nonmonotonic to drive dangerous instabilities. For parallel-anti-parallel steady-state injection, the most interesting mode is the finite-medium drift-Alfven wave described in Ref. 34; but even this mode is predicted to result, at worst, in a mild limitation on the allowed degree of central peaking of $n_b(r)$. The presence of the superthermal ion component does not serve to enhance ordinary plasma micro-instabilities, such as the drift waves and trapped-particle modes, nor does the spontaneous presence of these modes in the bulk plasma have an appreciable effect on the slowing-down or confinement of the superthermal ion component.

The study of two-component microinstabilities is of considerable intrinsic interest, since the results apply not only to the TCT regime, but to any kind of neutral-beam heating, and also to alpha-particle heating in ignited plasmas. If it is desired to study the excitation of such modes on TFTR, the most promising approach will be the creation of non-steady-state distribution functions by means of high-powered injection pulses, with superthermal ion buildup times shorter than τ_s . (Perpendicular injection should also be more strongly destabilizing than parallel injection; appropriate access ports would be available on TFTR.)

In addition to the favorable theoretical predictions, there is substantial experimental evidence to support the essential stability of two-component plasmas. On ORMAK and ATC, operation at the highest available injection power levels has resulted in the attainment of $\bar{\Gamma}$ -values of 0.4-0.8, with no sign of instability excitation. Detailed studies of the slowing-down process, based on the charge-exchange neutrals and the DD neutrons produced by injected ions in the 15-30 keV range, have shown excellent agreement with the predictions of the classical theory.^{37,38}

3.3.1.6 Applications of TCT-Type Reactors

Potential applications of TCT-type reactors include fusion engineering test facilities, and devices that would contribute to various facets of the nuclear fuel cycle in a fission power economy:^{39,40} the breeding of fissionable material, power production from a fissionable blanket (hybrid reactors), and the burning of long-lived actinide (transuranic) wastes. For these applications, the most important plasma consideration tends to be maximization of fusion power density, P_f , in order to minimize the device size (i. e., cost) for a given neutral wall loading.²⁸ At the same time, economic considerations demand that $Q \gtrsim 1/2$, since the investment in neutral-beam injectors, and the amount of electrical power and tritium that must be circulated, are inversely proportional to Q . Finally, the large duty factor required for satisfactory neutron fluence dictates a minimum burn time per cycle of several tens of seconds.

Given these general constraints, practical TCT reactors tend to have similar plasma parameters:

$$T_e \sim T_i \sim 5 \text{ to } 8 \text{ keV}, \quad n_e \sim 10^{13} \text{ cm}^{-3}, \quad W_{inj} = 150 \text{ to } 300 \text{ keV}.$$

Table 3.3.1-1 compares the TFTR parameters with those specified in illustrative conceptual designs of a tokamak engineering test reactor,²⁴ and a fusion-fission hybrid reactor.²⁵

The test reactor would take advantage of the 1 MW/m^2 neutron wall loading, which can be obtained over a useful test area of many tens of m^2 , to investigate bulk-radiation effects, as well as the heat-transfer performance of various wall materials and blankets. It could also serve as a test-bed for fusion reactor systems such as those for tritium breeding and processing.

TABLE 3.3.1-1 PARAMETERS OF TCT-TYPE REACTORS

	TFTR	Engineering Test Reactor ²⁴	Hybrid Reactor ²⁵
R_o (m)	2.5	3.05	4.3
a (m)	0.85	0.55	1.1
A	2.9	5.6	3.9
Plasma Elongation	1.0	2.0	1.0
B_t on axis (T)	5.2	3.8	5.0
I_p (MA)	2.5	2.4	2.8
q , at limiter	3.0	2.5	2.5
$\bar{n}_e$ (cm ⁻³)	5×10^{13}	1.2×10^{14}	1.1×10^{14}
$\bar{T}_e = \bar{T}_i$ (keV)	5.0	5.0	6.0
$\bar{n}_e \tau_E$ (cm ⁻³ s)	$\sim 10^{13}$	8×10^{12}	1×10^{13}
W_o (keV)	120/60	200	200
P_{beam} (MW)	35	177	360
$\bar{\Gamma}$	~ 0.5	0.85	0.84
β_p	2.0	3.5	3.5
Q_b (ideal)	~ 1	1.0	1.2
P_f (W/cm ³)	~ 1	4.3	4.3
Total neutron flux (n/s)	$\sim 5 \times 10^{18}$	6.4×10^{19}	1.6×10^{20}
Neutron wall loading (MW/m ²)	~ 0.1	0.96	1.4
Pulse duration (s)	0.5	50	50
Duty factor	0.002	0.8	0.8

A fissionable blanket of natural or depleted uranium would allow a high- P_f , low- Q TCT hybrid reactor to serve as a power generator, while producing plutonium for use in thermal fission reactors. With a plant factor of 75%, a beam-injector efficiency of 75%, and a depleted-uranium, fast-fission blanket, the hybrid-reactor of Ref. 25 would have an average net electrical power production of 1050 MW(e), and a net Pu-239 production of 9 kG/day. (At the same time, the tritium breeding ratio would be 1.05.) The overall plant thermal efficiency would be 30%. If an appreciable fraction of the plutonium generated is left in the hybrid blanket, the

power output is enhanced considerably, while the fissile doubling time can be relatively short.⁴⁰ Breeding of ^{233}U in a thorium blanket may possibly be economically viable if it is sold for use as makeup fuel in fission reactors of very high conversion ratios, such as the HTGR and CANDU reactors operating on the thorium cycle.

In a variant of the hybrid reactor, it is proposed to transform long-lived actinide waste products from fission reactors into shorter-lived radioactive waste products.^{39, 41} For this purpose, high-energy neutrons are most effective; the broadening of the energy spectrum of DT neutrons from the TCT plasma due to the unusually high energy of the reacting deuterons turns out to be significantly helpful.⁴² Since the actinides are transformed primarily by fission reactions, many tens of MeV are generated per fusion neutron, so that a TCT actinide waste burner would also perform as a generator of considerable electric power.⁴¹ In the design of Ref. 41, the half-life for transmutation of hazardous long-lived actinide isotopes would be less than ten years; the total thermal power production would be about 10,000 MW.

As evident in Table 3.3.1-1, most of the plasma parameters of the TFTR are already comparable to those of the illustrative reactor applications. The chief distinctions between TFTR operation, and operation of an economically practical TCT neutron source, are pulse length and duty factor. The plasma engineering problems associated with obtaining pulse times $\gtrsim 50$ sec include steady-state beam injection, particle exhaust, tritium fueling, and longtime impurity control. Their resolution will probably have to await the results of divertor experiments, such as PDX, and subsequently the implementation of fueling and divertor systems on a more advanced tritium-burning device, such as the engineering test reactor.

3.3.2 "PULSED IGNITION" STUDIES

3.3.2.1 Alpha Particle Heating

The attainment of truly ignited thermonuclear plasmas in the TFTR facility seems improbable, since an $n\tau_E$ -value exceeding $3 \cdot 10^{14} \text{cm}^{-3} \text{sec}$ would be required even in a perfectly pure DT plasma of 10 keV temperature. Here the concept of "ignition" refers to a situation where the plasma energy losses in quasi-steady state are compensated by the fusion-generated alpha-particle heat input to the plasma. In transient regimes, a much more lenient condition can be applied: if the plasma is initially far from wall contact, then its temperature can be maintained by the alpha particle heating — provided only that the heating power is stronger than the radiated power. In pure DT plasmas, the corresponding requirements, which is often called the "ignition condition," is merely that the plasma temperature should exceed ~ 5 keV. There is, of course an $n\tau$ -type condition for such effects to be seen at all: there needs to be time to build up the alpha-particle equilibrium population, which requires roughly $n\tau_{s\alpha} \sim 10^{13} \text{cm}^{-3} \text{sec}$. ($\tau_{s\alpha}$ is the alpha-particle slowing-down time.) This requirement corresponds exactly to the design objective

for the transient regimes of the TFTR. The fractional contribution of the DT alpha-particle heating in TCT operation is given by $\sim 0.2Q$. For the transient compressional modes of the TFTR, increments in T_e of order 20-30% are indeed computed, depending on whether or not the alpha-particle heating is present. More extensive computer studies are planned on this topic, including introduction of a more accurate model for the alpha particles, and extension of the studies to the bulk reaction of 50-50 DT plasmas. The alpha-particle orbit confinement in TFTR is representative of edge conditions in a larger reactor: Figure 3.1-8 shows that most of the alphas are confined even in the 1-MA discharge, but some 10-30% will intercept the limiter. In summary, it appears that, while the TFTR is most unlikely to become an ignition experiment in the practical steady-state reactor sense, it will be able to exhibit some of the alpha-particle physics of ignition, which takes place on the characteristic two-component reactor time scale of $n\tau \sim 10^{13} \text{cm}^{-3} \text{ sec}$.

Some expected effects of large alpha-particle populations on the bulk plasma are described in Ref. 26 and 43. As shown in Figure 3.3-5, the fractional energy $\bar{\Gamma}$ in alpha particles is appreciable in steady-state ignited regimes, giving rise to some of the same concerns about two-component plasma stability that have inspired the TCT-related investigation of this topic (cf. Section 3.3.1.5). As in the ordinary two-component case, it appears that the alpha particle population should not have any substantial adverse effect on bulk-plasma stability; bulk-plasma microinstabilities also should not cause substantially enhanced loss rates of alpha particles. One unique feature of the alpha particles is that their velocity exceeds the typical Alfvén velocity in tokamaks, while the injected-deuteron velocity is typically smaller. This circumstance could lead to the excitation of some modes that would have been stable in the ordinary TCT regime.³⁴ The very large "banana" orbits of the alpha particles might also be expected to cause unusual electrostatic effects at the plasma edge; our present estimates, however, are that these effects are weak.⁴³ Finally, it is clear that alpha particles trapped in the toroidal-field ripple (cf. Section 3.1.3.3) will escape with high probability, by drifting vertically to the vacuum wall. If one could measure the consequences of this drift for the edge plasma potential (and for the wall material) as a function of the strength of the TF-ripple, this would be a useful piece of engineering information for the design of reactor TF systems. As pointed out in Ref. 44, the basic orbit behavior of alpha particles with $\sim 3.5 \text{ MeV}$ energy could be given initial study — at reduced alpha intensities — in the stage prior to DT-operation, by making a He^3 plasma in TFTR and using the reaction

Figure 3.3-5 also shows the $\bar{\Gamma}$ -values associated with the neutral beams required to bring the plasma to ignition. Characteristically, they are the same magnitude as $\bar{\Gamma}_\alpha$; accordingly, the replacement of beam-injected ions with alphas, at ignition, might turn out not to introduce any new plasma-physical effects at all. In the same sense, the injected-ion effects in the TFTR experiments will tend to mask the alpha effects — unless the alphas produce distinctive new disturbances.³³ Such an outcome is entirely possible — and, from a practical point of view, of course, highly desirable.

Figure 3.3-5. $\bar{\Gamma}_\alpha$ = (alpha energy/bulk-plasma energy) in an ignited D-T plasma. $\bar{\Gamma}_b$ = (minimum beam energy/bulk-plasma energy) for deuteron beams injected to heat the plasma to ignition. (PPL 743711)

Considering the special status of alpha-particle-heated plasmas in fusion research, however, it seems worthwhile to have carried out a direct experimental test of alpha-particle-related effects. The DT-phase of the TFTR experiment is designed to represent a significant step in this direction.

3.3.2.2 Pulsed-Ignition Tokamak Reactor

A pulsed-ignition reactor has the advantages that no refueling or ash removal is required, appreciable impurity buildup can be avoided by transient impurity-control techniques, and the plasma current profile does not have time to shrink or expand under high-beta operation. The principal disadvantage of pulsed ignition schemes is that they tend to be characterized by low Q -values, because of the large energy investment in the ignited plasmas. In large reactors, this shortcoming can be avoided by using externally supplied energy to heat only a fractional-sized core to ignition, while the bulk of the plasma is heated by a thermonuclear burn that propagates outward into a cold, dense plasma mantle.²⁶

As shown schematically in Figure 3.3-6(A), the propagating burn can be established by plasma-handling techniques similar to those used in the strong-compression mode of the TFTR device. An ohmic-heated plasma is formed next to a limiter, and 200 to 300-keV D^0 beams are injected for heating the plasma to 6 to 9 keV. Major-radius compression then brings the final temperature to the 9 to 14-keV range, where ignition is attained. Table 3.3.2-1 compares parameters for an illustrative ignited core plasma, immediately after compression, with those of the Mode-B TFTR plasma, and with the compressed ATC plasma (cf. Figure 3.1-11(A)). The plasma radius, current, and expected $n\tau_E$ of the TFTR plasma are nearly the geometric means of the corresponding values of the pulsed-ignition device and the ATC — where the injection-compression technique has already been demonstrated successfully. The plasma densities and temperatures in the TFTR and ignited plasmas are comparable. Note that β_p for both these cases are near their MHD-limiting values; attaining this performance is one of the principal goals of the TFTR experiments.

After formation of the ignited core, neutral D and T gases are admitted to produce a cold outer plasma region that provides fusible material for the subsequent thermonuclear burn, and — as in proposed TFTR operation — served to isolate the ignited core from neutrals and impurities evolved from the wall. The energy deposition of fusion alpha particles tends to raise the temperature of the ignited core, but the MHD restriction on β_p forces the hot plasma to expand into the colder region where an inverted density profile is maintained, as in Figure 3.1-7. The burn propagates as a step heat wave,⁴⁵ the mechanism being a turbulent heat transport such that β_p remains at the MHD limit, with the plasma current programmed for $q = \text{constant}$. For an appropriately designed radial density profile, the pressure of the expanding plasma increases at roughly constant $T_e \approx T_i$, by virtue of the increasing density. For typical parameters, the entire plasma is ignited in less than 10 sec, regardless of size.²⁶ The discharge is terminated once the burn front reaches the first wall. The total fusion energy produced during the burn is 5 times the alpha energy deposited in the plasma. The fusion energy multiplication, relative to the neutral-beam and compression energy required for igniting the core plasma, is shown in Figure 3.3-6(B). Evidently, $Q \gg 10$ is attainable for moderate expansion ratios, a_m/a_0 .

Even for quasi-steady-state operation of the final plasma, the propagating-burn technique would be desirable for minimizing the required investment in neutral-beam energy.²⁶ Figure 3.3-6(C) shows that for the illustrative example of Table 3.3.2-1 ($R/a_m = 3, a_m/a_0 = 2$), the injected beams need supply less than 10% of the final plasma energy content. Another significant advantage is that the initial pre-compression plasma can be heated by relatively low-energy neutral beams.

Figure 3.3-6 (A). After formation of the initial ohmic-heated plasma, neutral beams are injected for heating to 6-9 keV. A subsequent major-radius compression ignites the plasma. Neutral gas is then admitted to form a cold-plasma blanket that provides fusible material for the burn wave propagating outward from the ignited core. The major radius is increased slowly during the burn. (PPL 743710)

Figure 3.3-6 (B). Fusion energy multiplication Q for a burn that propagates outward from core plasma of radius a_0 and is terminated when the burn front reaches a radius a_m . (PPL 743559)

Figure 3.3-6 (C). Fraction of the final energy content supplied by injected neutral beams. Upper curve is for no major-radius compression of the initial core plasma. (PPL 743520)

TABLE 3.3.2-1 PLASMA PARAMETERS AFTER NEUTRAL-BEAM HEATING AND
MAJOR-RADIUS COMPRESSION

	ATC	TFTR	Pulsed-Ignition Reactor
C	2.3	1.5	1.4
R (m)	0.39	2.1	15
a (m)	0.11	0.45	2.5
B_t (T)	4.1	6.2	6.0
$\langle n \rangle$ (cm^{-3})	$8 \times 10^{14}^*$	8×10^{13}	9×10^{13}
$\langle T_e \rangle$ (keV)	1.7*	~ 8	14
$\langle T_i \rangle$ (keV)	1.2*	~ 6	14
I_p (MA)	0.18	1.0	5.0
$\langle n \rangle \tau_E$ ($\text{cm}^{-3} \text{s}$)	$\sim 1 \times 10^{12}$	1.5×10^{13}	2×10^{14}
β_p	1.2	3.0	6

*peak values

REFERENCES

1. "PDX, The Poloidal Divertor Experiment," presented at the Joint EURATOM-US Workshop on Large Tokamak Designs, Culham Laboratory, 1-3 May 1974.
2. R. Hawryluk, J. A. Schmidt, "Effects of Low-Z Impurities During the Startup Phase of a Large Tokamak", Princeton Plasma Physics Lab. Report MATT-1201 (1976).
3. P. H. Rutherford, "TCT Start-UP: Phase I", private communication.
4. A. H. Glasser, J. M. Greene, J. L. Johnson, "Resistive Instabilities in a Tokamak", PPPL Report MATT-1163 (1975).
5. J. L. Johnson, private communication.
6. M. N. Rosenbluth, private communication.
7. W. M. Tang, P. H. Rutherford, H. P. Furth, J. C. Adam, Phys. Rev. Letters 35 (1975) 660.
8. W. Horton, Jr., "Drift-Wave Stability of Inverted Gradient Profiles in Tokamaks", submitted to Phys. Fluids.
9. G. V. Sheffield, "A Discrete Coil Torus with Lower Field Ripple", Princeton Plasma Physics Lab. Report MATT-1166 (1975).
10. J. A. Krommes, P. H. Rutherford, Nucl. Fusion 14 (1974) 695.
11. J. Tachon, et al., in Plasma Physics and Controlled Nuclear Fusion Research (Proc. 5th Int. Conf., Tokyo, 1974) 1, IAEA, Vienna (1975), 135.
12. J. P. Girard, D. Marty, P. Moriette, in Plasma Physics and Controlled Nuclear Fusion Research (Proc. 5th Int. Conf. Tokyo, 1974) 1, IAEA, Vienna (1975), 681.
13. F. W. Perkins, "The ICRH Tokamak Fusion Test Reactor," Princeton Plasma Physics Lab. Report TM-284 (1973).
14. J. M. Dawson, A. T. Lin, "Some New Ideas on Wet Wood Burners," UCLA Report PPG-218 (1975)

15. D. F. Duchs, H. P. Furth, P. H. Rutherford, in Controlled Fusion and Plasma Physics (Proc. 6th Europ. Conf. Moscow, 1973) 1, 29; Nucl. Fus. 12 (1972) 341; in Plasma Physics and Controlled Nuclear Fusion Research (Proc. 4th Int. Conf. Madison, 1971) 1, IAEA, Vienna (1971), 369.
16. P. H. Rutherford, et al, to be published.
17. H. L. Berk, et al, in Plasma Physics and Controlled Nuclear Fusion Research (Proc. 5th Int. Conf., Tokyo, 1974) 3, IAEA, Vienna (1975), 569.
18. J. Killeen, K. D. Marx, A. A. Mirin, M. E. Rensink, in Controlled Fusion and Plasma Physics (Proc. 7th Europ. Conf., Lausanne, 1975) 1, 22; also in Methods in Computational Physics, Vol. 16 (1975).
19. A. L. Merts, N. H. Magee, Jr., J. B. Mann, W. F. Huebner, R. D. Cowan, D. W. Barfield, M. F. Argo, to be published.
20. L. A. Berry, et al, in Plasma Physics and Controlled Nuclear Fusion Research (Proc. 5th Int. Conf., Tokyo, 1974) 1, IAEA, Vienna (1975), 101.
21. D. L. Jassby, R. M. Kulsrud, Y. C. Sun, in Controlled Fusion and Plasma Physics (Proc. 7th Europ. Conf., Lausanne, 1975) 1, 25; J. G. Cordey, W. G. F. Core, Nucl. Fus. 15 (1975) 710; D. L. Jassby, "Reactor Aspects of Counterstreaming-Ion Tokamak Plasmas," Princeton Plasma Physics Lab. Report MATT-1145 (1975).
22. R. G. Mills, ed., "A Fusion Power Plant," Princeton Plasma Physics Lab. Report MATT-1050 (1974).
23. B. Badger, et al, "A Wisconsin Toroidal Fusion Reactor Design, UWMAK-1," Wisconsin Nucl. Eng. Dept. Report UWFD-68 (1974).
24. R. W. Conn, D. L. Jassby, "A Tokamak Engineering Test Reactor," Princeton Plasma Physics Lab. Report MATT-1155 (1975), to be published in Proc. Int. Conf. on Radiation Test Facilities for the CTR Surface and Materials Program (Argonne, 1975).
25. D. L. Jassby, "Beam-Driven Tokamak Fusion-Fission Hybrid Reactors," Princeton Plasma Physics Lab. Report MATT-1115 (1975).
26. D. L. Jassby, "Alpha-Particle Heating of Tokamak Plasmas," Princeton Plasma Physics Lab. Report MATT-1074 (1974).

27. J. G. Cordey, W. G. F. Core, J. Sheffield, Nucl. Fus. 15 (1975) 755; E. A. Frieman, F. H. Tenney, Bull. Am. Phys. Soc. 19, (1974) 878.
28. D. L. Jassby, Nucl. Fus. 15 (1975) 453.
29. H. P. Furth, D. L. Jassby, Phys. Rev. Letters 32 (1974) 1176.
30. D. L. Jassby, H. P. Furth, "Two-Energy-Component Fusion Reactors with Sustained Suprathermal-Ion Energy," Princeton Plasma Physics Lab. Report MATT-1048 (1974).
31. G. H. Miley, Proc. ANS Topical Meeting on the Technology of Controlled Nuclear Fusion (San Diego, 1974) 1, 448; J. P. Boris, et al, Naval Research Lab. Report 2878 (1974). G. H. Miley, D. Driemeyer, G. Cordin, "Plasma Heating and Fusion Burns using Compression Clamping of Superthermal Ions," to be published in Proc. 6th Symposium on Engineering Problems of Fusion Research (San Diego, 1975).
32. J. G. Cordey, M. J. Houghton, Nucl. Fus. 13 (1975) 215.
33. H. L. Berk, W. Horton, Jr., M. N. Rosenbluth, P. H. Rutherford, Nucl. Fus. 15 (1975) 819.
34. M. N. Rosenbluth, P. H. Rutherford, Phys. Rev. Letters 34 (1975) 1428.
35. F. W. Perkins, "Ion-Streaming Instabilities: Electromagnetic and Electrostatic," Princeton Plasma Physics Lab. Report MATT-1141 (1975).
36. L. P. Mai, W. Horton, Jr., Phys. Fluids 18 (1975) 356.
37. R. J. Goldston, Nucl. Fus. 15 (1975) 651.
38. L. A. Berry, et al, in Plasma Physics and Controlled Nuclear Fusion Research (Proc. 5th Int. Conf., Tokyo, 1974) 1, IAEA, Vienna (1975) 113. J. G. Cordey, et al, Nucl. Fus. 15 (1975) 441.
39. B. R. Leonard, Jr., Nucl. Technology 20 (1973) 161; L. M. Lidsky, Nucl. Fus. 15 (1975) 151.
40. Proc. DCTR Fusion-Fission Energy Systems Review Meeting (Germantown, MD, 1974), U.S.E.R.D.A. Report ERDA-4 (1975) S. L. Bogart, ed.
41. R. P. Rose, et al, "Fusion-Driven Actinide Burner Design Study," Westinghouse Electric Corp. Reports WFPS-TME-003, -024 (1975).

42. H. H. Towner, D. L. Jassby, Trans. Am. Nuclear Society 22 (1975) 74.
43. D. L. Jassby, H. P. Furth, "Fusion-Reaction-Injected and Neutral-Beam-Injected Suprathermal Ions in the Tokamak," Princeton Plasma Physics Lab. Report TM-271 (1974).
44. L. M. Hively, G. H. Miley, "Comments about Proposed D-He³ Experiments in the TFTR," University of Illinois, Fusion Studies Center Report C00-2218-41 (1975).
45. B. Ahlborn, J. D. Strachan, Can. J. Physics 51 (1973) 1416.

4.0 CONCEPTUAL DESIGN STATUS

The overall evolution of the conceptual design has been described briefly in Section 2.1. This section presents a summary of the essential features of the reference conceptual design configuration. The development of design criteria, analytical considerations and system engineering studies is presented first, followed by a system-by-system review of the reference design, including a discussion of design consideration, design description, and the status of design alternatives.

4.1 SYSTEM ENGINEERING AND DESIGN CONSIDERATIONS

The material presented in Section 2.1 "Experimental Background" and 3.0 "Plasma Experiments" provides general guidelines for the design. Establishment of the detailed physical parameters of TFTR, consistent with the required plasma parameters, was accomplished through a series of design studies, tradeoffs, optimizations and compromises. At each stage a reference or baseline design was used to test the feasibility for the incorporation of new approaches and ideas. The following material presents the principal conclusions of the systems engineering studies leading to the definition of TFTR parameters.

4.1.1 REFERENCE DESIGN OVERVIEW

The TFTR design is based on plasma physics parameters that represent an extrapolation of experimental evidence obtained to date plus experimental plasma physics results from contemporary devices, such as the Princeton Large Torus (PLT).

Table 4.1.1-1 summarizes the design and operational parameters developed for the conceptual design. These parameters are to be met by the design of the following major systems: Tokamak System; Electrical Power Systems; Monitoring and Control Systems; Experimental Area Systems; Architectural and Facility Systems; Neutral Beam Systems; Diagnostic Systems; and Experimental Data Systems.

The Tokamak System provides the equipment to produce and confine a tokamak plasma of reactor grade. The equipment consists of: a stainless steel toroidal vacuum chamber constructed of bellows and ring-stiffened shells; nickel-alloy-steel-encased water-cooled copper toroidal-field coils, which provide the principal component of the magnetic field that confines the plasma; a composite system of equilibrium and ohmic-heating field windings, which position and stabilize the plasma within the vacuum vessel and induce a toroidal current in the plasma; lead-loaded borated mortar blocks to attenuate the gamma and neutron radiation during tritium operations; a structural support system

TABLE 4.1.1-1 TFTR SYSTEM DESIGN SUMMARY

<u>OPERATION</u>	
Expected Operation Span with Tritium	4 Years
Total Number of Pulsing Cycles	1×10^5
Pulse Repetition Period	300 Seconds
Full Power D-T Pulses — Total, Per Year, and Any Day	4000; 1000, 96
Neutron Diagnostics	Foil Activation, Proton Recoil Telescope
<u>NEUTRAL BEAM PARAMETERS</u>	
Total Power for Acceleration	110 MW
Total Power in Accelerated Beam	80 MW
Total D ⁰ (120 keV) Power to Plasma	20 MV
High Energy	120 KeV
Pulse Length	500 ms
<u>TOKAMAK SYSTEM</u>	
Vacuum Vessel Inner Minor Radius	1.1 m
Vessel Major Radius	2.65 m
Vessel Material	AISI 303 33
Vessel Construction	Bellows, Ring-Stiffened Shells
Vessel Surface Protection	Plates Over Bellows, Beam Catchers
Divertor	None
Plasma Cross-Section	Circular
Toroidal Field Coil Number, Type	20, Water-Cooled Copper
TF Coil Bore, Major Radius, Coil Current	2.8 m, 2.8 m, 73.2 kA
TF at Copper Inner Bore, Vessel Major Radius	9.54 T, 4.86 T
Flat-Top Time at Maximum TF	One Second
Ohmic Heating Coil Type	Water-Cooled Copper, Air Core with Iron Core Option
OH System Ampere-Turns Swing Capacity	16 MA Turns
OH Coil Primary Current (1 MA, 2.5 MA Plasma)	25 kA, 40 kA
Plasma Current Capability	2.5 MA
Equilibrium Field Coil System	Water-Cooled Copper
EF Coil Current (1 MA, 2.5 MA Plasma)	50 kA, 80 kA
Torus Vacuum Pumping	Four 24-inch Hg Diffusion Pumps
Test Cell Shielding	8-ft. Walls, 4-ft. Ceiling, Ordinary Concrete
Peripheral Device Shielding	Borated Metallic Mortar (Pb Loaded)
<u>TRITIUM GAS SYSTEM</u>	
Tritium Charge Range Per Pulse	50-150Ci
Tritium Inventory	24 kCi On-Line, 50 kCi On-Site
Tritium Supply	Solid Tritide (UT ₂) Gas Generator
Tritium Removal	Zr-Al Getters in Ballast System
Tritium Cleanup	Catalytic Oxidation, Desiccation
<u>POWER SYSTEM</u>	
Main Power Supply	2 AC Motor Generator Flywheel Systems
Energy and Power Rating	4500 MJ, 660 MW
Neutral Beam, TF, EF Power Conversion	Transformer-Rectifier
Ion Source Control	Triode, High Voltage Switch
OH Power Conversion	Transformer-Rectifier, DC Vacuum Switch
TF, EF and OH Control	SCR Voltage Controlled

consisting of two rectangular platforms which surround a central column and are supported on rollers, so that they can be moved away from each other and from the central column.

The Electric Power Systems provide the energy supply, including the large pulsed currents. Power at 138 kV is taken from the local utility and transformed by a power transformer and its associated power circuit breaker to 13.8 kV and 4.16 kV energy distribution systems. The 13.8 kV system provides power for pulsed loads such as two 16,000 HP wound-rotor motors driving two generator-flywheel units that supply power in response to the pulsed load requirements of the toroidal field, ohmic heating field, equilibrium field and neutral beam power supplies. The 4.16 kV system provides power for non-pulsed loads, including lighting, air-conditioning, water pumping and cooling, computers, breakdown oscillator and other auxiliary equipment.

The Monitoring and Control System provides the radiation and tritium monitoring and the central control and data handling operations. There will be an emergency power system which supplies those control systems necessary to shut down the facility safely.

The Experimental Area Systems support the operation of the Tokamak System: permitting machine repairs, changes, operations and maintenance before and after the machine has become activated; storing the on-site inventory of tritium and delivering it to the torus on demand; providing a means of decontamination and detritiation in certain areas; storing on-site inventories of high-purity stable isotope gases; providing cooling water for the machine field coils and pulsed power conversion systems; and providing the low base pressure required by the torus prior to beginning a pulse and removing the gases remaining after each pulse. This support is provided by individually designed systems which interface with and are integrated to the Tokamak System.

The Architectural and Facility Systems provide a self-sufficient experimental complex that includes space for the TFTR building Complex, Electrical Buildings, Field Coil Power Conversion Building, MG Building, Neutral Beam Power Conversion Building, Control Room, Transformer Yard and Facility Service Systems. The buildings house the tokamak device and the supporting equipment and systems required for the operation of the planned experiments. The Facility Service Systems provide the final heat sink for all cooling systems of the facility; the Fire and Explosion Protection System; the Radioactive Waste Handling System; the Ventilation System; and the Liquid Nitrogen and Gas Storage System.

The Neutral Beam Systems provide high current, low divergence beams of high energy neutral deuterium or hydrogen atoms for raising and maintaining plasma temperature at specified time intervals, and in the case of deuterium, trigger the fusion power production. The systems consist of neutral beam injectors plus an energy conversion system which converts ac current to high voltage dc, conditions it, and supplies it to the ion sources.

The Diagnostic Systems provide apparatus for the acquisition of information about the plasma. With this information, experiments can be evaluated and device operations can be controlled.

The Experimental Data Acquisition Systems provide computers, electronic data processing, data display and recording for the data obtained during the experimental programs.

The evolution of the TFTR Systems since the June 1974 design is traced in generic terms and listed in Table 4.1.1-2.

Some basic design choices have followed directly from the major objectives of the TFTR project.

Type of Toroidal Field Coils

Of the various design options evaluated during the course of the study, the choice between water-cooled copper coils and superconducting coils was by far the most important. An extensive evaluation was performed. It was recognized at the outset that the decision as to which of these two coil types to incorporate will have a far reaching impact on the configuration of the device, power supply requirements, schedule and risk associated with development of components, as well as on the amount of information that the TFTR will contribute to the design, construction, operation and maintenance of the EPR-1.

Turning first to considerations having to do with coil construction and device size, it appeared that the relatively limited current-state-of-the-art superconducting materials and technology would have to be employed if the construction of TF coils was to be completed on a timely basis. Thus, material considerations limited the toroidal field strength to a maximum of 7.5 tesla at the TF coil innerbore for the superconducting option vs. 10 tesla for the water-cooled copper option. The reduced field in the superconducting option must be compensated by increased size to reach the same plasma current. Furthermore, substantial space for shielding to limit the neutron-induced heat generation in the superconductor must be provided to prevent the coils from going normal. These factors, in addition to other design considerations, dictated a substantially larger device for the superconducting design version than for the version with water-cooled copper coils.

It was recognized that superconducting magnet technology will undoubtedly have to be developed for future experimental power reactors and will represent a vital part of the technology required for a demonstration fusion power plant. Therefore, the design of the superconducting option for TFTR was carried to a level of detail sufficient to provide a valid basis for comparison with the water-cooled-copper option on key items such as cost, schedule, risks and uncertainties associated with the development of new technology, and benefits to the CTR program.

TABLE 4.1.1-2 SUMMARY OF TFTR SYSTEM DESIGN VARIATIONS

System	JCDS July 1974	Addendum II, February 1975	Final Conceptual Design Report - October 1975
1.0 Tokamak System	TF Coil Stiffened by Inconel I-Section SS 305 Torque Frame Sacrificial Liner	Reduced Size TF Coil Stiffened by Nitronic I Section EF Moved Outside TF Coils Composite EF, OH Coil System Nitronic Torque Frame Bellows Cover Plates Refractory Metal Protection Plates	Torque Frame removed and replaced by V-shaped TF Coils having "Nitronic 33" Structures. Capability to insert iron core incorporated. One side of device fixed. Neutral beams rearranged to provide minimum bus runs to power supplies.
2.0 Experimental Area Systems			
2.1 Torus Vacuum Pumping System	No Associated Tritium Cleanup Pkg. 8 41-cm Hg D. P.	Includes Tritium Cleanup Package on Pump Exhaust Lines	Cleanup Package moved to Tritium Cleanup System. 4 61-cm Hg D. P.
2.2 Neutral Beam Injection System	Combination of LBL/LLL (500 A - 80 keV) Sources & ORNL (72 A - 150 keV) Sources	LBL/LLL - "50 A" Sources (720A - A11 En, 83 keV Max; 150 A - 125 keV), or ORNL - "1MW" Sources (75 A - 154 keV)	LBL, LLL - "50 A" Sources or ORNL 1 MW Sources (120 keV - 167 amps) 500 ms - both LBL/LLL and ORNL. Total Acceleration Power 80 MW
2.3 Tritium System	Monoenergetic Neutral Beam Eight Beam Lines 4100 Ci On-Line No Fuel Valt Single TFTR Cell Detritiation System	Direct Source - Neutralizer Coupling Six Beam Lines 24 kCi On-Line, 50 kCi On-Site One Generator Separate Fuel Fault Vault and Cell Detritiation Systems	Four Beam Lines 3 sources each with provisions for 2 additional lines. 25 kCi On-Line, 50 kCi On-Site NMC
2.4 Non-Tritium Storage and Delivery System	High Pressure Gas Storage	NMC	NMC
2.5 Machine Area Water Cooling System	3160 tons (11.1 MW) 300 second repetition rate	2460 tons (8.7 MW) 350 second repetition rate 7200 gpm	3300 tons (11.6 MW) 300 second repetition rate 3600 gpm Supply cooling for SCRs as well as field coils
3.0 Energy Supply and Distribution Systems			
3.1 Transformer and Switchyard	30 MVA	50 MVA	50 MVA
3.2 Motor-Generator-Flywheel System	2 shafts - 4 generators - 2 flywheels 630 MW, 4300 MJ	3 shafts - 3 generators - 2 flywheels 660 MW, 4000 MJ	2 shafts - 2 generators - 2 flywheels Provision for 3rd machine 660 MW, 4500 MJ
3.3 Standby Power System	150 kW battery-inverter and a 1500 kW diesel-electric generator	NMC	NMC
3.4 Auxiliary Power Supply and Distribution	No Transformer	15 MVA Transformer	3rd winding on switchyard transformer

TABLE 4.1.1-2 SUMMARY OF TFTR SYSTEM DESIGN VARIATIONS (Continued)

System	JCES July 1974	Addendum II, February 1975	Final Conceptual Design Report - October 1975
4.0 <u>Pulsed Energy Conversion Systems</u>			
4.1 TF Power Conversion	516 MW	440 MW	440 MW
4.2 OH Power Conversion	85 MW	60 MW	60 MW
4.3 EF Power Conversion	35 MW	240 MW	240 MW
4.4 Neutral Beam Power Conversion	Energy Conversion System for a Combination of ORNL & LBL/LLL Ion Sources 2 - 108 MW Systems	Capacity to handle Berkeley, 121 MW Systems (3 ion sources/power supply) or ORNL, 91 MW system (4 ion sources/power supply)	Capacity to handle either Berkeley or ORNL, 12 installed ion sources, 120 kV at 55 A to each grid No. 1 (80 MW total)
5.0 <u>Inst. & Control Systems</u>	High Speed Data Acquisition, Conventional, hard-wired safety and protection, digital process control.	NMC	All operations controlled from central control station with local control available for motor generator and remote handling.
6.0 <u>Civil, Arch. & Structural</u>			
6.1 Facility Site	General Site Definition, 250 m diameter controlled access area	James Forrestal Campus, Princeton, New Jersey	NMC
6.2 Facility Structures	Test Cell Size 100' x 110' Office Bldg. Cap. = 30 people One Electric Equipment Bldg. 2 Flywheel Pits	Test Cell 110' x 120' Office Bldg. Cap. = 20 people MG Bldg. (200' x 110') Power Equipment Bldg. (175' x 80') 2 Flywheel Pits	Test Cell 110' x 120' x 45' high MG Bldg. 110' x 200' x 45' high (3 flywheel pits). Field Coil Power Conversion Bldg. 100' x 165' x 25' high. Neutral Beam Power Conversion Bldg. 65' x 159' x 28' high. Control Room and Offices 85' x 115' x 16' high. Mechanical Equipment Room 50' x 105' x 15' high. Two Sides of Test Cell kept available for future expansion and RF heating.
6.3 Facility Services	One Detritiation System Conventional Remote Handling Manipulators and Equipment with Special Tooling	Four Detritiation Systems Addition of Portable Shielded Cell	Five Detritiation Systems Shielded Cab for Test Cell Remote Maintenance

* NMC - No Major Changes

It was concluded that the superconducting coil version would imply several years' delay in completion and would impose substantially higher technological risks. The results of this evaluation were described in the TCT Joint Conceptual Design Study (JCDS) completed in June, 1974. Balancing the immediate practical advantages of the copper-coil version against the reactor — technology contribution of the superconducting-coil version, a clear decision was made in favor of the copper-coil version.

Shape of Toroidal Field Coils

For tokamak devices that aim at maximizing the plasma-current magnitude and duration, D-shaped coils have several potential advantages. The D-shape serves to reduce the stresses in the coil structure by virtue of minimizing bending moments, thereby simplifying the coil design and allowing the fraction of copper to be increased relative to the fraction of steel reinforcement. This feature alleviates the problem of heating the TF coils during long pulses. In addition, the D-shaped coil would allow more room for vertically elongated plasma cross-sections. This feature lends itself to the passage of larger plasma currents. For a tokamak device that is designed primarily to achieve high plasma temperatures for moderate durations, however, the D-shape has the disadvantage of introducing some extra costs and complications.

Preliminary calculations for the TFTR application indicate that going to a D-shaped coil that eliminates the coil-bending moments would require about 1,000 MJ (20%) additional flywheel stored energy, about 61,000 kg (25%) additional copper, and about 30% more structural material to restrain the lateral moment. The ratings of the OH and EF supplies would also have to be increased about (25%) due to the increase in vertical height of the coil systems. An elongated plasma would be vertically unstable, thus complicating the position control problem.

Balancing the extra cost and technological complication, and the introduction of additional plasma-related uncertainties, against the advantage of increasing the TFTR capabilities in the eventuality of highly favorable confinement, the decision was made in favor of a standard round-cross-section plasma, vacuum chamber, and toroidal-field coil system.

Use of an Iron Transformer Core

The ohmic heating system represents one of the major cost components of a tokamak facility. If the OH system is of the air-core-transformer type, the magnetic energy stored in the primary winding and the power dissipated in maintaining the current, are relatively high. Use of an iron transformer yoke reduces the required primary current, and therefore the stored magnetic energy and power dissipation. In a tokamak device designed to maximize the plasma current strength and duration, the savings in the OH electrical system attendant on the use of an iron yoke are substantial. In addition, the reduction of the power dissipated in the copper primary windings alleviates the attendant problem of temperature rise during long, high-current pulses.

The main disadvantage of the iron transformer yoke is that its constricted central portion necessarily becomes saturated, even if the device design is specially adapted by reducing the toroidal field strength, and raising the aspect ratio R/a of the plasma beyond their usual optimal values. The saturation of the yoke not only limits the associated cost-saving, but also introduces serious new complications in respect to the interaction of the applied poloidal field with the tokamak plasma. In relatively low-current, short-pulse operations, such as the reference modes of the TFTR, the economic advantage of the iron transformer core is not compelling, and its interference with plans for the accurate control and symmetrization of the poloidal fields represents an important defect.

Balancing the potential advantages of an iron core for operation in optimistic plasma regimes against the added uncertainties that would be introduced into the reference operating modes of the TFTR, it was decided to provide for an optional cylindrical iron core that would be absent during initial TFTR operation, but could be inserted subsequently to increase the capability for taking advantage of favorable plasma confinement in high-current, long-pulse regimes. Without the iron core, the plasma current requires 300-700 msec to reach a maximum level of 2.5 MA, and can be maintained at this level for times of order 1 s., depending on plasma parameters; the total volt-seconds consumption is 15 V/s (cf. Sections 3.1.1, 4.1.2, and 4.1.5). Insertion of the iron core produces an additional 2 V/s; the current rises more rapidly to 2.5 MA and can be maintained for several seconds longer with the given OH electrical system.

A number of other less fundamental design choices are under continuous review. Five examples are given below:

- TF Coil and Torque Frame

The "I-beam" stiffened TF coil design and the associated torque frame of the initial TFTR design seemed to require an unreasonable degree of sophistication in remote assembly/disassembly operations. That design would, for example, require removal of the large torque-frame members to achieve access to TF coils and other components.

An alternate V-shaped box-beam design, now incorporated in the reference conceptual design, seems to offer significant advantages. The assembly of the TF coils into their cases is simplified, the routing of water and electrical leads is made easier and, most importantly, the need for the torque frame is eliminated. This allows much freer access to the machine components and, as a result, will reduce significantly the complexity of some remote handling operations.

Although the "box-beam" coil is more expensive, elimination of the torque frame results in an overall direct cost reduction, as well as substantial reductions in maintenance costs.

- Combination OH/EF Coils

Because of the close proximity of some OH conductors to EF conductors, combining the poloidal field windings in an effort to conserve space and to simplify the design was studied, with the result that this feature has been incorporated into the design with some cost savings. The effect on the cost of the power supplies is minimal. A useful feature of the conceptual EF/OH design is the ability to select one of three EF field curvatures by winding taps. Efforts are continuing to eliminate the remaining breakable electrical joints in the conductors; but, so far, the associated increases in cost of the power supplies seem prohibitive.

- Energy System

All elements of the energy system have been reviewed in order to optimize the supply/load costs. The number of generator shafts selected (2) appears to offer the best overall compromise between the obvious advantages of minimum number of units, each having individual auxiliary devices, the flexibility and high facility utilization achieved with more units, and total system cost. Provisions have been made in the MG building design to accommodate an additional vertical shaft machine, either as a spare to allow operational flexibility during maintenance periods or to provide additional experimental capability.

Alternatives to the presently planned neutral beam power supply have been considered, including direct utility line pulsing. These ideas will continue to be reviewed.

- Reactor Building and Shielding

Alternate Test Cell configurations were studied in an attempt to reduce the total facility cost. Ideas which were considered include: a rearrangement of the beam lines so that all enter from the same side of the device, providing a shield system that would be capable of being rearranged, and reducing the volume of the shielded region (reducing the height). These considerations were strongly dependent on the conceptual design of neutral-beam systems. In particular, the ability to rearrange the shielding was considered important in view of present uncertainties in the beam line design.

The rearrangement of the beam lines has been implemented and the overall height of the Test Cell reduced. The latter change will continue to be reviewed carefully, since it has reduced remote-handling flexibility to some extent.

- Neutral Beams

A decision was made at the neutral beam workshop held at LBL/LLL in July 1975 that the development programs at ORNL and LBL/LLL should be directed toward a single mode of operation for the TFTR injection system, viz, a total of 20 MW of 120 keV neutral beams per injection with a pulse length of 0.5 s and a total power in the accelerated ion beams of 80 MW. Four beam lines, each containing three ion sources, are to be utilized. Although the goals of the neutral-beam development program have been simplified, there has been no compromise in the overall plasma experiment objectives.

Consistent with recent beam-deposition computer studies the neutral beams are aimed tangentially to circles having major radii of 270 ± 20 cm for precompression injection, and 220 ± 20 cm for postcompression injection. These tangent radii are for the center of the central source in each line. Flanking sources are to be aimed within ± 20 cm. The reduction in maximum aiming radius (290 cm to 270 cm) has allowed the width of the aperture at the torus to be increased to 40 cm. The height of the aperture is 70 cm.

The beam lines will be designed to have a maximum length from the torus to the far end of the power-connection plug of 9 m, and will have a maximum height of 6.5 m. All of the arms will be located on one half of the torus, to minimize the Test Cell size, and to limit the length of the cables from the ion sources to the power conversion room.

The ion sources will be designed so that regular cleanup pulses between experimental pulse will not be required for producing a pure beam. This alleviates the gas-handling problem.

A single neutral beam power conversion system will be designed by PPPL, consistent with the operation of either ORNL or LBL/LLL ion sources.

The status of specific design alternatives is discussed in greater detail in Section 4.2 through 4.7 below, in the context of the descriptions of specific systems.

4.1.2 REFERENCE EXPERIMENTAL SCENARIOS

Some typical plasma experiments are discussed in Section 3.0 in terms of the desired evolution of the plasma parameters. Each of these experiments, as well as the many others that are possible, calls for a different setting of the magnitudes and time variations of the various fields and other external parameters of the machine. The sequencing of the machine parameters is computer-controlled by the Central Instrumentation and Control System, which also provides safety and protection, as well as data handling and display. The library for this computer will undoubtedly be modified and expanded as more and more experience is gained on TFTR. Three examples of initial programs

for this library are outlined in the following, which is illustrative of the detailed analyses which have been performed to see how the plasma physics objectives of TFTR might be implemented experimentally, and to establish functional and design requirements for subsystems and components.

Weak Compression (Mode A) Scenario

Consider the experimental procedure that is necessary to carry out a Weak Compression Experiment (Mode A of Figure 2.1-2). A toroidal plasma column having a minor radius of 60 cm is formed in the outer region of the vacuum chamber, at $R = 310$ cm. It is heated by high-energy (120 keV) neutral beam injection for 0.1 s, during (or following) the rise of the plasma current to 0.8 MA. The plasma is then moved inward to $R = 248$ cm, causing it to be compressed in major radius by a factor $C = 1.25$. The plasma current rises to 1 MA in this process. Immediately following compression, the plasma minor radius is 54 cm. The region between the hot plasma core and the smallest limiter-intercepting magnetic surface, at a minor radius of 85 cm, is then filled with cold plasma. The outer limiter is positioned at a major radius of $R = 370$ cm, in order to be located 3 cm inside the vacuum-vessel-protection plates. The inner limiter is located at a major radius of 163 cm.

In addition to the weak compression scenario ($C = 1.25$), similar experiments could be carried out for various compression ratios, ranging up to $C = 1.5$ for the strong compression case, (Mode B of Figure 2.1-2). Some plasma parameters for the compressional experiments, and also for the high current experiment to be described later, are given in Table 4.1.2-1.

TABLE 4.1.2-1 PLASMA PARAMETERS

	I(MA)	R(cm)	a(cm)	B_T (T)	B_p (T)	C	L(μ H)	V(cm ³)
Weak Compression	0.8	310	60	4.16	0.27		7.7	2.2×10^7
1 MA Plasma	1.0	248	54	5.19	0.37	1.25	5.9	1.4×10^7
Strong Compression	0.67	310	55	4.16	0.24		8.1	1.9×10^7
1 MA Plasma	1.0	209	45	6.16	0.44	1.5	5.8	8.4×10^6
High Current	1.8	280	90	4.6	0.40		5.3	4.5×10^7
2.5 MA Plasma	2.5	248	85	5.19	0.59	1.1	4.5	3.5×10^7

I	Plasma Current	B_t	Toroidal Field Strength
R	Major Radius	B_p	Poloidal Field Strength
a	Minor Radius	L	Plasma Inductance (uniform current density)
C	Compression Ratio	V	Plasma Volume

The programming of the main machine parameters for the weak compression experiment is illustrated in Figure 4.1.2-1. Several distinct phases in this sequence can be singled out for elaboration. It is assumed that all systems (Electrical Power Supplies and Conversion Systems, NB Injection System, Vacuum System, Gas Handling Systems, etc.) have been checked out and verified to be ready. The reference time, T_R , is the start of current flow in the plasma.

a. Buildup of Field Currents

The Toroidal Field Coil System has the longest time for current buildup. In the present case, the peak field value is required, and the TF current is initiated at approximately $T_R - 12$ s. As shown in Figure 4.1.2-1, the TF current reaches its peak of 73.2 kA at T_R , providing a magnetic field of 4.16 T at $R = 310$ cm.

The current for the ohmic heating field coil is initiated at $T_R - 3$ s, so that a current of 20 kA is available at time T_R . The buildup time for the current in the OH field coil is chosen long enough so that the time rate of change of current is less than 30 kA/s, or the voltage across the coil is less than 800V. A loop voltage of less than 4V is generated inside the vacuum vessel, thus preventing a premature discharge.

Figure 4.1.2-1. Weak Compression (Mode A)

b. Plasma Preionization

At $T_R - 1s$ the main valves of the mercury diffusion pumps have arrived at a fully throttled position, and a measured quantity of tritium gas is injected into the vacuum chamber. At $T_R - 10ms$ the breakdown oscillator (100 kW, 100 kHz) is turned on, supplying power to two sets of internal B-oscillator loops. These loops provide a high RF field in the outer region of the vacuum vessel and cause ionization of the tritium gas in a region located about 20 cm in from the outside limiter. After several milliseconds the electron density is of the order of 10^{11} cm^{-3} .

The localization of the breakdown of the gas can be enhanced by the creation of a hexapole null in the poloidal magnetic field (i.e., the field normal to the toroidal magnetic field). This null, located 20 cm in from the outside limiter, would provide a preferred location for the pre-ionization and discharge initiation. Studies are being made to determine if the fringing field from the OH winding can be adjusted to provide this hexapole null.

c. Discharge Initiation and Plasma Current Buildup

As soon as pre-ionization is verified, a command signal activates the OH Energy Conversion System. This system has its own sub-program which provides a prestrike plasma loop voltage to initiate the discharge. As shown in Figure 4.1.2-1 plasma current flow is detected at T_R (by definition of the reference time), when the plasma voltage is $\approx 150V$. This voltage is maintained for some chosen time ($\sim 5 \text{ ms}$) then reduced abruptly to about 75 V, followed by a slower decrease for the next $\sim 40 \text{ ms}$, which produces the plasma current rise shown.

The equilibrium field current is started at T_R and is programmed to rise with the plasma current. The rising EF gradually moves the center of the plasma column from its original position toward the desired position having a major radius of 310 cm. (If a hexapole null has been provided, it will be quickly overpowered by the EF and the poloidal magnetic field of the plasma current.) During this phase, increments in the plasma current tend to form on the outer surface, but as the plasma is compressed to a smaller major radius, this current moves to the interior of the plasma, while additional current is generated on the surface, by injection and ionization of neutral gas. In this manner, a plasma can be formed that has a prescribed profile of current density. By $T_R + 60 \text{ ms}$, the plasma is approaching its final position, carrying a current of $\sim 0.5 \text{ MA}$. The vertical position of the plasma column is maintained by the curvature in the EF flux lines, which have a pre-set magnetic index. Additional vertical control may be attained by a feedback system, still to be designed, that will be based on a small radial magnetic field produced by the addition of a few turns, or by some of the available EF/OH turns. The horizontal position of the column is maintained by feedback control of the EF power supply.

d. Neutral Beam Injection

The plasma may reach a temperature of the order of 1 keV due to the ohmic heating, but auxiliary heating is now required to drive the temperature higher. At $T_R + 15$ ms the fast valves in front of the NB injectors start to open. Opening at this time minimizes the amount of T2 that can diffuse into the beamlines, since the plasma column functions as a very good pump. At $T_R + 60$ ms, the position and density of the plasma is verified, the valves are wide open, and the NB injectors are turned on at high power. Injection of 30 - 35 MW of D^0 , including 20 MW of 120 keV D^0 , continues for the next 0.1 sec., after which time the fast valves are closed.

The plasma current reaches a full precompression level of 0.8 MA during the injection. The EF current rises in response to the increasing plasma pressure and current. Figure 4.1.3 1 shows the amount of this increase needed for the present example. Since the EF tends to drive the plasma current higher, the current in the OH Coil required to maintain the flux encircled by the plasma loop approximately constant is less than would have been necessary to drive the plasma to 0.8 MA without injection.

e. Plasma Compression and Reaction Time

Immediately upon termination of the NB injection, at $T_R + 160$ ms, the precharged capacitor bank in the EF power conversion system is fired, causing a fast ramp-up of the EF. The plasma column responds by moving inward to a major radius of 248 cm during a ~ 25 ms time interval, and the plasma current rises to 1.0 MA. The compressed plasma configuration can be maintained for times up to $T_R + 1$ s before the TF starts to decrease.

In order to prevent skin currents from forming on the highly conducting plasma during compression, the OH circuit must be used to make small adjustments in the total plasma current, so that it will follow the scaling $I \rightarrow R^{-1}$ of the plasma volume current. The power supplied to the OH during this phase is used typically to make the OH current more negative. If β_p increases appreciably during compression, however, the required OH current compensation is positive.

f. Discharge Termination

Discharge termination is started at (or before) $T_R + 1$ s, by decompressing the plasma to its starting position. The OH coil current is again swung abruptly toward zero, creating a large reverse loop voltage on the plasma, which drives the plasma current toward zero. The magnetic energy of the EF/OH/plasma system is removed at the terminals of the OH and EF coils, to avoid having it deposited in the vacuum vessel. A heavy gas may be injected to help radiate away the plasma heat energy and deposit it uniformly over the vacuum vessel wall. The EF current is driven down in synchronism

with the plasma current, so that the plasma column is displaced continuously outward against the outside limiter, reversing the initial plasma formation process.

High Current (Mode D) Scenario

The programming of the main machine parameters of the High Current Mode D experiment of Figure 2.1-2 is shown in Figure 4.1.2-2. The maximum toroidal field is required, so that the TF current program is the same as in Figure 4.1.2-1. The OH coil current is built up to its maximum value of 40 kA. The preionization and discharge initiation phases are similar to those discussed above for Mode A. Since a plasma current of 2.5 MA is desired, a high loop voltage is needed for a longer time to drive up the plasma current as rapidly as possible. By $T_R + 140$ ms the plasma current is 1.8 MA, and the plasma column has a minor radius of 90 cm centered at the major radius of $R = 280$ cm. A major-radius compression is now performed by a rapid increase of the EF which moves the center of the plasma column to $R = 248$ cm. The compression increases the plasma current slightly, as shown in Figure 4.1.2-2. At $T_R + 150$ ms the NB injectors are turned on at full power and heating continues for the next 0.5 s. The discharge is terminated in the same fashion as outlined for Mode A. The plasma current is driven down in synchronism with the EF to remove the magnetic and heat energy from the EF/OH/plasma system.

Figure 4.1.2-2. High Current Mode D

Discharge Cleaning Scenario

An essential procedure in preparing TFTR for an experimental run will be the discharge cleaning process, to remove rapidly the contaminants that invariably build up on the inner surfaces of the vacuum vessel. For instance, during an opening of the vessel for servicing, certain vapors and gases, e.g., H_2O , CO and various hydrocarbons, will enter the system and stick to the surfaces tightly enough so that they are not readily removed when the system is re-evacuated. Even after an in situ bakeout under vacuum, the walls are still covered with CO . In fact, all the materials of the vacuum system will continue to have finite outgassing rates of contaminants, which can become an appreciable part of the surface-layer gases during a shut-off period. Under discharge conditions, these surface contaminants will enter the plasma, leading at best to a series of non-reproducible discharge pulses.

At the normal repetition rate of one discharge every 300 seconds, the removal of these contaminants would require days, if not weeks, of continuous pulsing. Discharge cleaning is simply a rapid pulsing of the machine, so that the removal process is greatly speeded up. The net effect of discharge cleaning (which is a complex surface process, involving, among other things, ion bombardment of the walls as well as localized heating of limiters, etc.) is to remove the many layers of unwanted, loosely attached impurities and replace them with a minimum number of layers of more desirable gases, e.g., isotopes of hydrogen.

To accomplish rapid pulsing, the toroidal field will be run at a steady value of 0.6T. The working gas, a mixture of an isotope of hydrogen and a heavy gas such as argon, will flow through the system continuously at a pressure of 1.3×10^{-2} Pa (to give a peak plasma density of $\sim 10^{13} \text{ cm}^{-3}$).

The breakdown oscillator will provide a pre-ionization electron density of $\sim 10^{11} \text{ cm}^{-3}$. A special OH circuit will be used to provide plasma currents of up to 300 kA, lasting for 100 ms, at high repetition rates. This special circuit will be made up of various components of the normal OHCS and will consist of a high voltage capacitor bank (to affect discharge initiation and rapid current buildup) and a low voltage bank (to sustain the current). These banks will be resonated with the OH field coil. In this manner an energy of several hundred kJ will be deposited into the vacuum vessel in the form of radiation and bombardment by 10 - 50 eV ions. Thus, the average power heating of the vessel will be 10 to 20 times greater than the OH might deliver during normal operation at the 1 MA level. The EF will be programmed to provide a stable plasma during current buildup so that a plasma column filling the greater part of the vacuum vessel is obtained. The discharge will be terminated unstably by allowing the EF to get out of step with the decreasing plasma current.

4.1.3 NEUTRONIC AND PHOTONIC DESIGN

The conceptual design of the TFTR facility has been significantly influenced by the anticipated production of neutrons and gamma rays during the pulsing of the device with deuterium (D) or deuterium-tritium (D-T) fuel. The manner in which the design has been affected by the radiation fields is summarized in this section. Problems associated with handling the tritium are discussed in Section 4.1.8.

Generation of Radiation

Calculations on plasma heating in the production of fusion reactions have been described in Section 3.2. In the present section, the interactions of the resultant radiation with the TFTR apparatus is considered.

In the radiation analyses and shield design, it was conservatively assumed that 10 MJ of D-T fusion energy per pulse would be produced. This corresponds to the generation of 3.6×10^{18} neutrons per pulse. As described in Sections 3.2 and 4.1.2, there are many modes of operation where the injected energy and the neutron production per pulse will fall below this reference pulse by factors greater than 5. The peak fusion power is taken to be typically ~20 MW. Twenty percent or 4 MW of the fusion power is associated with alpha particles resulting from the fusion reaction, and the remaining 16 MW is from neutrons. A portion of the alpha particle power is absorbed by the plasma and the remainder by the inner surface of the vacuum vessel. Since the neutrons (16 MW) are more penetrating and can pass through the vacuum vessel, most of the power associated with the neutrons is deposited in the surrounding materials. Additional energy is released when the neutrons are absorbed, viz. the binding energy of the neutron in the residual nucleus. The nuclear heating rate in the vacuum vessel from neutron and gamma ray fluxes is relatively small, $\sim 1.5 \text{ W/cm}^3$, because of the low absorption probability. A prompt gamma ray flux will be generated by nuclear reactions as a result of the neutron flux, viz. radiative capture and inelastic scattering. In addition, activation of materials by the neutron flux during the pulses will cause a gamma ray field during periods between pulses.

The conceptual design is based on pulsing the TFTR once every five minutes. Therefore, it is expected that there will be many thousands of pulses per year. D-T operation will be limited to the equivalent of 1000 of the reference full-power pulses per year for four years; however, the TFTR objectives based on D-T operation, should be met during the allowed 4000 reference pulses or their equivalent.

The source neutron flux (~14 MeV) at the vacuum vessel is $6 \times 10^{12} \text{ n/cm}^2 \text{ -s}$ and the fluence for 4000 reference pulses is $1 \times 10^{16} \text{ n/cm}^2$. Transport calculations predict a total integrated neutron flux at the vacuum vessel of $\sim 3 \times 10^{13} \text{ n/cm}^2 \text{ s}$.

Consequences of Radiation

Consequences of the radiation levels and irradiation times are:

- Biological shielding must be provided.
- Activation of components will be significant, and remote maintenance will be required.
- Activation of the air in the device Test Cell and of the water in the cooling systems for the magnetic coils will be significant. This will affect access to the Main Test Cell and to the mechanical equipment (chiller) compartment.
- Radiation damage to structural metals and alloys will not be significant. The threshold dose rates for radiation damage will be reached for some polymeric materials, however, and the selection of materials to be used in the vicinity of the vacuum vessel must take into account the radiation levels. Additional shielding must be provided for diagnostic apparatus and solid state electronic devices located in the Main Test Cell.
- Nuclear heating will have a serious deleterious effect on the cryopanel used to provide the large pumping speeds required in the neutral beam arms, unless adequate shielding is provided.
- The emitted radiation will provide a useful diagnostic tool.

Shield Design

The radiation design philosophy adopted is that of providing a shielded Test Cell for location of the tokamak from which all personnel must be excluded during the pulsing of the device. The walls of the Main Test Cell are made of ordinary concrete and are 1.8 m (6') thick. Since the roof and nearby air space above the Test Cell will be inaccessible during operation, the ceiling of the Test Cell can consist of a 1.2 m (4') thickness of ordinary concrete. Radiation that passes through the roof of the Main Test Cell will be scattered by the air and will contribute to the dose rate at ground level outside the building. This scattered radiation (sky-shine) is the dominant radiation at the site boundary (125 m radius from the TFTR).

The calculated dose rates at the site boundary are 3×10^{-3} mrem/pulse due to sky-shine and 1×10^{-3} mrem/pulse due to the direct radiation. The integrated dose during a year is 4 mrem, which satisfies the design criterion of ≤ 5 mrem/year. The selection of this criterion indicates an intent to comply with the principle of "as low as practicable". This dose is $\sim 1/18$ of the natural background radiation. The external dose rate due to the prompt radiation during a pulse increases in moving from the site boundary to the outside wall of the Test Cell. The dose rate due to the direct radiation is more important

relative to the sky-shine in moving toward the Test Cell walls, and is the dominant radiation at the walls.

The calculated dose rate at the outer boundary of the Test Cell walls is 700 mrem/hr during a pulse, and the integrated dose per pulse is 0.10 mrem. Since there is a maximum of 12 pulses/hr, the dose is 1.2 mrem/hr. Also, based on 1000 pulses, the integrated dose is 100 mrem per year. These doses meet the criteria for an unrestricted area as defined in 10CFR20, viz., the dose shall be less than 2 mrem in any one hour and less than 100 mrem in any 7 consecutive days for an individual continuously present in the area, and less than 500 mrem to the whole body during one calendar year. Two of the outer walls of the Test Cell are 0.9 m thick, rather than 1.8 m; however, an equivalent thickness of soil, i.e., a berm, has been added to compensate. A 1 cm thickness of a commercially available boron plaster is applied to the inner surface of the concrete walls and ceiling; the plaster reduces the thermal neutron flux within the Test Cell by a factor of 10.

Radiation doses received at various locations around the TFTR site are presented in Table 4.1.3-1. In obtaining these doses, the additional shielding provided by the walls of these buildings has been ignored.

**TABLE 4.1.3-1 DOSES DELIVERED AROUND TFTR SITE
BY 20 MW, 0.5 SEC D-T PULSES**

Location	Dose per Pulse (mrem)	Annual Dose for 1000 Pulses (mrem)
Field Coil Power Conversion	0.020	20
Mock-Up Assembly Room	0.018	18
Control Room	0.15	150
Site Boundary (125 meters)	0.004	4

Inside the Test Cell, an additional shielding barrier is provided in the form of sectional shielding incorporated into the design of the device itself. An internal belt of shielding is fitted along the outer periphery of the vacuum vessel, composed of a borated lead-loaded mortar of density 4.2 gm/cm^3 . A segmented shield, consisting of a similar metallic mortar of density 2.5 gm/cm^3 , is fitted into the spaces between the TF coils in the upper part, or placed on the floor in the lower part. Finally, a capping disk, of the latter material is placed on the top of the device. The overall thickness of the device shield is in excess of 50 cm, (see Section 4.2.2 for design details). The attenuation of both the neutron and gamma ray fluxes through the bulk device shielding is in

excess of 3 decades. The prompt neutron and gamma ray fluxes throughout the Test Cell and its concrete walls are shown in Figures 4.1.3-1 and 4.1.3-2. An additional attenuation of ~ 6 decades of the neutron flux occurs through the concrete walls.

The function of the device shielding is to reduce general radiation fields outside the shielding and in particular to decrease the neutron flux so as to reduce the activation of the device components, the diagnostic equipment, the handling crane, and the surrounding air as well as to make possible personnel access to the inside of the Test Cell during the non-operating periods after a suitable cooling-off period. The dose rate in general within the Test Cell two hours after shutdown due to the activation of components and the Test Cell walls will be $\lesssim 100$ mrem/hr, and one day after shutdown $\lesssim 10$ mrem/hr, even after several years of D-T operation. Figure 4.1.3-3 shows the dose rate as a function of time following shutdown near a large stainless steel object located outside the device shielding.

However, there can be localized hot spots due to the activation of materials, arising from neutrons streaming through penetrations in the device shielding. Additional shielding is provided near the penetrations, for example around the neutral beam injector entrance ports, to reduce the streaming. Portable shadow shield panels will be provided in the Test Cell that can be moved near activated components to allow as much hands-on maintenance as possible. Portable shielding is also provided to enhance the shielding properties of the concrete wall behind the neutral beam arms. Streaming

Figure 4.1.3-1. Neutron Flux Within the Test Cell and Through the Concrete Walls in the Equatorial Plane

Figure 4.1.3-2. Gamma Ray Flux Within the Test Cell and Throughout the Concrete Walls in the Equatorial Plane.

Figure 4.1.3-3. Dose Rates from a Large 305 Stainless Steel Object Located Just Outside the Device Shield

through the large access ports for the neutral beam arms can result in local neutron fluxes at the inner surface of the Test Cell that are a factor of 5 higher than on the surrounding surface. Figure 4.1.3-4 shows the dose rate as a function of time near the stainless steel located at the back end of the neutral beam arm. It can be seen by comparing with Figure 4.1.3-3 that neutron streaming results in dose rates at this location that are about an order of magnitude higher; viz., ~50 mrem/hr due to long half-life isotopes.

The torus vacuum vessel cannot be shielded from the fusion neutrons, and the activation dose rate near it will be as large as several rem/hr due to long half-life isotopes. Figure 4.1.3-5 shows the contact dose rate due to the activation of the vacuum vessel as a function of time after shutdown for various irradiation histories. Remote maintenance is required after several hundred pulses.

The device shielding is very effective, due to the large lead content, in reducing the dose rates outside the shield arising from activation gamma rays. However, remote maintenance is necessary for the vacuum vessel and for other components in the vicinity of the vacuum vessel that are not adequately shielded. The segmented device shield is intended to be portable so that access by a remote manipulator is possible. During such disassembly operations the removal of the segments of the device shield will result in higher radiation levels in the Test Cell.

The device shield composition has been selected to minimize radioactivity. The activation is due mainly to impurities, and the dose rate at the irradiated surface of the shield is ~10 mrem/hr three days after shutdown.

An ordinary concrete shield, with a thickness of 0.76 m, is located in the support structure. The floor is 1.2 m thick and consists of ordinary concrete. Hence, there is a thickness of ≈ 2 m of concrete between the device and the diagnostics equipment located in the basement. This amount of shielding is required to permit unlimited access to this diagnostic gear between pulses and to reduce the noise level in the diagnostic sensor heads during a pulse. The basement diagnostic area may not be occupied during a pulse, since a single pulse would deliver a dose of ~50 mrem, with a larger dose in the vicinity of a penetration. Dose rates from neutron activated objects in the basement immediately after pulsing are $\lesssim 3$ mrem/hr, except for diagnostic gear located at a penetration and cooling water flowing to the chillers.

Activation of Cooling Water

As cooling water flows through the electromagnetic windings in the vicinity of the torus, the water becomes activated by the $^{16}\text{O} (n, p)^{16}\text{N}$ reaction. The water flows in a closed loop through the Main Test Cell, the basement, and the utility tunnel into the mechanical equipment (chiller) room. Even though the ^{16}N activity has a short half-life, $T_{1/2} = 7.4$ s, the time of travel to the chiller room is such that the dose rate next to

Figure 4.1.3-4. Dose Rates from Back End of Neutral Beam Injector

Figure 4.1.3-5. Dose Rates from the TFTR Vacuum Vessel

the apparatus is ~ 1 rem/pulse, and the chiller room cannot be occupied during operation. Because of the short half-life, the ^{16}N activity does not accumulate, and the dose rate 2 minutes following a pulse is down to an acceptable 0.01 mrem/pulse. The activated water will not affect the measurements being made with the diagnostic apparatus located in the basement since the pulse is completed prior to the activated water reaching the basement.

Activation of Main Test Cell Air

The air in the Test Cell becomes activated as a consequence of the neutron flux. ^{41}Ar ($T_{1/2} = 6.6 \times 10^3 \text{s}$) results from reactions induced by thermal neutrons, and ^{13}N ($T_{1/2} = 600 \text{s}$) and ^{16}N ($T_{1/2} = 7.4 \text{s}$) from reactions induced by energetic (>10 MeV) neutrons. Prompt access to the Test Cell can be prevented by this activity. The activity is reduced either by its decay or by venting the air to the atmosphere. The air in the Test Cell can be discharged with an e-fold time of 1800 s. The time (less than 2 hours) for the gaseous activity to reach a tolerable level depends on the number and type of pulses in the experimental sequence. The annual dose at the site boundary due to the release of gaseous radioactivity is ~ 1 mrem. The air in the Test Cell does communicate with the air in the basement.

Hot Cell Design

A Hot Cell has been provided for maintenance of radioactive components. This cell has ordinary concrete walls, 0.61 m thick. These walls provide enough shielding to reduce the dose rate from any of the large activated components, e.g., half of a torus vacuum vessel after irradiation by several thousand pulses, to less than 2 mrem/hr. Lead glass windows are available in the walls to augment the TV system, just as in the Main Test Cell. The windows are used in conjunction with the remote manipulators for carrying out maintenance operations.

Neutral Beam Test Cell

A test cell is also provided for conditioning and maintaining the neutral beam injection arms. Neutrons having an energy of 2.5 MeV are generated in an injection arm when a 120 keV deuterium beam impinges on a calorimeter or beam dump in which deuterium has been implanted by previous pulses. After equilibrium has been established (following ~ 130 injector pulses) $\sim 1 \times 10^{13}$ neutrons are produced per pulse. Concrete walls having a thickness of 1.2 m are required to reduce the dose rate to a tolerable level (<1 mrem/hr), assuming that deuterium is used during the rapid pulsing, viz., 1 pulse every 30 seconds, which is required for conditioning of the ion sources.

Radiation Analyses

The analyses used for the TFTR radiation design have been based primarily on results obtained with the one-dimensional discrete-ordinates code ANISN-W, utilizing anisotropic

scattering to the P-3 approximation, supplemented by analytical calculations of radiation streaming. This approach is justified for the conceptual design because of the frequent design changes that occur during the preliminary stages. Higher-dimensional radiation analyses will be obligatory in the next design phase, however, since the arrangement of the device, the definition of penetrations and the size and shape of the concrete Test Cell will be more firmly established. For example, two-dimensional calculations will be required using a discrete-ordinate code, point kernel calculations, and Monte Carlo studies of radiation transport through the complex geometry of the shields, coils, and penetrations. These calculations will be used to verify and fine-tune the design, but no major modifications are anticipated.

4.1.4 MAGNETIC FIELD TOPOLOGY AND DESIGN

A tokamak-type magnetic-confinement device employs three principal magnetic field producing coil sets. The coil sets produce: (a) the main toroidal field (TF) or confinement field; (b) the ohmic heating (OH) or transformer flux; and (c) the equilibrium (EF) or centering field. Figure 4.1.4-1 shows the location of the turns for the composite EF/OH winding relative to the TF coils for the TFTR conceptual design.

Toroidal Field

The TFTR toroidal field is produced by circular coils equidistant from the major axis of the torus and lying in planes passing through the major axis. The ideal field would be produced by a current sheet in the shape of a torus generated as a surface of revolution with a circle as the generatrix, but for practical considerations, e.g., access for diagnostics and neutral beam injection, a group of 20 individual and discrete coils uniformly spaced about the axis is employed.

The TF coils are centered at a major radius of 2.8 m, have an inner bore radius of 1.4 m and an outer radius of 1.94 m. The inner and outer radii of the copper are 1.45 m and 1.89 m, respectively. It is worth noting that, in order to increase the field strength and decrease the field ripple within the vacuum chamber, the vacuum vessel has been moved to as small a major radius (2.65 m) as is practical. To provide a 4.86 T field in the center of the vacuum vessel at a 2.65 m radius, a total of 64.4 MA turns is required. This is achieved with a winding that has a total of 880 turns with 73.2 kA in each turn. The maximum field in the copper occurs at the inner radius of the copper in the horizontal symmetry plane and has the value 9.54 T; this occurs at $R = 1.35$ m.

Figure 4.1.4-2 is a plot of the magnetic field strength as a function of radius in the horizontal symmetry plane. At radii where the ripple is negligible, the magnitude of the toroidal field can be obtained from the expression

$$B_T = 0.01760 I(A)/R \text{ (cm)} \text{ Tesla}$$

where the percent ripple is defined as

$$\% \text{ Ripple} = 200 \left[\frac{B_T^c(R) - B_T^m(R)}{B_T^c(R) + B_T^m(R)} \right]$$

Figure 4.1.4-1. Location of the Turns for the Composite EF/OH Winding Relative to the TF Coils for the TFTR Conceptual Design

Figure 4.1.4-2. Magnetic Field Strength in Horizontal Symmetry Plane

where $B_T^c(R)$ and $B_T^m(R)$ are the magnetic field strengths on the same circle (which is centered on the major axis) at the plane of a coil and in the plane midway between coils, respectively.

The ripple contours in a vertical plane are shown in Figure 4.1.4-3. Since the contours of constant ripple are obtained by interpolation, smoother contours can be obtained by increasing the number of mesh points in the contour calculation. Figure 4.1.4-3 gives the expected result that the largest ripple occurs in the horizontal symmetry plane, and also demonstrates that there is a large volume within the vacuum chamber where the ripple is less than 1%.

Ohmic Heating Field

The ohmic heating coils provide the flux which, in the process of reversal, induces the breakdown voltage to initiate the plasma discharge, drives the plasma current to its desired peak, and regulates the plasma current distribution during compression. Ideally the flux links the toroidal plasma, and is confined to regions external to the vacuum vessel, in order to avoid the appearance of stray magnetic fields in the plasma region, which could modify the effects of the equilibrium field.

Figure 4.1.4-3. Ripple Contours in a Vertical Plane

Studies are being made currently to determine whether or not the OH coil design can be modified so that any fringing field that does occur within the vacuum vessel has a sharp null at $R \approx 350$ cm and is characterized as a hexapole (or octopole) field. This field, which is transverse to the primary confining toroidal field, would be useful in defining the location, viz., at the null, for the initial breakdown of the plasma.

For the conceptual design, the OH coil consists of 206 turns operating with a maximum current rating of 40 kA. Upon reversal of the 40 kA current, the flux change will be 10.8 Wb (or volt-seconds.) As described below, during an experiment the EF field must increase as the plasma current increases and will contribute to the flux change within the toroidal plasma. This flux change also drives the plasma current. For a discharge with a 2.5 MA plasma current, the total flux change is typically 14.8 Wb.

Figure 4.1.4-4 shows the fringing OH magnetic field in the vicinity of the plasma. Contours of constant OH magnetic field intensity are shown throughout the vacuum chamber. For a plasma centered at $R = 248$ cm, the fringing OH field seen by a 1 MA plasma is less than 2.5×10^{-3} T and by a 2.5 MA plasma is less than 3.5×10^{-3} T.

The equilibrium field serves the primary purposes of controlling the radial location of the hot plasma, stabilizing against vertical movement, and forcing the plasma inward for adiabatic compression. The field intensity required to maintain the plasma in equilibrium is a function of the plasma's location, radius, pressure, current, and current density distribution.

Figure 4.1.4-4. Contours of Constant OH Magnetic Field Intensity (T)

For the conceptual design, the EF winding consists of 56 turns. The maximum current rating is 80 kA; this will provide a field of 0.3 T at the center of the vacuum vessel, which is sufficient to hold a 2.5 MA plasma in equilibrium or to carry out the compression scenarios of interest. An index of the field-line curvature was defined as:

$$n = -\frac{R}{B_z} \frac{\partial B_z}{\partial R}$$

For the standard EF winding, $n = 0.21 \pm 0.03$ over the confinement volume. Figure 4.1.4-5 shows the contours of constant radius of curvature within the vacuum chamber. Eight additional turns are provided for the EF winding for experimental flexibility. By connecting them in series with the other EF turns, so that they in effect either add or subtract turns, an index of curvature of 0.38 ± 0.07 or 0.05 ± 0.04 can be attained. If the index is high, there is high vertical stability of the plasma but low radial stability. Conversely, if the index is low (but not negative for a plasma with circular cross section) there is low vertical stability, but more radial stability. Since radial stability is more critical and more expensive to cure by feedback control, a nominal value of 0.2 has been chosen for the index of curvature. It is anticipated that there would not be need for frequent changes of the index of curvature, and that the additional turns would be reconnected mechanically.

Figure 4.1.4-5. EF Index of Curvature Contours $n = 0.21 \pm 0.03$

Design Evolution

The present configuration of the EF/OH winding system is the result of an evolution of the total device design and of the design objectives for the fields. An early configuration had the OH winding outside of the TF coils and the EF winding between the vacuum vessel and the TF coils. It was proposed that there be a set of EF coils located next to the vacuum vessel and another set of coils located next to the inside surface of the TF coils. The purpose of this arrangement was to reduce the torque resulting from an interaction of the EF and the TF current. The idea was demonstrated to be technically feasible but was dropped because of the large EF power supply associated with it.

For reasons of economy a decision was made to place the EF turns outside the TF coils and to reduce the dimensions of the TF coils. A study of the induced eddy currents in the TF coils showed that this could be done with no serious consequences. The EF turns were to be intermingled with, or alongside, the OH turns. It was realized that during the flat-top portion of the plasma current, the OH currents and the EF currents in the inboard or throat region were opposing each other. This led to the idea of combining the two windings in this region into "hybrid" turns which carry the sum of the EF and OH currents. This reduces the space required in a critical region, as well as the cost of the winding and the ohmic power losses. In the design of the composite EF/OH winding, as few turns as possible are located on the outboard side of the TF coils since disconnects must be provided for these turns in order to disassemble the device.

At one time, an effort was made to increase the inductive coupling of the EF winding with the plasma. However, it was found that an appreciable improvement could not be gained without major and expensive changes in the winding configuration.

Mechanical constraints have had a large influence on the design of these windings. Two of the more difficult constraints have been the space required for the neutral beams and the large number of turns required by the throat region. Other significant constraints are the spaces required for diagnostic ports and for mechanical supports.

The evolution toward higher winding currents and larger but fewer conductors has had the effect of producing more mechanically robust windings and fewer cooling water connections. An important benefit has been the possibility of using bolted connections on the outboard coils for taking these coils apart.

Design Code Evolution

Design techniques using codes based on linear programming were used first on the EF winding. As experience was gained and requirements added, the code evolved into its present form. The design codes in general minimize the winding volume subject to a variety of constraints on performance and available space. This approach permits relatively explicit constraint statements as to the performance requirements.

The more recent OH winding designs have also used a linear programming approach, the major constraints being an upper limit on the ohmic heating field in the plasma.

Modifications Needed in the Future

Work is now progressing on the development of a code which is aimed at modifying the design of the OH winding. This modification will cause the OH winding to produce a hexapole null for use in controlling the plasma position at startup. Since this null results from the difference of large opposing fields, it will be necessary to use integer linear programming so that the null is not lost in making an integer turn approximation to non-integer design. This modification will involve removing some turns from the present OH winding and adding other turns elsewhere.

The provision for a controlled radial field which can be used in a feedback system to control the vertical position of the plasma has been proposed and is under study. This field may be generated by coils that have an additional use, such as producing fields that will approximately cancel out any stray radial fields.

The modification or time delay of rapidly changing fields due to eddy currents in the TF, EF, and OH conductors, as well as in conducting structural members, requires further study. An integral equation formulation of the problem has been completed and will be developed into a computer code. This will provide convenient calculation of these eddy current effects.

4.1.5 EF/OH/PLASMA CIRCUITRY

Figure 4.1.5-1 is a schematic of a simplified circuit that shows the essentials of the OH and EF circuits including the mutual coupling with the plasma. A computer code COPE, has been developed for analyzing these circuits. Some results obtained with the code are given in this section. The COPE code has been designed to analyze two modes of operation: (1) non-compression or plasma development for a specified geometric location, and (2) a pre-compression or plasma development as for case 1 followed by compression of the plasma induced by pulsing the equilibrium field. During the compression there may be a controlled ohmic heating pulse so regulated that no skin current develops on the plasma.

Plasma Buildup Non-Compression (2.5 MA Plasma)

For this discussion refer to Figure 4.1.5-1.

Prior to the signal given to initiate the plasma, switch 1 is closed, switch 2 is open. As an example assume that initially cur 1 = 40 kA, and that it is desired to create a 2.5 MA plasma. At this time cur 1 and cur 4 are identical, and the ohmic heating coil is fully energized with 5.4 Webers of flux encircled by the vacuum vessel that contains the gas to be heated. At this moment the magnetic energy stored in the ohmic heating coil is 21.8 MJ.

Figure 4.1.5-1. EF/OH/Plasma Circuit

At the instant the plasma is to be initiated, switches S1, S3, and S4 are in the open position while switch S2 is closed so that the ohmic heating coil current (i_1) flows in the circuit containing C, RC, ROH, LOH and RH. The capacitor C is biased at the voltage EC, which may be zero if so desired, and RH the series resistance is selected such that the bias EC plus all the resistance voltage drops will have a combined voltage sufficiently high to give a plasma open-circuit voltage of sufficient magnitude to initiate the discharge. For the conceptual design, the pre-strike volts per turn is assumed to be 150 V, with the bias and resistive voltage totaling approximately 31 kV at the ohmic heating coil terminals. Studies are being made of the consequence of increasing the initial plasma voltage to 300 V. In general, computer studies show that larger plasma densities can be tolerated with larger breakdown voltages. Larger impurity densities can also be tolerated at the higher voltages.

A large resistance RH is needed to establish the pre-strike volts in the plasma, but the losses in the combined circuit resistance would cause an extremely rapid collapse of the ohmic heating field and seriously limit the amount of energy transferred to capacitor C at the time the current passes through zero. Since it is advantageous for the capacitor to accept as much energy as it is capable of storing so that it may effectively build up a reverse current in the ohmic heating coil, the switch S3 is closed ~ 5 ms after the initiation cycle is begun. The reduced resistance RHP allows 7.5 MJ to be stored in the capacitor when the current in the capacitor and ohmic heating coil passes through zero. To conserve the energy of the capacitor as it drives the ohmic heating coil current after current zero, switch S4 is closed to eliminate RHP from the circuit and to minimize losses in the circuit from that time on. The voltage EC at its peak will be several thousands of volts and the capacitor will drive the ohmic heating coil and the plasma at a decreasing rate as the voltage EC falls off with the transfer of energy to the ohmic heating circuit. When EC reaches the value EOH, i.e., the rating of the ohmic heating power supply voltage, S1 is reclosed. (It must be recognized that voltage reversing switches not shown will have reversed the polarity of the EOH power supply during the period when S1 was open and i_1 decayed to zero.) After the voltage EOH is applied, the plasma current will be driven upwards so long as EOH exceeds the resistive volts drop in the circuit and so long as i_1 increases at such a rate that the voltage drop around the plasma is exceeded by the voltage induced by the ohmic heating coil rate of change of flux.

During the time that the ohmic heating current changes are proceeding (essentially without control), the plasma current is started, reaches a value of 1 MA in 60 ms and is raised to a level of 1.8 MA in 150 ms. The time for the plasma current to rise to 1.8 MA can be varied by changes in the value of the series resistor RHP. However, it must be recalled that the energy dissipated in that resistor will also change. To keep the plasma from wandering off into the vacuum vessel walls, a precise current is required at each moment in the EF winding. Hence, a precise forcing voltage is required of the EF power supply. From 150 ms to 780 ms, the current in the OH coil is driven more negative by the OH power supply, which in turn drives the plasma current from 1.8 to 2.5 MA. The currents in the OH coil, the EF coil, and the plasma for this example are shown as a function of time in Figure 4.1.5-2.

Figure 4.1.5-2. Currents in the EF & OH Coils & Plasma as a Function of Time for 2.5 MA Plasma

The voltage E_{OH} has been assumed to be 1.5 kV in this calculation. In the course of the discharge, about 15 Vs are consumed of which 2 Vs are expended on the plasma resistance. Of the 15 Vs, ≈ 11 Vs are provided by the OH system and ≈ 4 Vs by the EF system. A peaked current distribution has been assumed for determining the internal inductance of the plasma, and $\beta_p = 0.7$ has been used in determining the required EF current.

The value of 0.7 represents the final value of β_p . Actually the plasma particle pressure will increase during the time interval from 150 ms to 650 ms, as a result of energetic neutral beam injection: the kinetic temperature of the core of the plasma will increase from ~ 1 keV to > 6 keV and the plasma current from 1.8 MA to 2.4 MA during this same time interval. During the earlier portion of the pulse this has $\beta < 0.7$, which tends to make the required EF field smaller; however, there is a compensating effect: during the plasma formation, the plasma minor radius will be smaller than the assumed radius, and this makes the required EF field larger again. At 2.5 MA the required EF current is 75 kA; the power supply is capable of delivering 80 kA.

Figures 4.1.5-3 and 4.1.10-1 show the instantaneous power requirements as a function of time for the EF and OH circuits during the formation of a 2.5 MA plasma.

Figure 4.1.5-3. EF and OH Instantaneous Power Requirements for 2.5 MA Plasma

Plasma Compression (1.0 MA Plasma)

A pre-compression plasma is formed at a major radius of 310 cm, as described in Section 4.1.2. This phase is analyzed with the COPE code by carrying out a calculation similar to the one described above for the high-current discharge, i.e. a non-compression case. However, to form the pre-compression plasma, not as much energy need be stored in the OH coil initially since the plasma current is only 0.8 MA (weak compression) or 0.67 MA (strong compression). As an example, consider a weak compression experiment. During the formation of the pre-compressed plasma, and either prior to or subsequent to attaining a plasma current of 0.8 MA, 32 MW of energetic deuterons are injected into the plasma for a period of 0.1 s. As described in earlier sections, e.g. Section 4.1.2, the current in the EF winding will be increased during this time interval so as to move the plasma inward to its final pre-compression position at $R = 310$ cm. The movement of the plasma during this period assures a plasma current that is distributed throughout the plasma volume. For the scenario depicted in Figure 4.1.2-1, it was assumed that the plasma current increases from 0.5 MA to 0.8 MA during the injection. However, the plasma current could be raised to a higher level prior to injection and within the same time interval, ~ 50 ms, by storing a higher energy initially in the OH coil, and by leaving a high series resistance RHP in the circuit so as to maintain a high plasma voltage and also to dissipate a large portion of the stored energy. If β_p increases from 0.17 to 2.4 during the 0.1 s injection

interval, as computer calculations indicate for plasma formation prior to weak compression, the current in the EF coil (cur 3) must be increased by about a factor of 3 in order to arrive at a plasma centered at $R = 310$ cm. This increase in EF current will become the dominant factor in driving the plasma current as the demand on the OH circuit decreases.

The voltage and current requirements in the EF and OH circuits are sensitive to the rate of change of β_p as well as to its magnitude. During the injection phase (as described above) the voltage that is available (1500 V) to drive the OH circuit has less margin for the weak compression mode than it has for the strong compression mode. The reason for this is that a plasma current of 0.8 MA must be achieved, as compared to 0.67 MA for strong compression. In addition, for these particular calculations $\beta_p = 2.4$ following injection for the weak compression mode as compared to $\beta_p = 3$ for the strong compression mode, and hence the EF circuit provides more driving force for the plasma during the strong compression mode. Immediately following the injection, the plasma is compressed. To compress the plasma (0.67 MA to 1.0 MA for strong compression or 0.8 MA to 1.0 MA for weak compression), the equilibrium field current must be driven to higher values in a relatively short time interval (~ 25 ms for weak compression). The driving voltage required will far exceed the 3 kV rating of the power supply. To augment the voltage and to supply some of the EF coil input power, an auxiliary capacitor bank ZC is pre-energized with 7.5 MJ at several thousands of volts. Upon closing switch S5, the capacitor voltage is added to the power supply voltage and as the current rapidly rises, the plasma moves to a smaller major radius. Since the plasma is highly conducting, the magnetic flux within the current loop cannot change during the compression. On the assumption that surface or skin current should not be permitted on the plasma surface, the COPE code can determine the ohmic heating current required to conserve the flux within the plasma current loop while the current distributed within the plasma increases as $1/R$ during the adiabatic compression.

In contrast to the pre-compression case, with the EF circuit a slave to the unregulated OH forcing function, the roles are reversed during compression with the OH circuit a slave to the unregulated but programmed response of the combined capacitor and EF power source voltage.

Just prior to compression, switch S2 is opened so that the capacitor voltage $EC = EOH$ will not tend to "clamp" the voltage across the OH coil at such times that EOH is controlled and swinging at a high volts-per-unit-time rate.

A simplified set of COPE calculations was carried out for scoping purposes. These calculations were carried out assuming β_p to be constant during compression; an average value of $\beta_p = 2.4$ was assumed for weak compression. The self inductance of the plasma was based on a peaked current distribution. The results of the COPE calculations indicate that the 1500 V available from the OH power supply is sufficient to drive the OH coil as required during a 25 ms weak compression, when the EF winding is connected to have an index of curvature of $n = 0.05$, but not for the larger values of n .

However, since the OH current requirements during the compression portion of the pulse are much less than the rated maximum current of the OH power supply, the rectifiers could be reconnected while the OH rectifiers are removed from the circuit, i. e., immediately following initiation of the plasma current, via relatively easy switching so as to provide twice the voltage at half the current. This will allow weak compression for all possible values of n .

The total powers required from the MG sets for the EF plus OH circuits during weak compression, and under the assumptions given above, are shown in Figure 4.1.5-4. Since a total of 660 MW are available and 380 MW are required for the TF coils, 280 MW are available for the EF plus OH circuits. It should be noted that no power is required for the neutral beams during compression (the plasma would represent a moving target). It can be seen from Figure 4.1.5-4 that for weak compression the power requirements for the EF + OH circuits for all three indices of curvature are well under 280 MW.

It can be seen from Figure 4.1.5-4 that in order to eliminate surface currents, the OH power requirements during the 25 ms compression period are greater the larger the index of curvature of the EF. However, the larger the index of curvature, the smaller the change in the EF current required to carry out the compression and consequently the smaller the energy demand from the series capacitor in the EF circuit.

Figure 4.1.5-4. Power requirements for 1.0 MA plasma during weak compression experiment. OH current swing 20 kA to ≈ -20 kA, $\beta_p = 2.4$. No plasma surface currents. Peakod current distribution.

If the EF winding is connected (see Section 4.1.4) for an index of curvature of 0.38, strong compression with $\beta_p = 3$ can be accomplished in 35 ms with the power supplies specified for the conceptual design. At the completion of the strong compression there would be a plasma surface current of -10% for the nominal 1 MA plasma current.

Figures 4.1.10-2 and 4.1.10-3 show the instantaneous power requirements during a weak and strong compression pulse, respectively. For the weak-compression example an index of curvature for the EF of 0.05 has been used, and for the strong-compression example a value of 0.38 has been used. During the weak compression it has been assumed that β_p increases from 2.4 to 2.7, and for the strong compression an average value of 3.0 has been assumed. The powers at no instant exceed the 660 MW that are available.

4.1.6 THERMAL AND HYDRAULIC DESIGN

Thermal and hydraulic analyses were performed for the toroidal field coils, ohmic heating coils, hybrid, and equilibrium field coils, to determine the thermal performances of these components. The results indicate that the cooling water system is more than adequate to remove the heating energy of the field coils and the silicon controlled rectifiers for the nominal design flat-top time of 1 second and pulse cycle time of 300 seconds. The field coil temperatures are below design limits. The thermal energy balance and the design parameters of the cooling water system are shown in Table 4.1.6-1 and the field coils thermal design summary is shown in Table 4.1.6-2. Multiple-pulse thermal analysis of the vacuum vessel wall protection plate was also performed so that the maximum neutral beam power can be selected. A summary of the thermal and hydraulic design results is presented in this section.

TABLE 4.1.6-1 COOLING WATER SYSTEM DESIGN SUMMARY

Water chiller capacity	11.3 MW (3300 Tons)
Main Circulating Pump Capacity 2 at 0.114 m ³ /s (1800 gpm) and 6.9 x 10 ⁵ Pa (100 psi)	0.227 m ³ /s
Pump Motor Capacity 2 at 74.6 kW (100 HP) each	149.2 kW (200 HP)
Ballast Tank	41.6 m ³ (11,000 gal*)
Distribution Line Size	45.72 cm (18 in.)
TF, OH, EF & SCR Loads TF = 8.84 MW OH = 0.15 MW EF = 0.17 MW SCR = 0.74 MW	9.9 MW
Estimated System Pressure Losses	5.9 x 10 ⁵ Pa

*Ballast tank volume of 5500 gallons was used in analysis.

TABLE 4.1.6-2 TF, OH & EF COILS THERMAL DESIGN SUMMARY

	TF	OH	EF
Number of coolant channels per conductor	1	1	1
Conductors per coil	88	1	1
No. of parallel cooling channels per coil	44	40 ⁽¹⁾	40 ⁽¹⁾
Channel Length, m	22.7	52.3	52.3
Coolant Velocity in channel, m/s	3.3	1.3	1.3
Cooling Channel Diameter, cm	0.953	0.953	0.953
Maximum Coil Temperature Rise, K ⁽²⁾	43.0	36.0	29.0
Maximum Radial Temperature Difference, K	12.0	1.0	2.0
Maximum Coolant Temperature, K (°C)	305.2 (32.2)	312 (39)	308 (35)
Channel Pressure Loss, Pa ⁽³⁾	3.58×10^5	1.3×10^5	1.3×10^5

Notes:

- (1) Total parallel cooling channels for OH, HY and EF Coils
- (2) Based on 1-second flat-toptime and initial coil temperature of 283 K
- (3) For a straight-through passage. Losses due to coupling, fitting, turns, etc. along the channel not included.

Toroidal Field (TF) Coil Analysis

The TF coil design used in this analysis is shown on Figure 4.1.6-1 and differs slightly from the recently changed conceptual design shown on Figure 4.2.3-2. Initially the thermal design criterion selected was based on an allowable thermal stress of fifty percent of the fatigue strength of oxygen-free hard copper. The number of operating cycles was determined by the requirement that a power pulse would occur every 5 minutes for five years, or 1×10^5 cycles. With these constraints the maximum allowable temperature difference on any one cross-section was 40 K in the copper coil. The pressure drop for each cooling passage was assumed to be no more than 5.2×10^5 Pa (75 psi) for practical purposes. The width of the copper bars dictated that the hole size be in the range near to or less than 1 cm (0.40 in) which determined the coolant water velocity to be 6.7 m/sec (22 ft/sec) with a copper-to-water heat transfer coefficient in the range of 1.7 to 2.7 watts/cm² K U.C. depending on the water temperatures.

Two designs, one using a single elliptic hole and another using two elliptic holes in each turn of the copper coils, were evaluated. With the conservative transient analysis used initially the single-hole design was found to be marginal while the two-hole design was adequate, serving as the basis for the present design approach. The basis of the temperature design criteria has also been re-examined. A better thermal stress analysis, considering plastic deformation, led to a coil temperature limit of 393 K. Convenience and safety consideration indicated that no steam generation should occur in the event of the rupture of a pressurized connection. Hence, a maximum coolant temperature operating limit of 100°C was also imposed.

The insulation integrity temperature limit is 423 K and hence is not the governing criterion.

An analysis of the current reference design was carried out for the worst turn which has the minimum cross-sectional area at the shaved region, to determine the peak coil temperature and the maximum transverse temperature difference between the peak temperature and the cooling wall surface temperature in a multiple pulse operation. This turn was analyzed using a detailed heat conduction and fluid flow model. The model includes the following major cooling water system equipment: the TF coil with coolant velocity of 3.3 m/sec, ballast tank (5500 gallons), chillers (3000 tons out of the total 3300 tons for TF coils cooling) and the connecting pipes. The physical location of the chillers relative to the ballast tank in the flow stream and the scheme of cooling the TF coils conductor by one turn per circuit or by two turns per circuit were first investigated to establish a suitable cooling system arrangement. It was determined that for the most effective operation of the chillers, the chillers should be located

downstream of the ballast tank (see Section 4.1.15). Also it was determined that no significant difference in coil and coolant temperatures was introduced by cooling two turns per circuit (Section 4.1.15). Therefore the scheme of cooling by two turns per circuit was adopted. In the conduct or arrangement as shown in Figure 4.1.6-1, the cooling water enters turn No. 1 and exits before cooling turn No. 3. Another parallel channel enters turn No. 3 and exits before cooling turn No. 5 and so on. The channel consisting of turns No. 3 and 4 was the one analyzed.

The calculations were performed using the TAP-B code⁽¹⁾. The operating characteristics of the chillers were simulated by the code. The internal heat generation rate of the coil material was calculated by inputting the field current as a function of time and the copper electrical resistivity as a function of temperature. The TF coil current profile during a pulse is shown in Figure 4.1.6-2 in which the OH and EF field currents are also shown. The results of OH and EF coils analyses are described in the next section. The copper resistivity used had a linear relationship with comparative being 1.6 ohm-cm at 273 K and 2.0 ohm-cm at 383 K. It is noted that the TF coil current profile assumed a 14-second buildup time and a 7-second decay time. The longer times result in a higher integrated energy deposition, and the coil and coolant temperature calculated would then be on the high (conservative) side.

To start the calculation, the initial coil material and the water temperatures were assumed to be at 283 K, which was achieved by chilling the entire system from ambient conditions with the chillers on and without field current. A 300-second pulse cycle with one second of flat-top time was used as the nominal operating conditions. Starting from the first pulse, the peak coil temperature of the hottest coil increases slightly after each pulse until an equilibrium condition is reached in which the heat generated can be removed by the water during the cool-down period. Figure 4.1.6-3 shows the maximum and the average coil temperatures of the hottest turn during the 7th pulse, at which equilibrium conditions were first reached. The maximum coil temperature is only about 326 K. The coolant temperatures at the coil exit and at the chiller inlet and outlet are shown in Figure 4.1.6-4.

With the same model, the effect on the maximum coil temperature of extending the flat-top time to more than one second was also investigated. The results are shown in Figure 4.1.6-5. The solid line AB in the figure shows the cooling system is capable of being operated with a flat-top time up to 4 seconds. Beyond 4 seconds the heat load is above the capability of the chillers. The peak coil temperature continues to increase after each pulse with more than 4 seconds of flat top time, as shown by the dashed lines.

The maximum coil temperature difference gradient (ΔT) is shown in the figure. For one second of flat-top time the maximum temperature difference is about 10.5 K and for 4 seconds of flat-top time, about 14.5 K.

(1) "TAP-B, A General Computer Code for Coupled Hydraulic and Heat Conduction Analysis", A. Y. Lee, WANL-DRM No. 54186, June 18, 1971.

Figure 4.1.6-2. TF, OH and EF Field Current Profiles for Thermal Analysis

Figure 4.1.6-3. Material Temperatures of Hottest Turns

Figure 4.1.6-4. Coolant Temperatures

Figure 4.1.6-5. TF Coil Thermal Operating Limits

The pressure losses along the cooling system for a water flow rate of $0.227 \text{ m}^3/\text{s}$ (3600 gpm) was estimated to be about $5.9 \times 10^5 \text{ Pa}$ (85 psi).

Besides the analysis for nominal operations, two abnormal or faulted condition operations were analyzed: (1) coolant channel blockage and (2) loss of chillers refrigeration. For the channel flow blockage, the case with one second of flat-top time and coolant flow rate reduced to 1/3 of the nominal value (reduced to coolant flow velocity of 1.1 m/s from 3.3 m/s) was analyzed. The calculated maximum coil temperature at the worst turn was 338 K as compared to the nominal value of 326 K. It appears that the coils could still be operated with as little as 1/3 of the nominal flow rate. This is mainly due to the extremely high thermal conductivity of the copper, so that the heat generated at the minimum area section is conducted away axially to other portions of the coil. For the case of loss of chiller refrigeration, because of the heat capacity of the water in the ballast tank, the coil could still be operated for 6 pulses before the coolant temperature reaches the design limit of 373 K.

Steady state operation of the TF coils was also analyzed. With a 3000-ton chiller capacity for TF coil cooling, the maximum permitted steady state TF coil current was determined to be about 20% of the maximum value of 36.6 kA/conductor (73.2 total). This 20% steady state capability may be needed if the discharge method is used for cleaning the vessel wall.

Ohmic Heating, Hybrid and Equilibrium Field Coils

The ohmic heating (OH), hybrid (HY) and equilibrium field (EF) turns as well as the TF coils, are cooled by water from the machine area water cooling system. The dimensions of the copper conductors for the OH and HY turns are $3.81 \times 3.81 \text{ cm}$ (1.5×1.5 ") and those for the EF turns are $5.08 \times 5.72 \text{ cm}$ (2.0×2.25 "). The single cooling hole per conductor is 0.95 cm ($3/8$ ") in diameter. The total length of the coil was calculated to be about 2090 m. One-half of this total is located at the top half of the toroid and the other at the bottom half of the toroid.

Within the overall pressure drop and available coolant from the cooling water system, 40 parallel coolant channels were selected. The length of conductors to be cooled per channel is then 52.3 m. A flow rate of 0.091 kg/s (0.2 lb/s) per channel was used for the analysis. The field current profiles for the OH and EF turns were shown previously in Figure 4.1.6-2. The summary of the analysis results is shown in Table 4.1.6-3. It is seen from the table that under normal operating conditions, the maximum temperatures of the OH and EF turns are below the copper peak temperature design limit of 393 K for flat-top time up to four seconds, which is the upper limit of flat-top time for the TF coils discussed previously in this Section. However, under faulted conditions, when the EF/OH is disrupted, the full amount of EF current would apply to the HY coils. The maximum hybrid coil temperature becomes 348 K with one second of flat-top time and over 423 K with four seconds of flat-top time. In order to have 4 seconds of flat-top capability for all the coils, additional coolant or a shorter channel length for the HY turns would be required.

TABLE 4.1.6-3 SUMMARY OF OH, & EF COIL COOLING DESIGNS

Turn Type	Dimensions	Peak Field Current (kA)	Flat-Top Time (s)	Max. Coil Temp. (K)	Max. Axial ΔT (K)	Max. Radial ΔT (K)	Channel ΔP 10^5 Pa
OH	3.81 x 3.81 cm	40	1	319	28	1.0	1.38
			4	361	60	2.0	1.38
HY	3.81 x 3.81 cm	(80) ⁽¹⁾	1	348	52	3.0	1.38
			4	> 423	—	—	—
EF	5.08 x 5.72 cm	80	1	312	24	2.0	1.38
			4	387	87	3.5	1.38

Operating Conditions:
(a) No. of parallel channels = 40
(b) Length of Channel = 52.3 m
(c) Flowrate per channel = 0.091 kg/sec (0.2 lb/s)

Notes:

- (1) Hybrid turns under faulted condition at which the EF field current is applied to the turns resulting in a higher coil temperature than if the nominal current is applied to the hybrid turns.
- (2) The 1.38×10^5 Pa (20 psi) is the momentum and frictional pressure losses in 52.3 m (2058 in) of straight-through channel. Losses of turns from one conductor to another within the circuit should be added to obtain the overall pressure loss in the circuit.

Vacuum Vessel Wall Protection Plate

The TFTR is designed to have a number of neutral beam injectors to inject energy into the plasma during operation. Under normal operating conditions, the bulk of the energy will be absorbed by the plasma, while the remainder would impinge onto the vacuum vessel. During a faulted condition, the unattenuated energy reaching the vacuum vessel wall would cause failure. Tungsten plates are therefore used locally to protect the vessel wall.

In its current design, the vacuum vessel wall protection plate consists of an 0.95 cm thick tungsten plate with an 0.33 cm thick molybdenum backing plate. The inside surface of the plate is cut into small squares to minimize thermal stress. The analysis assumes that a portion of the protection plate located at the axis of the neutral beam receives the peak of the attenuated energy from the neutral beam. This worst case was therefore analyzed. The mode of heat transfer considered was heat input to the tungsten surface and heat losses by thermal radiation from the front and back surfaces to the surroundings. The beam injection duration and the pulse cycle time used in the calculations were 0.5 seconds and 300 seconds, respectively. Since the details of the neutral beam design are not available at this time, multiple pulse transient analysis was carried out parametrically, using different values of input heat flux to determine the maximum beam power allowable at different values of plasma attenuation and angle of impingement under normal and abnormal operating conditions.

The calculated protection plate surface temperature responses during the first pulse, with an input heat flux of 7.0 kW/cm^2 and of 1.0 kW/cm^2 , are shown in Figure 4.1.6-6. The figure shows that the plate surface temperature increases rapidly during the 0.5 second of energy deposition period, drops off also rapidly in one second or two and decays slowly afterwards. At the low level of input flux (1.0 kW/cm^2) the surface temperature cools very slowly below 448 K, due to the ineffectiveness of radiative heat transfer at low surface temperatures. However, the peak temperature reached at the end of the 0.5-second energy deposition time and the temperature at the end of the pulse, continue to increase after each consecutive pulse, until an equilibrium condition is reached in which the energy deposited during the first 0.5 second can be dissipated during the remaining 299.5 seconds of cooldown. Figure 4.1.6-7 shows the plot of the peak plate surface temperatures vs. number of consecutive pulses (300 seconds period) at different surface heat fluxes reaching the surface. It is seen that the peak surface temperature levels off after a few pulses, especially at higher input heat fluxes. This indicates that the protection plate can be effective for a large number of consecutive pulses during a test series under normal operating conditions. For example, with an attenuated heat flux of 1.0 kW/cm^2 reaching the surface, the peak surface temperature is below 1243 K under normal operating conditions.

Under faulted operating conditions the heat flux reaching the plate surface would be very high. For example, as seen from Figure 4.1.6-6, if the unattenuated heat flux

Figure 4.1.6-6. Peak Protection Plate Temperature Vs. Neutral Beam Flux

Figure 4.1.6-7. Peak Surface Temperature Vs. Neutral Beam Flux and Number of Pulses

reaching the surface was 7.0 kW/cm^2 , the peak plate surface temperature rise is about $3477 - 283 = 3194 \text{ K}$. This magnitude of temperature is already quite close to the melting point of the tungsten, 3643 K . The results of a parametric study to determine the maximum allowable neutral beam peak flux at a faulted condition, which might occur for any one pulse during a multiple shot test, is shown in Figure 4.1.6-8. The figure shows that the maximum value is between 6.5 and 10 kW/cm^2 depending on the plasma attenuation and the angle of impingement. The power of the beams can therefore be selected based on the faulted condition constraint, so that they can be operated under both normal and abnormal conditions in a multiple pulse series of tests.

4.1.7 DEVICE STRUCTURAL ANALYSIS

Design adequacies have been demonstrated by structural evaluations of key areas. These analyses include: the toroidal field, ohmic heating and equilibrium field coils along with their supporting structure, the vacuum vessel and the machine substructure. The major loading condition used in these analyses was structural fatigue caused by the $100,000$ pulse cycles. Allowable stresses were obtained from various sources of published data and the amounts of conservatism are not known. Therefore, it is planned to establish a project materials manual with consistent and conservative design values for the next phase of the work. Although ASME "N" stamping is not required, and code materials are not always used, the intent of the ASME code is as a guide in the structure analysis. The discussions that follow provide a brief summary of these calculations.

The most significant change since the original design concept produced in March 1974 has involved the TF coil support, including the external torque frame. It was felt that supporting the TF coils at the top and bottom by the torque frame was not adequate, because: (1) the coil slopes would be large at the horizontal midplane, and (2) the torque frame was unnecessarily massive. The present integral concept of plates between the TF coils essentially supports the coils along most of the outboard circumference with cylinders which are the most efficient type of torque carrying structural members.

Toroidal Field Coils and Support

The structural evaluations performed on the toroidal field coil assembly include sensitivity studies for the coil angular deflections induced by vertical fields and for the coil stresses from combined field-loading conditions.

Toroidal Field Coil Slope Sensitivity Analysis

The coil angular deflection sensitivity study provides comparative results for: a) three different coil-to-coil bolt patterns, b) analytical models using material property variations resulting from tolerances on the copper and epoxy winding and potting dimensions, c) shear panel bolt tolerance, d) coil-to-coil interface bolt tolerance, e) coil-to-coil interface friction conditions and f) the load variations for different vertical field indices of curvature and for the condition of an equilibrium field with no plasma.

Figure 4.1.6-8. Maximum Neutral Beam Flux at the Protection Plate

The matrix describing the variables of the lateral deflection study is provided in Table 4.1.7-1 by a comparison of eight different finite element computations, numbered 21 through 21G. The copper centroid slope is defined on Figure 4.1.7-1, which shows the variation of slope with coil angular position; the maximum value of the slope shown here for each computation is indicated in Table 4.1.7-1. The comparison provided by computations 21 and 21A shows that a 13% reduction in the shear modulus of the copper/epoxy winding resulted in a 2% increase in the coil slope. The smaller value of shear modulus here corresponds to fabrication tolerances of maximum epoxy and minimum copper thicknesses. Computation 21B shows virtually no change from 21, where the increased epoxy thickness between the copper turns is taken away from the epoxy radial fill between the steel case and the copper turns. The maximum bolt shear load for this bolt configuration (21, 21A, 21B) was 0.672 MN (151,180 lb) from computation 21 for the bolt at the horizontal midplane ($\theta = 0$). The minimum bolt shear load was 0.187 MN (42,251 lb) for the bolt located at $\theta = 100^\circ$ (1.75 rad). The computations 21, 21A and 21B were all made with the condition of no friction forces at the bolted interface between adjacent toroidal field coils, and therefore the bolts are assumed to carry the entire shear load. By comparison, if relative motion between coils is prevented at that interface by design or by large contact forces acting on frictional surfaces, there would be no direct shear on the bolts, and the coil lateral deflections would be less than one milliradian per computation 21C.

TABLE 4.1.7-1 A PARAMETER COMPARISON FOR THE TF COIL LATERAL DEFLECTION (SLOPE) SENSITIVITY ANALYSIS

Computation Number	Maximum Copper Centroid Slope (milliradians)	Composite Material Shear Modulus (GYZ)	Epoxy Radial Fill Thickness	Epoxy Lateral Fill Thickness	Bolt Pattern at Coil Angular Location* (θ , Degrees)	Bolt Tolerance	Magnetic Field Crossing the TF Coil Current and Coil Moment about Copper, Hor. Q_L	Coil Interface Condition	
21	2.67	1.33 x 10 ⁶ psi (9.15 GPa)	0.200 in. (0.508 cm)	0.100 in. (0.254 cm)	13 Bolts at 0 +20 +40 +60 +80 +100 +120	P E R F E C T F I T	EF + Plasma 50.64 x 10 ⁶ in-lb (5.72 MN-m)	** No Friction	
21A	2.72								
21B	2.67								
21C	0.79		0.100 in. (0.254 cm)					No Slip	
21D	2.59		0.404 in. (1.016 cm)	0.354 in. (0.900 cm)	13 Bolts at 0 +4 +20 +40 +60 +80 +120	0.006 in. (0.015 cm)	EF + Plasma 53.90 x 10 ⁶ in-lb (6.1 MN-m)	** No Friction	
21E	3.86						EF, No Plasma 78.37 x 10 ⁶ in-lb (8.85 MN-m)		
21F	4.05								
21G	3.82				13 Bolts at 0 +4 +20 +40 +60 +80 +120	P E R F E C T F I T			

*The coil angular location, $\theta = 0$ is at the maximum machine radius as indicated on Figure 4.1.7-1

**The bolts carry all shear load.

Figure 4.1.7-1. Toroidal Field Coil Slope Versus Coil Angular Location

The computations 21D, E, F and G are based upon the worst possible combination of copper and epoxy, giving a low shear modulus for the composite and likewise giving very thick epoxy fillers between the windings and the steel encasement. The equilibrium field coil configuration used to produce the overturning moment on the toroidal field coils was a hybrid (the H7 configuration) design that could provide an index of curvature of 0.15, 0.33 or 0.52. A tradeoff study was conducted to determine the index of curvature condition which would produce the largest moment on the TF coil. The greatest moment corresponded to the 0.15 index of curvature which was applied to the TF coil for both the case of EF with no plasma and the case of EF plus the field from a 2.5 MA plasma at a machine radius of 2.48 meters.

The results of computation 21D show an improved bolt configuration when compared to computations 21A or 21B. The maximum bolt shear load for the 21D configuration was 0.515 MN (115,903 lb) for the bolt located at $\theta = 88^\circ$ (1.54 rad). The minimum bolt shear load was .348 MN (78,278 lb) for the bolt located at $\theta = 20^\circ$ (0.35 rad). The magnetic field for computations 21E, F and G, where there is no plasma, produces the worst case bolt shear loads for an expected operating condition and it is this condition which must be superimposed with toroidal field loads and thermal loads for the combined stress condition on the toroidal field coils. Two comparisons are drawn from these computations. They are: a) two different bolt patterns, and b) the effect of bolt tolerance upon TF coil lateral deflection. The computations 21E and 21G provide the comparison between bolt patterns. The 21E configuration gave a maximum bolt shear load of 0.875 MN (196,882 lb) for the bolt at $\theta = 1.54$ rad, and the 21G configuration gave a maximum bolt shear load of 0.886 MN (199,281 lb) for the bolt located at θ of 1.19 rad. The 0.015 cm (6 mil) tolerance shown for computation 21F is a tolerance on the bolt diameter. All of the bolts were given this tolerance and were permitted to slip by that amount in the direction of the shear load computed in computation 21E. The bolt loads were about 2% lower for the loose condition as opposed to a perfect fit, but by comparison to 21E the copper slope for 21F increased by about 5%.

The other assembly tolerance that can affect the TF coil lateral slope is the shear pin arrangement in the coil shear panels. If all pins are assumed to be centered in the oversized holes in the panels, they can rotate freely in taking up the tolerance by an amount equal to the diametral tolerance divided by the distance between pins that make panel contact upon rotation. With an 0.010 cm (4 mil) clearance on the pin diameters and with a 40.64 cm (16 inch) space between pins, the TF coils could, as an upper bound, rotate on additional 0.25 milliradians prior to shear panel action. Using computation 21D as a baseline the upper limit for expected coil angular rotation for the operating condition of equilibrium plus plasma fields crossing the TF coil current is $(2.59 + .05)$ $(2.59) + 0.25$ about 3 milliradians, and the lower limit is (.79) about one milliradian based upon the results of computation 21C. The results of computation 21E or G show that for the condition of equilibrium field with no plasma, the bolt shear loads could be (in the absence of friction) as great as 0.89 MN (200,000 lb). With a 4.08 cm (2 in.) bolt diameter, the shear stress is about 441 MPa (64,000 psi). This compares to the selected inconel 718 bolt material with a 709 MPa (103,000 psi) shear strength.

Toroidal Field Coil Stress Analysis

The toroidal field coil stress behavior has been evaluated for the separate and combined load conditions of equilibrium field with no plasma, toroidal field and thermal field. A summary of the predicted coil stresses and deflections for these load conditions is provided in Table 4.1.7-2. Here the principal stresses and the maximum shear stresses are provided for the steel structure and the copper winding. Similarly, for the different load conditions of the TFTR device, axial TF coil deflection at the coil top is predicted, along with the device radial deflection at the TF coil outer radial machine position. The stresses shown for the separate load conditions (21H, 21I and 21J) are not additive. The combined primary stresses produced by the superposition of the magnetic field loads are provided by computation 21K. The superposition of thermal stress with the primary stress from the equilibrium and toroidal fields is indicated as computation 21J and 21K.

The results of computation 21H were produced from shear and deflection boundary conditions only. That is, the node point loads from the equilibrium field crossing the TF coil current were not used. The output from computation 21G, defined by Table 4.1.7-1, was used as the loading condition for computation 21H. A comparison of computed stress by computations 21G and 21H showed excellent agreement as less than 1% difference at the finite elements of significant (>5000 psi) stress.

The results of computation 21I were produced by application of the loads produced by the toroidal field crossing the single TF coil current of 3,200,800 amp-turns.

The results of computation 21K were obtained by a superposition of the stress components from 21G and 21I, and then the principal stresses were computed from these combined stresses.

The results of computation 21J were obtained by subjecting the structural finite element model to a uniform temperature change of 306K (33°C). The steel portion of the structure was given a fictitiously low coefficient of thermal expansion to simulate virtually no temperature change in the steel, and the copper/epoxy portions of the model were given true coefficients of thermal expansion.

The primary plus thermal stress combination was made by superposition of the stress components from two load cases 21J and 21K.

The worst combination of stress in the copper occurs at the machine horizontal mid-plane. The stress direction is along the machine axis and the location is at the coil inner radial surface of the copper windings. The maximum combined tensile stress in the copper is 91 MPa (13,212 psi = 17,907 psi primary minus 4695 psi thermal), and this compares to a tensile yield strength of about 55 MPa (8000 psi) for annealed copper and a reported fatigue strength of 117 MPa (1700 psi) for 100,000 cycles of reversed flexure. The maximum combined wedge compression in the copper is about 76 MPa (11,000 psi). This, when magnified by the reduced cross sections and the stress concentration factors of the coolant holes, will result only in localized inelastic flow in the

**TABLE 4.1.7-2 A PARAMETER COMPARISON FOR THE TF COIL COMBINED
STRESS AND LOAD SENSITIVITY ANALYSIS**

		Stress					
		Steel Structure		Copper Winding		Deflection	
Comp. Number	Load Conditions	Maximum Principal Minimum Principal Maximum Shear (psi) (M Pa)		Maximum Principal Minimum Principal Maximum Shear (psi) (M Pa)		At Coil Top, Device Axial Direction	At Coil Outer Radius in Device Radial Direction
21H	Equilibrium Field (No Plasma)	29897 -19158 15291	206 -132 105	5617 -7197 3661	39 -50 25	0.00	0.00
21I	Toroidal Field	38480 -30029 23615	265 -207 162	17545 -7595 9690	121 52 67	.192 in. (.488 cm)	-.0372 in. (-.095 cm)
21J	Thermal	23454 -5919 12532	162 41 86	469 -7925 3103	32 55 21	.047 in. (.119 cm)	.026 in. (.066 cm)
21K = 21I + 21G	Equilibrium & Toroidal	56959 -30077 25819	391 -207 178	17907 -7600 9950	124 -52 69	.192 in. (4.88 cm)	-.0372 in. (-.095 cm)
21J + 21K	Primary + Thermal	72146 -26457 32552	497 -182 225	13212 -10476 9104	91 -73 63	.239 in. (.607 cm)	-.0112 in. (-.029 cm)

regions of the coolant holes. If further analyses show this is detrimental, a relief machined from the copper at the mating interfaces in the wedge region could be provided to reduce that stress (from 11,000 psi to 4000 psi).

There are two regions of significantly high stress in the steel structure. One location of high axial by directed stress is at the machine horizontal midplane in the wedge region at the coil inner radial surface. Here the maximum combined tensile stress is 366 MPa (53,010 psi = 37340 psi primary plus 15,670 psi thermal). This compares to the minimum yield strength of 379 MPa (55,000 psi) for the Armco 18-3 MN stainless steel and a reported fatigue strength of 331 MPa (48,000 psi) for bending with a complete 100 million stress reversals on a rotating beam test specimen. The other region of high stress in the steel structure is at the intersection of the box side wall and the box inner flange, at a location just below the attachment location of a coil shear panel. At this location there is a steep change in curvature on the box side wall in making the transition from a highly stiffened region of shear panels to the open midplane region with no shear panels. The maximum combined stress here is 500 MPa (72,146 psi = approximately 28,100 psi EF plus 28,300 psi TF plus 15,700 psi thermal). This is an area that will require some additional detailed analysis with design improvements.

EF/OH Coil Assembly

The Equilibrium and Ohmic Heating Field Coil System consists of groups of copper windings surrounding the toroidal field coils. The inboard windings are supported from a central support structure. The outboard windings are supported from the toroidal field coils.

The inboard assembly has been arranged in three modules. There is a center section with mirror-image sections above and below. The top and bottom sections are composed of ribs on which various coil groups are spanned. The forces on all windings resulting from the OH, EF and plasma currents have been calculated by using a code known as "Dynamic Forces Code" supplied by Princeton Plasma Physics Laboratory. The magnetic forces produce hoop and bending stresses in the inboard copper windings. The stresses and deflections due to magnetic forces have been calculated. Except for the two outermost pairs of coil stacks, the maximum stresses in all the other coils have been found to be less than 117 MPa (17,000 psi), the fatigue limit of annealed OFHC copper for 100,000 cycles. Some additional reinforcement of these two outermost stacks, possibly in the form of increased cross-sectional area of the windings, is necessary. Thus the inboard windings can be structurally adequate to span the twenty ribs of the support structure.

All the bolts used to hold the assembly together are made of 305 SS and have been shown to be adequately sized. The criterion used was to have sufficient preload torque to maintain all surfaces preloaded throughout the operating life and at the same time keep the bolt stress less than twice the design stress intensity values (S_m) as per ASME boiler and pressure vessel code.

The coil stacks are anchored to the ribs by means of plastic cover plates and through bolts. The maximum stress in the cover plates has been found to be 25.7 MPa (3735 psi), which is below the 100,000 cycle fatigue limit of 103 MPa (15,000 psi) for fiberglass-reinforced plastic.

The top and bottom modules of the support structure have been analyzed for adequacy to resist the magnetic forces acting on the coils. Each of the two modules consists of twenty cantilevered ribs of fiberglass-reinforced plastic. The maximum computed support rib stress is 37.2 MPa (5400 psi). The stress level is within the 100,000 cycle fatigue limit for the fiberglass-reinforced plastic. The computed maximum absolute displacement of the coils (rib deflection plus the span between the ribs) is 1.07 cm (0.42 in.). Even though there is not a deflection criterion to compare against, this value appears reasonably small and its effect on the magnetic field appears to be advantageous. The outboard copper windings are split at the device parting plane and do not take any radial forces. The radial magnetic loads are transmitted to pretensioned 305 stainless steel bands which enclose the coil groups. The radial components of the magnetic forces appear as additional hoop stresses in the bands. The maximum hoop stress in the bands has been calculated to be 64 MPa (9270 psi) which is well within the range of the 100,000 cycle fatigue limit for the AISI 305 stainless steel band material. The maximum circumferential deflection of the bands has been calculated to be 0.8 cm (0.324 in.).

The vertical support rings (made out of fiberglass-reinforced plastic) for the outboard coil groups have been analyzed for vertical magnetic loads in the coils and for the span between toroidal field coils. A computed maximum stress of 13.8 MPa (2010 psi) is within the 100,000 cycle fatigue limit.

Each toroidal band has two tensioning devices located at the device parting plane. The tensioning devices have been found to perform adequately with the required pretensioning levels.

Vacuum Vessel

The TFTR vacuum vessel consists of straight hollow pipe sections which are joined alternately by stiffening rings and bellows covers to form a torus. The principal axis of the stiffening ring and bellows section is rotated 8×10^{-2} rad ($4\frac{1}{2}^\circ$) relative to the pipe section principal axis. Since the stiffening rings and bellows sections have a circular cross section, it is required that the vessel pipe sections be slightly elliptical ($4\frac{1}{2}^\circ$ ellipse) in cross section. The vessel also has a number of different size nozzle penetrations which provide for neutral beam injection, vacuum gauges, various probes, etc.

The design environment assumed for the vessel analysis was that of a uniform external pressure equal to one atmosphere and a vessel temperature of 366 K (93°K). Various failure criteria were investigated, such as vessel stability behavior, stress level, and

deflection. The required vessel wall thickness for the 305 SS vessel material in the design environment was evaluated to be 0.772 cm (0.304 in.). This conclusion is based on conservative evaluations of the vessel stability behavior (safety factor of 3.0 for elliptical cylinders with stiffeners), which was found to be the governing failure criterion. It was also shown to be in agreement with results obtained by an application of the ASME Nuclear Vessel Code for a circular cylinder with limited out-of-roundness tolerances. The use of vessel stiffening rings with a 4.1 cm x 7.6 cm (2 in x 3 in) cross section was shown, by two independent methods, to be structurally adequate.

Other vessel design criteria investigated include vessel stress and deflection. Conservative vessel stress calculations indicate the maximum compressive primary membrane plus bending stress to be 36.1 MPa (5210 psi) [allowable stress is 207 MPa (30,000 psi)] and the compressive longitudinal membrane stress to be 7.79 MPa (1130 psi) [allowable stress is 51.7 MPa (7500 psi) based on longitudinal stability]. The computed maximum vessel deflection is less than 0.457 cm (0.18 in) in a direction normal to the vessel wall. Due to the conservatism of the analysis model, this deflection level is admittedly larger than possible; nevertheless, it is not felt to be an objectionable amount.

The subject of vessel penetrations was handled by presenting a design method which closely conforms to ASME Nuclear Vessel Code rules for internal pressure. Although not finalized, these guidelines should provide for adequate vessel penetration support through provision of proper nozzle spacing and reinforcement. Applications of these design guidelines in vessel penetration design eliminate the necessity for detailed stress evaluations of penetrations subjected to uniform external pressure during conceptual design development.

An analysis of the bellows indicates that stress level and not stability is the governing failure criterion. A conservative evaluation shows the allowable external pressure is 0.175 MPa (25.4 psi). The level of lateral deflection in any given bellows diaphragm was conservatively computed to be 0.041 cm (0.016 in) relative to the welded ends of the diaphragm, thereby eliminating any interference between adjacent diaphragms.

The bellows cover assembly serves as the load path for the toroidal compressive loading in the vacuum vessel at the bellows sections. In arriving at the bellows cover design, a parametric study was made to observe the effects on component stress level of varying several different component parameters. An analysis of a design in which the bellows cover plate is offset by flanges from the vessel wall indicated the governing failure criterion to be stress level. The evaluated minimum required cover plate thickness of 0.952 cm (0.375 in.) and required flange thickness of 1.32 cm (0.520 in.) was based on satisfying the ASME Nuclear Vessel Code allowable stress limits. The need for welding a number of rectangular gussets at the vessel-cover flange interface was demonstrated for this type of design configuration. The maximum compressive deflection occurring in this type of bellows cover assembly was conservatively computed to be 0.010 cm (0.004 in.) in the toroidal direction, a negligible deflection for the bellows to accommodate.

Based on the additional cost of the offset bellows cover plate type of design, the bellows cover section configuration was changed to eliminate the offset and the resulting required gussets. This was accomplished by increasing the vessel wall minor radius so that it would agree with the cover plate radius. Thus, the required bellows cover plate thickness can conservatively be estimated to be equal to that of the vessel wall which is 0.772 cm (0.304 in.).

Substructure

The tokamak device substructure serves as the support structure for the vacuum vessel, toroidal field coils, and much of the device shielding. The substructure itself is composed of a glass fiber reinforced epoxy sheet that is directly supported by transverse-I-beam joists. The joists, in turn, are supported by transverse I-beam girders, perpendicular to the joists, and resting on roller bogies.

The substructure analysis as well as a tradeoff study to minimize fabrication cost were performed on the design concept of June '75. However, the arrangement and number of joists and girders have changed for the present reference conceptual design. Therefore the results (I-beam sizes and stresses) of the earlier analysis are no longer valid. The procedures shown are still applicable and were used to qualify the conceptual design even though not reported.

The reported analysis includes a design optimization procedure for minimizing the cost of the I-beams based on equipment dead weight. Even though the seismic floor response spectrums (vertical or horizontal) have not been generated for the tokamak device in the test cell building, it is expected that the ground motion of 0.05 g, which is considered a peak value for the locale, is sufficiently small that a substructure of this or similar configuration could easily be made to handle any realistic amplification factor.

4.1.8 TRITIUM HANDLING

The safety and environmental issues involved in tritium handling are of uppermost importance. Close attention to these issues dictated the design guidelines for tritium handling that were adopted at the outset of the conceptual design study and which have prevailed since then. These guidelines include:

- Minimizing site constraints, with assurance of virtually zero release for credible accidents.
- Assuring that the maximum potential tritium release will be within regulatory limits.
- Minimizing the tritium inventory on the site, consistent with safety and operational program requirements.

- Utilizing simple, reliable handling systems.
- Maximizing system hardware and work area containment and minimizing dilution of releases, within constraints of equipment accessibility and maintainability.
- Providing means for rapid facility cleanup and availability for both normal and emergency release situations.

Enroute toward selecting the design approach to be followed for tritium-handling-related systems, the options indicated in Table 4.1.8-1 were considered. The options that were deemed best suited and were selected for TFTR on the basis of compliance with the aforementioned safety and operations guidelines are indicated by the asterisk in Table 4.1.8-1.

TABLE 4.1.8-1 TRITIUM RELATED DESIGN OPTIONS

● Tritium Handling   ● Gaseous vs. solid tritide inventory* ● One-time use and trap* vs. recover-purify-recycle at site ● Containment   ● Complete device and equipment enclosure vs. selected equipment enclosures* ● Facility sealed tritium boundary* vs. open construction ● Closed cycle ventilation* vs. in-out ventilation during pulsing operations ● Emergency Cleanup of Accidental Releases   ● Trap and dispose* vs. recover-recycle ● General facility system vs. separate dedicated systems* ● Cleanup of Normal Releases   ● Trap and dispose* vs. recover-recycle ● General facility system vs. separate dedicated systems*

Implementation of Safety and Operational Guidelines

The following tabulations summarize in general, the ways in which the selected tritium handling approaches implement the safety and operational guidelines previously prescribed.

A. Solid State Storage of Tritium

- Tritium gas is evolved by thermal (575-625 K) dissociation of uranium tritide, at pressures not exceeding 4×10^4 Pa (300 torr).
- Spent tritium and other plasma hydrogenous gases are trapped on zirconium-aluminum getter cartridges, within canisters that also serve as part of the foreline ballast volume for the Torus Vacuum Pumping System.

B. Maximum Gaseous Tritium Inventory

- The maximum amount of gaseous tritium on hand is that required for one pulse $\sim 50 - 200$ Ci (< 20 mg) located either in the delivery system "measuring-volume" or the torus, and that filling the gas space of the tritium generator ~ 250 Ci (< 25 mg).

C. Reference Tritium Inventory

- 25,000 Ci (~ 2.6 g) of tritium is on-line at any time during operations with tritium. The on-line inventory refers to the sum, at any time, of all the tritium in the operational generator and getter assemblies. At the beginning of a fuel cycle, all of the on-line tritium will reside in the generator. As the experimental program progresses, the generator will be depleted and the spent tritium will accumulate in the getter assemblies.
- 50,000 Ci (~ 5.2 g or 19.4 STP liters) of tritium is the maximum on-site inventory, of which 25,000 Ci is in solidified form within a standby generator (not on-line). This spare generator, identical to the on-line generator, is housed within a sealed, thick-walled steel tritium generator enclosure which is also located in the fuel vault.
- Holding the on-line tritium inventory to no more than 25,000 Ci is assured by: (1) limiting the uranium content (~ 73 g) of the generators to that corresponding to chemical combination of 25,000 Ci, based on forming $UT_{2.8}$, (2) replacing the spent plasma getter cartridges simultaneously with replacing a depleted tritium generator with a fully charged generator.

- The basis for the 25,000 Ci generator capacity is that this limits both the tritium transportation and re-supply phase operations to reasonable, safe, operations and ensures experimental program flexibility.

D. Containment and Engineering Safeguards

- Multiple levels of containment are provided for the tritium inventory, along with appropriate engineered safeguards, such that no single failure or common-mode failure could result in release of a significant portion of the on-site tritium inventory to the environment.
- The tritium generator internals are doubly contained by a pair of thick-walled stainless-steel cylinders. Both walls are cooled by closed-cycle water flow through the annulus. This configuration: (1) minimizes tritium permeation through the walls, (2) confines the tritium permeated to a small volume of water.
- The tritium generators are provided with automatic power cutoff at upper limits of both temperature and pressure.
- The tritium generators are housed within sealed, thick-walled steel enclosures, in which an inert atmosphere (argon) is maintained. This inert atmosphere guards against uranium tritide reaction with air (oxidation, nitriding) in the event of leak development or rupture of generator.
- The tritium measuring and dispensing equipment is housed within an argon-inerted glove-box which is mated to a tritium generator enclosure.
- The common inert atmosphere of the mated tritium generator and dispensing system containments is continuously cleaned by a closed cycle atmosphere detritiating system that is located within the dispensing unit glove-box.
- Means are provided to valve off and evacuate the lines interconnecting the tritium system enclosures, and vacuum jackets are provided about these lines and the tritium supply conduit to the torus to afford additional protection against tritium releases and line damage.
- A third level of containment is provided by housing the tritium generating and dispensing equipment and their enclosures within a small doubly sealed fuel vault. This vault is maintained at subatmospheric pressure to prevent tritium outleakage and is provided with its own air detritiation system which can be operated either intermittently or continuously.

- Passage of particulate material into the torus via the injected tritium is prevented by three barriers — a palladium alloy diffuser and two stages of porous metal filters located in a series array in the tritium generation equipment train.
- Protection for the getter cartridges is afforded by housing them within stainless steel canisters of 2.5 cm (1 in.) wall thickness. This provides both mechanical security against accidental breaching of the walls and high impedance to tritium permeation.
- Gas holding tanks are provided for interim storage of non-hydrogens and non-getterables generated by the plasma discharges. Depending on the tritium content of these, as determined by monitoring, they can be discharged directly to the environment or first detritiated before such release.
- All of the susceptible operations areas are located within the facility "tritium boundary", and are in most instances both compartmented for isolation and provided with appropriately sized dedicated detritiating systems.
- The ventilation systems are designed with flexibility, permitting operation in a number of modes, adequate to cope with all foreseeable failures. This flexibility, in the case of the TFTR Test Cell, is indicated schematically in Figure 4.1.8-1. Changeover from one mode of operation to the other is accomplished by fast acting valves in response to commands dispatched from the central control.

E. Tritium Cleanup System

- The tritium cleanup system, comprising five separate installations, detritiates the atmospheres of facility areas where tritium is handled or released, down to levels below regulatory established maximum permissible concentrations ($5 \times 10^{-6} \mu \text{Ci/cm}^3$ for controlled areas). Two of these installations are intended to function primarily for facility emergency recovery operations, in the event of an accidental tritium release. The larger of these is designed to process the air of the TFTR Test Cell, but can also be called upon to detritiate the air of the Transfer Bays. The smaller of the emergency cleanup systems treats the air of the Hot Cell. The other three installations, which are relatively small, operate routinely on the atmosphere of the Tritium Storage and Delivery System containment (argon), fuel vault (air), torus and spent plasma gas residues (experimental gases and air).

EMERGENCY CELL CLEANUP OPERATION

NORMAL CELL SEALED OPERATION DURING CONDUCT OF PULSES

NORMAL CELL OPERATION DURING OUTAGES

Figure 4.1.8-1. Schematics of TFTR Test Cell Air Treatment Hookups

- Operation of the two emergency tritium cleanup systems is triggered by detection of a pre-set tritium activity level in the air of the cells or observation of substantial rate of vacuum loss in the torus.
- The torus cleanup system detritiates the torus (residual and absorbed gases) prior to opening the torus in preparation for disassembly-modification operations. This unit serves in a dual capacity. It is piped so that it can also operate on the non-hydrogens and non-getterables accumulated in the gas holding tanks. It may also be called into service in the event of a burn-through of the torus wall.
- Operation of the tritium cleanup systems for a detritiation campaign entails closed-loop recycling of the work area atmosphere being treated until monitoring indicates cleanup of this atmosphere to activity levels below maximum permissible concentration.
- The operating basis for all five cleanup systems is catalytic oxidation of gaseous tritium followed by trapping the tritiated water thereby generated on a chemical reacting type of desiccant. A chemical reacting type of desiccant is preferred over a physi-sorbing desiccant or cryo-absorption on the grounds of operational simplicity and safety. Chemically reacting desiccants do not release the trapped tritiated water unless heated to relatively high temperatures (~ 475 K) whereas physi-sorbing desiccants release it readily at ~ 333 K.
- Means are provided in some of the systems for enhanced decontamination by incorporating catalytic crackers to fragment tritiated non-getterables, and radiochemical carrier techniques such as hydrogen injection (TFTR cell and Hot Cell cleanup systems), and "moisture-swamping" (Tritium Cleanup System).
- No attempt is made to regenerate the desiccant. The spent desiccant is disposed of by off-site burial.
- The cleanup systems are sized so that detritiations subsequent to maximum possible releases (2-50 kCi) are effected in relatively short times (24 - 72 hours).

F. Conduct of Hazardous Operations Off-Site

- The tritium is to be obtained from an ERDA facility as uranium tritide, prepared by reacting high-purity depleted uranium metal with thermal diffusion purified tritium gas in situ in the previously mentioned tritium generators.

- The generators are to be shipped for initial charging and subsequent replenishment between TFTR and the ERDA facility, on a schedule geared to the TFTR needs. The generators and their associated auxiliaries (palladium alloy diffuser, valves, manifolds, instrument sensors, etc.) which are permanently affixed in sealed thick-walled stainless steel modules, are to be additionally housed and shipped as Type B Packages in approved Department of Transportation Specification Shipping Containers.
- These approaches: (1) minimize the handling of gaseous tritium as a consequence of obtaining the tritium in the solidified form required for ultimate use, (2) substantially reduce the risk of tritium releases during transport and connection to the dispensing system as a consequence of shipping the material in chemically combined solid form, (3) minimize both the on-site required gaseous tritium inventory and the chances of release of substantial amounts of tritium gas to the TFTR environs.
- The spent plasma gettering cartridges and their canister housings are likewise to be shipped to an ERDA facility for regenerative reclamation, or for disposal in the event of substantial irreversible loss of gettering capability. In this connection, the getter cartridges are equipped with integral resistance heaters which are used during initial thermal-vacuum activation and regeneration of the devices. Regeneration also affords an opportunity to recover the tritium, providing it is not diluted overly with hydrogen and deuterium. Such operations can be conducted conveniently at an ERDA facility that treats millions of curies of tritium routinely.

TFTR Cell and Fuel Vault Ventilation Considerations

Ventilation of the TFTR Test Cell is an integral part of the containment operations, but is treated here as a separate issue. The design and operating mode selected has been guided by the near-to-zero-release motivation.

During normal operations, the ventilation system for the TFTR cell is designed to function in the following modes, with transfer from one mode to the other through appropriate valve settings and blower activations achieved automatically:

- The cell is always maintained at subatmospheric pressure (~ 2.5 cm of water negative) so as to prevent out-leakage and to promote in-leakage, estimated as $0.0567 \text{ m}^3/\text{s}$ (120 cfm).
- During inoperative periods (no pulsing with tritium, or outages) the conditioned ($\sim 21^\circ\text{C}$, 40% absolute humidity, two stages of filtering) air flow in the cell is two changes per hour, on a once-through basis (middle mode shown in Figure 4.1.8-1) corresponding to $10.9 \text{ m}^3/\text{s}$ (23,000 cfm).

- During operative periods (pulsing with tritium) and during standby conditions subsequent to completing a pulse sequence, the Cell is sealed and the Cell air is recirculated within the Cell through the air cooling package to provide the requisite heat dissipation capability, at a rate equivalent to two Cell turnovers per hour (upper mode shown in Figure 4.1.8-1). During these periods the chiller condensate is directed to the monitored liquid-waste-retention tanks.
- During all operation modes an amount of air equivalent to the in-leakage is exhausted to the environment, estimated as $0.0567 \text{ m}^3/\text{s}$ (120 cfm).
- The duration of the sealed Cell mode for a post-operative period is determined by decay requirements of neutron-activation-generated nitrogen (^{13}N , 598 d, ^{16}N , 7.14 s) and Argon ^{41}Ar 6590 s) nuclides. Typically, once-through ventilation will not be resumed until completion of the last pulse of a tritium pulsing sequence.
- The pressure in the fuel vault is maintained at a slightly negative pressure with respect to ambient by tying it to the TFTR Test Cell through a double set of check valves which permit flow only from the vault to the cell. The vault normally is in a sealed condition, so that the flow involved is both very small and intermittent. The objective is to ensure that any leakages for the vault will always be inward from door seal leaks rather than outward. The vault is not otherwise ventilated.

In the event of an abnormal tritium discharge into the TFTR Test Cell, the air-handling system is designed so as to accomplish the following actions (see lower mode shown in Figure 4.1.8-1):

- The Test Cell is always maintained at subatmospheric pressure, except for unusual situations when both access doors may be open. If the Test Cell access sealed double doors were open at the time of the release, these doors would be closed rapidly to place the Test Cell in isolation with respect to the environment.
- The Test Cell Tritium Cleanup System would be activated and operated until monitoring of the Cell atmosphere indicates a reduction of the activity level below maximum permissible concentration.
- During the cleanup campaign the blower of the ventilation system would be operated in closed cycle mode at full capacity in order to promote cell air mixing.

- As in the case of normal operation, the maintenance of subatmospheric pressure in the Test Cell is accomplished through compensation of the in-leakage by bleeding a corresponding amount of cleaned air from the cleanup system to the facility exhaust duct via the 0.0567 m³/s (120 cfm) blower.

Table 4.1.8-2 lists the principal features and major design parameters for the systems involved in tritium handling. Additional details regarding design and operational parameters for the tritium storage and delivery and tritium cleanup systems are presented in tabular form in Sections 4.4.2 and 4.4.3.

4.1.9 ASSEMBLY AND REMOTE SERVICING

The Assembly and Remote Servicing System provides the handling equipment, special tools and software required to assemble the tokamak and ancillary equipment and to perform the servicing, handling and maintenance tasks necessary to keep the TFTR operational. The system will be capable of performing remote servicing and handling in a radiation field that is hostile to human operators as well as contact hands-on operations in low radiation fields. In both operations, remote or contact hands-on, the principal requirement is to perform the tasks safely with high reliability. The major design objectives are to maximize the use of existing technology, minimize equipment costs, and avoid unnecessary delays in TFTR operations.

The Assembly and Remote Servicing Systems control the work that is normally performed in the TFTR Test Cell, Hot Cell, Transfer Bays, Neutral Beam Test Cell, and Mock-up Assembly Building. Remote servicing operations will be possible prior to, and subsequent to, a TFTR experimental pulse. The system does not have the capability of performance during an actual D-T experimental pulse but is designed for use, in the TFTR Test Cell, 15 minutes after the completion of a D-T experimental pulse.

Servicing or maintenance operations in the TFTR basement area will be performed using contact and/or semi-remote methods, and normally will not be performed during the actual TFTR D-T experimental pulse.

The principal functions of the Assembly and Remote Servicing Systems are as follows:

- Initial assembly of the tokamak and ancillary equipment.
- Performance of preventive maintenance operations, such as cleaning, lubricating and tightening bolts.
- Performance of diagnostic operations to identify and locate malfunctions (e.g., inspection and testing of major assemblies).
- Adjusting equipment having adjustment features.

TABLE 4.1.8-2 TRITIUM SYSTEMS DESIGN SUMMARY

Design Basis Full Power D-T Pulses Per Year	1000
Design Basis Maximum Full Power D-T Pulses Per Day	96
Expected Full Power Tritium Pulses Per Day	40
Design Life, yr	4
Time Interval Between Full Power D-T Pulses, s	300
Tritium Injection Per Pulse, Ci	50 - 200
Reference On Line Tritium Inventory, Ci	25,000
Maximum On-Site Tritium Inventory, Ci	50,000
Tritium Hold Time for Experiments, sec	
Pre-plasma, Formation Burn	1
Post-pulse, Up to Pump-Down Initiation	5
Initiation of Cell Ventilation, After	Last Pulse of Day
Vacuum Pump Capability (Hydrogen), m ³ /sec	33
Pump-Down Base Pressure, Pa (torr)	$<8 \times 10^{-6}$ (6×10^{-8})
Adsorbed Layer Population Density, molecules/cm ²	8.5×10^{14}
Assumed Outgassing Rate, g mole/cm ² sec Pa m ³ /s m ²	5.9×10^{-15} (10^{-7})
Getter Cartridge Reversible Absorption Capability, g mole/cartridge (Pa m ³)	8.2×10^{-1} (1900)
Target Tritium Level for A Cleanup μ Ci/cm ³ (mpe)	5×10^{-6}
Maximum Potential Tritium Released, Ci	
Tritium Generator Enclosure	25,000
Fuel Vault	50,000
Torus	2,000
Hot Cell	2,000
Test Cell	2,000
Design Basis Detritiation System Cleanup Time, Hours	
Tritium Generator Enclosure	24
Fuel Vault	72
Torus	24
Hot Cell	48
Test Cell	48
Tritium Supply Form	Uranium Tritide
Getter Material	Zirconium-Aluminum
Tritium Cleanup	Catalytic Oxidation and Cracking, Desiccation

- Removal (and reinstallation) of major and minor components from the tokamak system.
- Disassembly, repair, and reassembly of costly components that are suitable for repair.
- Investigation of failure causes, using post-mortem techniques.
- Transportation of TFTR components between cells and storage areas of the facility.
- Performance of miscellaneous operations that can be performed more efficiently or more safely by using remote maintenance methods instead of conventional maintenance methods.

The Test Cell remote handling system is designed to service, maintain, inspect, repair and/or replace the various components of the tokamak neutral beam injectors and in-cell diagnostic instrumentation. An overhead bridge and trolley crane system (sized to lift an assembled neutral beam injector) is utilized to lift and remove components from the test assembly. An electro-mechanical manipulator system, designed to traverse and travel with the overhead bridge crane, is used to perform remote handling operations with the crane. The operations of the crane/manipulator configuration are designed to be controlled from a shielded cab or from a central maintenance control room with closed circuit television (CCTV) and direct, shielded visibility to the Test Cell.

The objective of the combined remote handling system is to achieve maximum remote servicing capability, close-in visual control via the shielded cab, observation with CCTV, central control from a maintenance center, as well as to minimize (reduce) the handling system equipment envelope. The smaller handling system envelope permits lowering the Test Cell roof, realizing a significant reduction in building costs. Remote-operating servicing equipment and tools are required to support major servicing operations, such as cutting and welding the vacuum vessel parting plane, as well as small, minor-bolt-removal operations. In general, the complexity of servicing techniques and the extent of remote handling are a direct function of the expected radiation fields in the local work area.

In developing the design requirements and concepts of the vacuum vessel cutting and welding equipment, the relatively high local radiation levels necessitate a fully remote operating system. The subject equipment is to be handled remotely, positioned adjacent to the vacuum vessel, and aligned for the required cutting/welding operation; the cutting or welding head is then inserted through a previously opened access port, adjustments are made to confirm the correct setting for the specific operations, and the equipment is started to complete the operation. The control of this operation will be mainly from the shielded cab/electro-mechanical manipulator system. If the local radiation field is lower than the expected levels, portions of the operation will be

assisted by the hands-on contact method. However, the basic requirement is that the system function satisfactorily in the fully remote operating mode. The criteria for all servicing/maintenance operations, including the previously discussed minor-bolt-removal operation, is to design the system for remote handling as related to expected, local radiation levels. If during the operation the radiation levels are lower than expected, assistance by the controlled hands-on contact method will be permitted.

The Test Cell handling system is designed to transfer an assembled Neutral Beam Injector from the Test Cell to a transport rail car located at the entrance of the Neutral Beam Test Cell. The transport car is designed with a turntable to facilitate rotating and placing the neutral beam injector in suitable positions for maintenance. The direct transfer of the Neutral Beam Injector, by crane, to the Neutral Beam Test Cell entrance reduces the handling and moving operations. The Neutral Beam Test Cell is sized to receive an assembled Neutral Beam Injector for maintenance and/or testing. The space above the Neutral Beam Test Cell is reserved for crane parking, and the space in front of the cell is utilized for manipulator/shielded cab parking. Radiation shadow shielding is designed into each parking area.

The transfer of a large component from the Test Cell to the Hot Cell requires the use of the Test Cell crane and a rail car, the Test Cell/Transfer Bay shielded sliding door is opened and the car is moved into the Transfer Bay. This operation joins the Transfer Bay controlled-air system. After the rail car/component is moved, the shielded door is closed and the Test Cell and Transfer Bay again become separate controlled-air systems. The Hot Cell shielded door is then open and the rail car moved into the Hot Cell. Controlled-air areas are established in each movement of the rail car and associated opening and closing of shielded, air-sealed doors. The control of potentially loose radioactive contamination is achieved by the opening of only one sealed door at a time. The door opened first must be closed prior to opening the second door. A similar system using smaller floor hatches can be utilized when small components are to be transported from the Test Cell to the Hot Cell. A conventional overhead crane, electro-mechanical overhead manipulators, shielded windows, and mechanical thru-wall manipulators provide the remote handling system inside the Hot Cell. The Hot Cell is sized to receive a half-section of the vacuum vessel.

The Mock-up/Assembly area is to be used for staging and pre-assembly of components prior to final assembly in the Test Cell. The rail transport system, used to move components from the Test Cell to the Hot Cell, will be used to move large components from the Mock-up/Assembly area to the Test Cell. The shielded doors will remain open during the assembly period. Cleanliness requirements will require temporary closures during specific assembly operations. In addition to the Mock-up/Assembly Area, the Neutral Beam Test Cell will be used as an assembly area for the neutral beam injectors.

4.1.10 POWER SUPPLY

The main pulsed loads of the TFTR are the toroidal field (TF) coil, the equilibrium field (EF) coil, the ohmic heating (OH) coil, and the neutral beam (NB). These loads fluctuate up to as much as 660 MW over the few seconds of the operating cycle, and are then off for nearly five minutes. A means of storing the energy at a low power level (~ 30 MW) for nearly five minutes and releasing it at a high rate (~ 660 MW) has been developed using small motors (16.5 K HP) driving large ac generators (330 MW) with large flywheels (~ 4500 MJ). The ac generators feed phase-controlled rectifier supplies, which convert and control the power to the large pulsed loads.

The power requirements are basically established by the physics and the mechanical parameters. For example, the largest load, the TF coil supply, is determined essentially by the required magnetic field, the coil major and minor radii, and the cross-section available for copper.

The power-supply motor-generator sets and converter equipment, which provide the energy to drive the TF, OH, and EF coils, will experience several different operating modes. The equipment design takes into account the fact that these modes are not fixed and immutable, but are part of an experimental program. To accomplish this, the power equipment is sized thermally for slightly more than the reference conditions, and will provide the versatility to operate satisfactorily at least down to 10% of the reference conditions of current and voltage.

The principal operating modes are:

Mode 1 - Plasma Experiment without Compression

Mode 2 - Plasma Experiment with "Weak" Compression

Mode 3 - Plasma Experiment with "Strong" Compression

Figures 4.1.10-1, 4.1.10-2 and 4.1.10-3 illustrate power demands typical of the corresponding operating modes. The absolute time and power values in these figures are representative values, not absolute parameters of a possible set of experimental values.

For each of the operating-mode cases, there will be four distinct phases:

1. Pre-start
2. Startup
3. Running
4. Shutdown

Figure 4.1.10-1. Load Sequence - No Compression. $\beta_{pol} = 0.7$

Figure 4.1.10-2. Load Sequence - Weak Compression. During Compression β_{pol} increases from 2.4 to 2.7

Figure 4.1.10-3. Load Sequence - Strong Compression
During Compression $\beta_p = 3$, and $n = 0.38$

The pre-start phase is the interval between plasma experiments. During this phase the generator will be brought up to full speed (~ 90 Hz) and the pulse-conversion rectifiers will be blocked by disabling the thyristor gate drives.

The startup phase consists of bringing the generator field excitation up to a value which provides 13.8 kV at the generator buses. Then the TF rectifier will be turned full-on, to bring the toroidal field value up to the desired level for the next experiment. As the TF approaches full level, the OH rectifier will be turned on, to bring up the OH current. The OH rectifier will be operated in a phased-back condition, to limit the rate of rise of the OH field. This phase draws approximately 80% of the flywheel energy but, with only two of the four main loads drawing power, it does not set the generator rating.

The running phase consists of operation to satisfy the needs of the experiment. This period lasts up to one second.

The shut-down phase will consist of ramping the OH and EF currents down rapidly, followed by reversing the TF rectifier voltage to bring the current to zero and recover part of the energy stored in the TF coils. All rectifiers will then be blocked.

Table 4.1.10-1 summarizes the generator load requirements with small margins provided to account for design variations in the coil systems. Based on this table, the reference design value of 660 MW for the generators allows a margin for experimental variations of 9 MW.

TABLE 4.1.10-1 SUMMARY GENERATOR LOAD LIST*

	Non-Coincident Power		Constant Rated Current		Coincident Power (Strong Compression)	
	Peak MW	Peak MVA	MW	MVA	Peak MW	Peak MVA
TF SYSTEM	501	611	380	459**	380	459*
EF SYSTEM	286	348	84	174***	264	348***
OH SYSTEM	70	85	45	85	7	34
NB SYSTEM	129	157	129	157	---	---
	986		638			

*TF, EF, and OH coil inductance 10% above nominal, TF coil resistance 5%, EF and OH resistance 10% above nominal.

**1/4 of TF rectifier bypassed.

***1/2 of EF rectifier bypassed when NB on.

In order to limit the generator voltage swing during pulsing, it is necessary to hold down the MVA load. This can be accomplished by bypassing up to 1/4 of the TF during the flat-top period and bypassing up to 1/2 of the EF when the neutral beams are on, if the EF current exceeds 40 kA during the time the NB load is on.

4.1.11 FACILITY

The TFTR Facility will be located at C Site, James Forrestal Campus, Princeton University on a site east of and contiguous to the existing Plasma Physics Laboratory complex. The buildings and structures will support the experiments planned for the TFTR.

Figure 4.1.11-1 shows the TFTR complex along with existing laboratory buildings and structures. Included in the TFTR facility are the following:

Experimental Area

The building contains the Test Cell, Neutral Beam Test Cell, Hot Cells, Transfer Bays, Mock-up and Assembly Bay, Observation Galleries, and basement areas for diagnostics, Tritium Vault, Tritium Cleanup, vacuum pumps, and utility services.

Control Room, Offices, and Mechanical Room

Located adjacent to the TFTR Building, this structure houses the Central Tokamak Control Room, Computer/Data Center, Rad-Safe Laboratory, offices, and mechanical equipment areas.

Motor-Generator Building

Two motor-generator sets, a local control room, emergency power, assembly/services area, and switchgear are contained in this building, which occupies a site about 45.7 m from the Test Cell.

Field Coil Power Conversion Building

This building is sited close to the Test Cell (24.4 m) and provides space for rectifier/transformer modules for the TF, EF, and OH Coil Systems, switching and control, and external capacitor yards.

Neutral Beam Power Conversion Building

Voltage modulators, transformers, and rectifiers for the Neutral Beam Injection System are located in this building which abuts the Main Test Cell and Neutral Beam Test Cell.

SITE PLAN

Figure 4.1.11-1. TFTR Facility Complex

Power Transformer Yard

The existing PPPL 138 kV switchyard will furnish space for transformers, breakers, switches, etc., to reduce the incoming 138 kV power to 13.8 kV for Motor-Generator set motors and to 4.16 kV for auxiliary power.

Facility Services

In addition to normal building services and site utilities, the facility will provide a Facility Cooling Water System, a Fire and Explosion Protection System, a Radioactive Waste Handling System, a Ventilation System, and a Liquid Nitrogen and Gas Storage System.

The objective in conceptual design of the facility was to meet functional requirements both within and among the buildings and structures at minimum cost. In reaching this objective, the first step entailed arriving at the most effective functional facility layout by preparing an inventory of space requirements showing for each space:

1. Function(s) performed,
2. Proximity to tokamak (a requirement),
3. Space requirement (length, width, and height), and
4. Most desirable location.

Table 4.1.11-1 presents the Inventory of TFTR Space Requirements. Two spaces not identified in previous Conceptual Design Reports are the Mock-Up and Assembly Bay, and the basement under the Test Cell, Neutral Beam Test Cell, and the Hot Cell. The first space will provide a facility for training operators, checking out remote maintenance procedures, and assembling TFTR components. The requirement for a basement evolved with better definition of equipment such as that associated with the Tritium Vault, Tritium Cleanup, Diagnostics and Vacuum Systems for the Tokamak and Neutral Beam Injection System.

With the above inventory, a Functional Layout for the facility was prepared which has gradually evolved to that shown in Figure 4.1.11 1.

Design requirements for TFTR facility buildings are summarized in Table 4.1.11-2. Table 4.1.11-3 lists design requirements for facility services for TFTR facility buildings.

TABLE 4.1.11-1 INVENTORY OF TFTR SPACE REQUIREMENTS

Facility & Space	Function(s) Performed	Proximity to TCT Device	Space Required	Location
1. Motor-Generator Bldg.	Source of Power supplies for: TF, EF, OH, and NB systems	N/A	33.5 m x 60.7 m in plan. 1 high bay, (15.2 m high) 2 low bays (7.6 m)	Close as possible to P. E. Bldg. and not more than 304.8 m cable run
a. MG Power Transformer Yard (outside)	Transforms incoming 138 kV to 13.8 kV for MG motors and to 4.16 kV for auxiliary power. Has associated breakers, switches and relays.	N/A	24.4 m x 24.4 m	In existing 138 kV switchyard at PPPL. Not more than 304.8 m of cable run to MG Bldg.
b. MG Control Room	Control operation of MG Bldg. equipment including: 3 MG sets Emergency diesel generator Uninterruptable Power Supply	N/A	6.1 m x 33.5 m x 3.0 m	Along one side of MG Bldg. (Mezzanine)
c. Diesel-Electric Standby Power	Provide emergency power in event of power loss from utility (PSE & G)	N/A	7.0 m x 13.1 m	Corner of MG Bldg.
d. Assembly and Service Area	Assemble and maintain MG sets	N/A	22.9 m x 25.9 m	Corner of MG Bldg.
e. Motor Control Switchgear	Control operation of MG flywheel sets	N/A	4.6 m x 38.1 m for switchgear; and 4.6 m x 38.1 m for generator field rectifier and motor rotor rectifier inverter	Along wall adjacent to MG sets

TABLE 4.1.11-1 INVENTORY OF TFTR SPACE REQUIREMENTS (Continued)

Facility & Space	Function(s) Performed	Proximity to TCT Device	Space Required	Location
f. Uninterruptable Power Supply and Battery	Maintain continuous AC power to essential loads	N/A	7.0 m x 9.1 m	Adjacent to emergency diesel generator
g. Generator output and load switch-gear.	On/off. Protect generator and load circuits against faults	N/A	11.0 m x 38.1 m (Reserved for present function plus allowance for future switch-gear)	Adjacent to MG sets
h. MG sets (2) plus space for a third	Provide power for TFTR pulsed loads	N/A	13.4 m x 38.1 m	MG Bldg
2. Field Coil Power Conversion Bldg	House power equipment for TF, EF, and OH Systems		30.5 m x 50.3 m x 7.6 m	Adjacent to Test Cell
a. FCPC Bldg Transformer Yard (Outside)	Provide space for following: <div> <div>System</div> <div>Equipment</div> </div> <div> <div>TF</div> <div>Rectifier Trans-formers (8)</div> </div> <div> <div>EF</div> <div>4 Rectifier Trans-formers</div> </div> <div> <div>OH</div> <div>2 Rectifier Trans-formers</div> </div>	N/A	<div>9.1 m x 50.3 m</div> <div>9.1 m x 24.4 m</div> <div>9.1 m x 12.2 m</div>	All Rectifier Trans-formers located along bldg. periphery Center-to-center distance to rectifiers is about 7.6 m
b. OH Capacitor Yard (outside)	Reverse OH Current	N/A	24.4 m x 24.4 m	As close as possible to OH Power Supply
c. EF Capacitor Yard (outside)	Compress the torus plasma	N/A	24.4 m x 24.4 m	As close as possible to EF Power Supply
d. TF Rectifier Modules (inside)	Convert AC to DC	Maximum 76.2 m bus length.	6.1 m x 48.8 m	Along one side of bldg. for full length

TABLE 4.1.11-1 INVENTORY OF TFTR SPACE REQUIREMENTS (Continued)

Facility & Space	Function(s) Performed	Proximity to TCT Device	Space Required	Location
e. EF Rectifier Modules (inside)	Convert AC to DC	Maximum 76.2 m bus length	6.1 m x 24.4 m	Along one side of Bldg.
f. OH Rectifier Modules (inside)	Convert AC to DC	Maximum 76.2 m bus length	6.1 m x 12.2 m	Along one end of Bldg.
g. Switching and Control	Switching and control of EF, TF, and OH systems	N/A	12.2 m x 45.7 m	Occupies the remaining bldg. area
3. Neutral Beam Power Conversion Bldg.	Support NB system: (1) Transform and rectify	Transformer/Rectifier in combined unit as close to Modulator as possible	60.0 m x 36.9 m x 8.5 m	Locate immediately adjacent to Test Cell
	(2) Modulate voltage	Require minimum possible cable run to device	19.8 m x 48.5 m x 8.5 m	Locate immediately adjacent to Test Cell
4. Control Bldg.	Provide Control Room for Test Cell, Computer/Data Center, Rad. Safety Lab., and offices	Control Room should be as close to Test Cell as possible	25.9 m x 38.1 m x 4.9 m	Adjacent to TFTR Bldg
5. Mechanical Equipment Spaces	House air conditioning and cooling water equipment	Close to Test Cell	Basement level mech. equip. 12.2 m x 15.2. Second floor equipment rooms 15.2 m x 38.1 m overall.	Adjacent to TFTR Bldg

TABLE 4.1.11-1 INVENTORY OF TFTR SPACE REQUIREMENTS (Continued)

Facility & Space	Function(s) Performed	Proximity to TCT Device	Space Required	Location
6. Experimental Area				
1) Test Cell	Permit assembly, test, minor maintenance and disassembly	Houses Tokamak	33.5 m x 36.6 m x 13.7 m	Center of exclusion perimeter
2) High bay area	Permit assembly, test, minor maintenance and disassembly	Houses Tokamak	33.5 m x 36.6 m x 13.7 m	Center of exclusion perimeter
3) Diagnostic Basement (Diagnostic & Equip.)	Provide shielded region for diagnostic equipment	4.6 m for probes	9.1 m x 9.8 m x 3.0 m high 7.0 m x 12.7 m x 3.0 m	Under Tokamak (2 areas)
4) Tritium Vault	Provide safe storage of tritium	Near	7.3 m x 11.0 m x 3.0 m high	Adjacent to Diagnostic Basement
5) Hot Cell	Permit maintenance of activated TFTR components	Near	13.1 m x 19.8 m x 9.1 m high	Access to Test Cell and Transfer Bays
6) Transfer Bays 1 and 2	Permit contact maintenance on cold or slightly activated components	Near	13.1 m x 18.3 m x 9.1 m 13.1 m x 13.1 m x 13.7 m	Access to Hot Cell N.B. Test Cell
7) Shielded Bay	Provide storage area for handling fixtures and prevent activation of crane and O-man	Adjacent	13.1 m x 36.6 m x 4.3 m high	Access to Test Cell
8) Galleries	Provide space for viewing interior of Test Cell	Adjacent to Test Cell	2 corner galleries 3.0 m x 4.3 m x 4.3 m x 8.5 m high (2 levels) triangular in section	In 2 corners of Test Cell
9) Maintenance	Provide viewing and control capabilities for maintenance operations in Transfer Bays, Hot and Test Cells	Adjacent	4.6 m x 13.8 m x 3.6 m (2 levels) 1.8 m x 13.3 m x 3.6 m (1 level)	Adjacent to Hot and Test Cells

TABLE 4.1.11-1 INVENTORY OF TFTR SPACE REQUIREMENTS (Continued)

Facility & Space	Function(s) Performed	Proximity to TCT Device	Space Required	Location
10) Neutral Beam Test Cell	Permit testing and maintenance on neutral beam injector	N/A	11.9 m x 20.7 m x 8.5 m high	Adjacent to Test Cell
11) Mock-Up and Assembly Bay	Permit checkout of special features of TFTR by trial assembly. Mock-ups would be used to aid in verifying remote maintenance procedures and training operating personnel.	N/A	29.9 m x 36.6 m x 14.9 m high	Adjacent to Hot and Warm Cells
12) Tritium Cleanup Equip. (basement)	House Tritium Cleanup System Equipment	As close as possible to Test Cell and Hot Cell	12.3 m x 15.2 m x 3.6 m high	Basement
13) Utility Space (basement)	Used to stage equipment and as air lock	N/A	14.0 m x 32.9 m x 3.0 m high	
14) Shielded Storage (basement)	Temporary storage for radioactive components/materials	N/A	3 areas: 4.6 m x 4.9 m x 3.0 m high 4.9 m x 7.0 m x 3.0 m high 7.9 m x 9.4 m x 3.0 m high	Beneath Hot Cell
15) Storage Area (basement)	Storage for slightly contaminated large components	N/A	8.2 m x 23.2 m x 3.0 m high	Under Hot Cell
16) Vacuum Pump Equipment (basement)	Space for vacuum pumps for torus and neutral beam evacuation	As close as possible	Annular area approx. 761.8 m ²	Beneath Test Cell

TABLE 4.1.11-2 DESIGN REQUIREMENTS, TFTR FACILITY BUILDINGS

Building	Controlled Atmosphere	Shielding Requirements	Structure Requirements	Utility Services	Other																														
Experimental Area	Tritium boundary required in Test Cell, NB Test Cell, Transfer Bays, Hot Cell and in basement	<table><thead><tr><th></th><th>Space</th><th>Thick m</th><th>Wall</th><th>Roof</th></tr></thead><tbody><tr><td>Test Cell</td><td>1.8</td><td>1.2</td><td></td><td></td></tr><tr><td>Transfer Bays</td><td>.6</td><td>.6</td><td></td><td></td></tr><tr><td>Test Cell</td><td>.6</td><td>.6</td><td></td><td></td></tr><tr><td>N. B.</td><td>1.2&1.8</td><td>1.2</td><td></td><td></td></tr><tr><td>Test Cell</td><td></td><td></td><td></td><td></td></tr></tbody></table>		Space	Thick m	Wall	Roof	Test Cell	1.8	1.2			Transfer Bays	.6	.6			Test Cell	.6	.6			N. B.	1.2&1.8	1.2			Test Cell					Reinforced concrete to support D. L., L. L., crane & TFTR device	Electrical, cooling water for coils, radioactive waste, ventilation	Air leakage at 2.54 cm H ₂ O diff. pressure not to exceed .057 m ³ /sec
	Space	Thick m	Wall	Roof																															
Test Cell	1.8	1.2																																	
Transfer Bays	.6	.6																																	
Test Cell	.6	.6																																	
N. B.	1.2&1.8	1.2																																	
Test Cell																																			
MG Building	N/A	N/A	Steel frame. Center span to carry 181 tonne bridge crane. 1 high bay, 2 low bays	Normal building services plus cooling water for MG sets and emergency generator	Generator pits 12.2 m diam. by 8.2 m deep																														
Field Coil Power Conversion Building	N/A	N/A	Steel frame designed for normal loads 1-story	Normal building services. Cooling water and environmental control for Rectifier Modules. 480 V, 3 ϕ required.	Length of bus from rectifier Modules - not more than 76.2 m																														
Neutral Beam	N/A	N/A	1-story steel frame with a portion for voltage modulation and remainder for transformer/rectifier	Normal building services. Cooling water for transformer/rectifier and voltage modulator areas.	Length of cable from 2 types of modules - should not exceed 30.5 m																														
Control and Mechanical Equipment Building	N/A	N/A	3-story steel frame. Mechanical spaces on 2 lower levels and Control Room and offices on 3rd level	Normal building services, cooling water for chillers	Special temperature humidity control in computer space																														

TABLE 4.1.11-3 DESIGN REQUIREMENTS, FACILITY SERVICES

System	General Requirements	Design Life	Principal Features
Fire and Explosion Protection	<p>Functions in Experimental Area to detect and initiate corrective action on:</p> <ul style="list-style-type: none"> • Temp >395 K • Temp rises >10 K/sec for 10 sec. • Dangerous concentrations of Hydrogen • Over pressure >.021 MPa 	TBD	<p>Principal elements:</p> <ol style="list-style-type: none"> 1. Detectors and alarms 2. Fire fighting equipment 3. Control of ventilation and electrical power.
Experimental Area Ventilation	<ul style="list-style-type: none"> • Provide 2 changes of air/hr (once through) during non-operational periods and re-circulated for operational • Remove 97-99% particulate • Direct 70-90% of air past torus for cooling and sweeping tritium • Maintain cell at 2.54 cm differential pressure • Provide 293 K, 40% RH air for test cell with ambient air at 308 K, 90% RH (summer), and -261 K, 10% RH (winter) 	TBD	<p>Major components:</p> <ol style="list-style-type: none"> 1. Intake louvers and screens 2. Exhaust blower with flow adjusters/damper 3. Exhaust stack 4. Tritium monitor/control system 5. Air Conditioning Unit 6. Absolute filters

TABLE 4.1.11-3 DESIGN REQUIREMENTS, FACILITY SERVICES (Continued)

System	General Requirements	Design Life	Principal Features
Facility Cooling Water	<ul style="list-style-type: none"> ● Absorb 23.4 MW from TFTR facility ● Makeup water of 0.016 m³/sec. 	15 years	Other Major Components: <ol style="list-style-type: none"> 1. 1.01 m³/sec pumps 2. 8.95 x 10⁴ W capacity motors at .52 MPa delivery pressure 3. Cooling towers rated at 44 MW
Radioactive Waste Handling	<ul style="list-style-type: none"> ● Collection depot for solid waste in isolated, controlled area ● Liquid waste handling system with holding tank capacity equal to volume of cooling systems for TF, OH, and EF coils and Test Cell fire system water ● Monitoring system to determine radioactivity levels 	TBD	Essential Features: <ol style="list-style-type: none"> 1. Storage tanks 2. Pumps/piping 3. Demineralizer 4. Filter 5. Holding pond
Liquid Nitrogen and Gas Storage	<ul style="list-style-type: none"> ● High pressure system for gas storage. Use valves, piping/fittings for 17.2 MPa pressure. 	15 years	Essential Features: <ol style="list-style-type: none"> 1. Liquid N₂ recirculator and 37.8 m³ dewar. 2. Gas farm provide high pressure cylinder capable of recharge from mobile truck tube suppliers for nitrogen, argon, helium, hydrogen.

4.1.12 DIAGNOSTICS

All information about the plasma behavior, and hence about the device performance, is obtained through diagnostics observing various properties of the plasma. This information is used to evaluate the behavior, to provide data for establishing new operating conditions, and to provide control signals for the operation of the device. A list of the proposed diagnostics with an abbreviated statement of their purpose is given in Table 4.1.12-1. All diagnostic apparatus will be available on the device for the first hydrogen discharge and all should be able to operate during a series of full-power D-T discharges.

Line-of-sight access to the plasma is required from both horizontal and vertical directions so that spatial variations in plasma parameters can be measured. Shielding is required to enable most of the diagnostic equipment to operate in the presence of the nuclear radiation generated during D-T operations. Local shielding can be used around equipment that is located on the Test Cell floor. There is a thick-ceiling basement area under the Test Cell to provide good shielding for diagnostic systems that use vertical access to the plasma and for components that do not have to be on the Test Cell floor.

The considerations in the Conceptual Design with regard to diagnostics are as follows:

- Diagnostic equipment must be capable of operating in and surviving the nuclear-radiation environment created by D-T pulsing. The activation level of this equipment over the life of the device must not be sufficient to prevent "hands-on" maintenance, except close to the vacuum vessel. This requirement leads to local shielding of equipment located in the Test Cell. Hence, the Test Cell floor must be capable of supporting about 0.6 m of shielding material enclosing this equipment.
- There must be sufficient shielding between the plasma and a basement diagnostic area to permit access to this area between pulses. The activation level has to allow "hands-on" maintenance. The basement will be within the tritium enclosure.
- The diagnostic equipment must be shielded against stray magnetic fields, as required to permit satisfactory operation. In turn, such magnetic shielding must not affect the field back at the plasma significantly.
- There must be access to the plasma along sight lines near the horizontal symmetry plane. The corresponding horizontal access ports should be large, so that each port can be shared by several diagnostics or heating systems and so that the plasma can be viewed over a wide solid angle. There should be as many large horizontal access ports as possible, so as to maximize access to the plasma. All these ports should be of the same size.

TABLE 4.1.12-1 PROPOSED TFTR DIAGNOSTIC SYSTEMS

Diagnostic Name	Brief Description of Purpose
Voltage Loops	Voltage around toroid - ohmic power, impedance
Rogowski Loops	Total ohmic current and its time rate of change - ohmic power, impedance
B_p Loops	Current channel position, gross fluctuations
Diamagnetic Loops	Plasma Pressure
4 mm μ -wave Interferometer	Check effectiveness of initial breakdown (for control system)
2 mm μ -wave Interferometer	Check sufficient plasma density prior to beam injection (for control system)
Visible Spectrometers	Impurity content close to plasma surface
U. V. Spectrometers	Impurity content, power radiated
HCN Laser (or CH_3F laser or 1 mm μ -wave) Multichannel Interferometer	Radial, temporal behavior of electron density
Thomson Scattering	Spatial variation of electron temperature and density at several times
Neutron Activation Detectors	Neutron fluence from D-T reactions - symmetry and total yield
Epithermal Neutron Detectors	Real time neutron flux in Test Cell
Neutron Spectroscopy	Ion temperature, beam thermalization
X-ray Pulse Height Analysis	Electron temperature, electron velocity distribution, impurity concentration
X-ray Crystal Monochromator	Impurity concentration, power radiated
X-ray Wave Detectors	Look at location of internal plasma fluctuations, $q(r)$
X-ray Position	Look at Bremsstrahlung - determine position of hot plasma
Hard X-ray Detectors	Examine runaway electron effects
Very Soft X-ray Detectors	Power radiated in 100 eV - 1 keV range
TI Beam (CO_2 laser or H_2 Beam)	Current density distribution, $B_p(r)$, $q(r)$
Charge Exchange Neutral Beam (or CO_2 laser)	Measurement of ion temperature
Surface Physics Analysis Station	Examine wall surface effects
Plasma TV	View inside vacuum vessel during pulse
4 mm Microwave Scatter	Fluctuations in outer regions of plasma
Synchrotron Radiation	Measure thermal electron cyclotron emission
Fast Ion Gauge	Measure pressure in vessel

- The longest straight rigid extension from any horizontal port will be a 15-cm diameter pipe with its flanged face 12.2 m from the minor axis of the vacuum vessel. This vacuum pipe is for entry of a laser beam for Thomson scattering measurements.
- There must be access to the plasma along vertical lines of sight. Some 15-cm and 23-cm diameter ports are required on the vertical center line of the vacuum vessel. About eight planar arrays of 10 to 12 vertical line-of-sight access tubes (about 7 cm diameter), extending above the device to beyond the EF and OH coils and below the device to the basement, will be necessary. The arrays above and below the vessel must line up, so that the plasma can be viewed unobstructed from the basement.
- Up to 6 of these planar arrays require that the access tubes are evacuated. These 6 arrays should be located on the fixed half of the device.
- There must be access above and below the device, so that diagnostic equipment can be connected to the vacuum vessel and information gotten to sensor heads. Interferences that have been considered and modified involve the location of EF and OH coils, the vacuum vessel support structure, the TF coil frame structure, the device support frame (and rails) and shielding.
- There must be clear access for diagnostic beams through all device components above the vessel minor axis at at least two locations. Horizontal ports must also be available at these locations.
- All diagnostic interfaces at the vacuum vessel must be designed to allow movement of the vessel through the TF coils. All vacuum seals must conform to the high vacuum standards for the vessel and must be capable of being made up remotely.
- Data handling capability must be provided to acquire and process the output from the diagnostic sensors.
- There must be provision for safe transmission of data from diagnostic sensors to the control room and computer, so as to protect personnel from any high voltages that may appear on the diagnostic equipment as a result of device malfunction.

Some typical diagnostic ports are shown in Figure 4.2.1-1. A more detailed discussion of how the various diagnostic requirements are met for various diagnostic systems is given in Section 4.6 of this volume.

4.1.13 NEUTRAL BEAM INJECTION SYSTEM

During specified time intervals the Neutral Beam Injection System must provide high-current, low-divergence beams of high-energy neutral deuterium (or hydrogen) atoms. Such beams are generated by the acceleration of ions, which are subsequently neutralized by charge-exchange reactions with neutral gas atoms in a neutralizing cell. Any ions which may remain in the beam are then removed by magnetic separators, while the remaining neutral particles enter the plasma without deflection. Ultimately, they become ionized and are trapped in the torus magnetic field. In the early portion of the pulse, the function of the trapped ions is to provide energy to raise the temperature of the plasma by sharing their kinetic energy with the plasma particles. In later portions of the pulse, their main function is to maintain the temperature of the plasma and also to undergo a small number of fusion reactions that yield approximately as much energy as the energy injected into the plasma.

The initial designs for the Neutral Beam Injection System were produced by Westinghouse and PPPL, but as they were developed, ORHNL and LBL/LLL became more involved. A series of workshop meetings between PPPL, ERDA, LBL/LLL, ORHNL and Westinghouse has resulted in agreements being reached on the major parameters of the system. The final Conceptual Designs have been produced separately by LBL/LLL and ORHNL, and these are included in Volumes IV and V of this report. Initially there were 8 injection arms each with 3 sources. As a consequence of a reduction in the plasma power requirements, plans now call for 4 arms with a total of 12 ion sources, and provision for optional future installation of two additional arms, each containing 3 sources. The various operational modes of the neutral beam injectors initially identified have been combined into a single mode. This requires 20 MW of 120 keV deuterium atoms, which are accompanied by 12-15 MW of lower energy components, to be injected into the plasma. To do this, each ion source must supply an ion current of 56A. In the initial designs, the neutral beam injection arms were mounted in orthogonal pairs around the device. However, in an effort to reduce the Test Cell size and to minimize the length of cable from the power supplies, the arms were all moved to one side. In this new arrangement, they are mounted individually with half of them injecting in the direction of current flow and the other half injecting in the opposite direction.

A number of internal arrangements of the injection arms were considered during the conceptual design phase. To keep the gas flow into the torus below $0.13 \text{ Pa m}^3/\text{pulse}$, which is the maximum allowed to dilute the plasma, there must be either a number of low conductance sections separated by pumping stages, or a high-conductance duct containing very-high-speed pumps. Because of beam losses on the duct walls and limitations in overall length, the latter approach was selected for both designs.

A valve is required to isolate the injection arm from the torus, and also open and close quickly to limit tritium diffusion into the injection arm. In initial studies, a single valve was proposed to fulfill both of these functions. However, in the current designs the fast valve is separate from the hard-seal valve. The latter can now take some seconds to

operate, and need be closed only infrequently. During the pulse, when the fast valve is open, some tritium will diffuse back into the beam line. Because of this, during tritium operation, the vacuum system must be closed to prevent the escape of tritium into the atmosphere. This can be accomplished by the use of getters and a cleanup system similar to that used in the Torus Vacuum Pumping System.

To increase the neutralization efficiency, a "Race-Track" to direct un-neutralized ions back through the neutralizer cell was considered. However, this approach was judged to be too speculative, and it was discontinued in favor of deflecting the un-neutralized ions into a beam dump.

The Neutral Beam Injection System must operate under a number of constraints imposed by the device. A neutron fluence of $8 \times 10^{16} \text{ n cm}^{-2}$ and a γ -ray dose of $3 \times 10^7 \text{ rads (C)}$ exists at the torus wall and this requires that all components of the beam line be selected to withstand some irradiation without degradation. The magnitude of the irradiation will decrease along the injection arm toward the ion source. Neutron irradiation causes activation of the various components of the injection arm. Because of this, remote handling techniques must be used for the servicing and replacing of these components. The magnetic fields in the region of the injection arm decrease with distance from the torus. In the neutralizer and source regions, the field is in the range 0.01 T to 0.03 T and these components must be magnetically shielded to prevent deflection of the ion beam.

Pains have been taken to provide mechanical access to the torus to allow tangential injection of the neutral beams. The geometrical limitations on the size of the injection duct are illustrated in Figure 4.1.13-1. In the vertical direction, the opening is limited to $\sim 70 \text{ cm}$ by the EF, OH coils assembly. In the horizontal direction, for injection at any specific radius, the duct size is limited by the TF coils. For the current aiming radii this dimension is limited to $\sim 40 \text{ cm}$. A band of armor plates is provided on the walls of the torus vessel to protect it from the unattenuated portion of the beam, which passes through the plasma.

Both the ORHNL and LBL/LLL designs use an array of three ion sources connected to separate neutralizer ducts. The un-neutralized ions are deflected by a magnet into a beam dump. In the ORHNL approach, water-cooled copper swirl tubes form the dump. LBL/LLL favor a serrated molybdenum slab to act as a beam dump. In both cases, the pumping for the injectors is supplied by large arrays of helium-cooled cryopanel, which are protected from direct radiation by LN_2 cooled chevrons. These panels can be heated up and unloaded at regular intervals, with the gas being removed by a pumping system which is similar to that of the Torus Vacuum Pumping System. The refrigerators for the helium and nitrogen systems will be located outside of the Test Cell and connected to the device with vacuum insulated lines.

Calorimeters of a design similar to that of the beam dumps are located just ahead of the fast valve. The calorimeters are removed before the fast valves are opened and a neutral beam pulse is injected into the plasma. The ORHNL valve is in the form of an aperture

Figure 4.1.13-1. Neutral Beam Injector Duct

in a rolled-up sheet, which moves across the duct. The opening in the LBL/LLL valve is formed by two plates which are operated by hydraulic actuators. The final component in both designs is a hard valve having an opening 0.9 m in diameter, which is attached to the plenum that leads to the torus.

The length of the injection arm is 9 m, the height 6.5 m, and the width between 2 and 4 m. This governs the size of the Test Cell and of the Neutral Beam Test Cell, which is used to maintain the injectors and to pre-condition the ion sources. To limit the length of the power supply cables, the ion sources on the injection arms must be located close to the wall between the Test Cell and the power supply area. In addition to these power supply cables, extensive instrument and control circuits are required between the injection arms and the control room.

4.1.14 CONTROL SYSTEMS

Normal operating procedure will be to maintain the facility in a "standby" condition, in which control of the TFTR facility will be centralized. A small number of operators will be able to operate the controls to carry out the experiments and return the facility to standby. This operating philosophy will require that control and monitoring equipment be provided in the centralized control station to perform necessary experimental functions.

Control and monitoring of the TFTR facility is required for the following operating conditions:

- Post-installation checkout
- Initial startup
- Normal experimental operations
- Standby conditions between experimental operations
- Maintenance and testing of facility equipment and systems including remote maintenance
- Transient and emergency operations

Installation and startup of the TFTR systems and components will require coordination of a large amount of equipment. Control modes and operating parameters will be selected by the operator(s) at the central control consoles by means of keyboard entries and console switches. During normal operation, the equipment will be monitored continuously. The control and critical monitoring functions will be performed automatically by the Central Control and Monitoring System (CCMS), providing panel readout displays of operating parameters as selected by the operator(s). Shutdown operations are performed by the CCMS with appropriate operator inputs similar to that used for startup. Local stations will be provided for test equipment and maintenance. This combination of central and local control will provide both normal testing control and extended monitoring capabilities for testing hardware. It will be necessary to monitor and record over one-half million words of data.

A high-performance automatic control system is, therefore, required to assure flexible, reliable and repeatable performance.

Design Considerations

The Central Instrumentation and Control System (CICS) provides central control and monitoring, safety and protection, data handling and display, and radiation and tritium monitors required for TFTR operation. Specific functions include:

- Central control, timing, and monitoring to enable remote control from a central control room of all TFTR systems, and to enable preprogramming of precision timed events for each TFTR pulse operation.
- Safety and protection, including a full complement of hard-wired function and area interlocks, key switches and alarms as required for personnel safety and equipment protection.
- Data handling and display for use by the operators and experimenters.
- Radiation monitors, including gamma, neutron, tritium and effluent monitoring, with measurements, indication, recording and alarms.
- Control and monitoring for operation of the gas, vacuum and neutral-beam injection systems from the central control room.

Central Monitoring System

- The Central Control and Monitoring System (CCMS) verifies operational readiness of all systems before providing the VERIFY READY STATUS permissive signal.
- The CCMS provides a control operating station, a supervisory and annunciator station, and plasma experimenter stations, all located in the central control area.
- The CCMS generates the required timing, as typically defined in Table 4.1.14-1, to synchronize this total operation of the TFTR.

Safety and Protection System

- The safety and protection system (SPS) will be hard-wired and will be designed to minimize situations which are hazardous to personnel or equipment, to monitor the conditions which exist, and in the case of undesirable conditions, cause the equipment to revert to a safe condition and sound an alarm.

TABLE 4.1.14-1 CENTRAL CONTROL, TIMING AND MONITOR REQUIREMENTS

Central Control and Monitor Function	Control Mode	Timing Requirements		
		Nominal Set Time (Time Range Referenced at T_R)	Resolution (Precision of Timing Signal)	Operational Data System Time Response
Verify Ready Status [NB Injection System, Vacuum System, Gas Systems, Elect. Pwr. Supplies]	Man. & Auto	T_R - 13 sec	—	0.25 sec
Initiate TF Current Buildup	Auto	T_R - 12 sec	0.1 sec	—
Initiate OH Current Increase	Auto	T_R - 3 sec	0.05 sec	—
Verify TF Current at Demand Value	Auto	T_R - 0.40 sec	—	0.05 sec
Verify OH Current at Demand Value	Auto	T_R - 0.40 sec	—	0.05 sec
Open Valve Command to Gas Supply	Auto	T_R - 0.35 sec	0.5 msec	—
Multipole Null Field Current ON Command	Auto	T_R - 0.25 sec	0.1 msec	—
Verify Gas in Vacuum Vessel	Auto	T_R - 55 msec	—	5 msec
Verify Multipole Null Field Current at Demand Value	Auto	T_R - 15 msec	—	2.5 msec
Breakdown Oscillator ON Command	Auto	T_R - 10 msec	0.1 msec	—
Verify Osc. Pwr. Supply Output	Auto	T_R - 7.5 msec	—	2.5 msec

TABLE 4.1.14-1 CENTRAL CONTROL, TIMING AND MONITOR REQUIREMENTS (Continued)

Central Control Function	Control Mode	Timing Requirements		
		Nominal Set Time (Time Range Referenced at T_R)	Resolution (Precision of Timing Signal)	Operational Data System Time Response
Verify Preionization Density	Auto	$T_R - 4.5 \text{ msec}$	—	1 msec
Initiate OH & EF Power Supplies to Begin Action to Change Current for Plasma Discharge Initiation	Auto	$T_R - 3 \text{ msec}$	10 μsec	—
Detect Plasma Current Rise to Trigger EF Current Program Demand	Auto	T_R Reference Time	—	0.5 msec
Verify Plasma Density	Auto	$T_R + 25 \text{ msec}$	—	2.5 msec
ON Command to NB Injectors - Preheat	Auto	$T_R + 46 \text{ msec}$	0.01 msec	—
ON Command to NB Injectors - Fusion Heating	Auto	$T_R + 89 \text{ msec}$	0.01 msec	—
ON Command to NB Injectors - MODE II (if used)	Auto	$T_R + 145 \text{ msec}$	0.01 msec	—
Activate Feedback Control Plasma Horiz. Position	Auto	$T_R + 50 \text{ msec}$	0.01 msec	—
Activate Feedback Control Plasma Vert. Position	Auto	$T_R + 50 \text{ msec}$	0.01 msec	—
Activate EF Circuit Compression Mode (if used)	Auto	$T_R + 100 \text{ msec}$	0.01 msec	—
Start Shutdown	Auto	$T_R + 1 \text{ sec}$	1 msec	—

- The SPS provides interlocks for all electrical systems and for access to all electrical systems that can create hazardous conditions. A key exchange interlock system will be provided.
- The SPS provides annunciators to indicate the presence of a fault in a sub-system. Means are provided in the central control room, in the power equipment building, motor generator building and in the test cell and other strategic locations for emergency disable and/or shutdown of the equipment.

Data Handling System Design Requirements

Data Acquisition and Storage System

- The Data Acquisition and Storage System (DAS) provides data collection during the course of the experiment and stores it for later analysis and display.
- The DAS includes standard minicomputers, complete with interfaces to the main data-reduction computer.
- The DAS provides the capability to sample and hold simultaneously 100 analog input channels, with a maximum throughput of 50,000 samples/second on any one channel, and 500,000 samples/second total maximum sampling rate.

Data Reduction and Display System

- The Data Reduction and Display System (DRDS) is provided to analyze and display data obtained from the experiment in a time frame suitable for determining the results of one pulse prior to the next pulse.
- The DRDS provides multi-user, time-shared computer facility for program preparation and background processing.
- The DRDS provides interactive analysis and design facilities and gives real-time guidance in conducting experimental runs.
- The DRDS provides temporary and archival storage for all data collected.
- The CPU features a computer system architecture that maximizes overall throughput speed under the conditions of the TFTR operating environment. This throughput speed is measured by means of benchmark tests that include typical program functions written for data acquisition/reduction and real-time monitors.

- The DRDS includes Peripheral Equipment to be supplied with the computer system and is completely software supported by the monitor and is flexible with respect to expansion of the system. Each peripheral device is fully interfaced to the computer, including the buffering necessary to accomplish high-speed data transfers to and from a peripheral. In addition, each peripheral device is interfaced to function under interrupt control.

Major peripheral devices include the following:

Console Keyboard-Printer

Magnetic-Tape Equipment

Moving-Head Disks and Controller

Line Printer

Card Reader

Swapping Disk System

Incremental X-Y Plotter

Graphic Display System

Memory Port Multiplexer

Line Scanner

Memory Link Interface Units

- The DRDS includes six (6) computer display terminals, Tektronix Model 4012 or equivalent. Each display terminal is mated to a hard copy unit, Tektronix Model 4610 or equivalent.

Radiation and Tritium Monitor Design Requirements

- Tritium monitoring devices will be of the continuous reading type, with recording and output signals if T_2 level exceeds a predetermined set point.
- Tritium monitors will be situated at specified locations which include: tritium storage and generation areas, test cell, exhaust duct, tritium cleanup equipment area, exhaust lines of dessicant unit of cleanup systems, exhaust from vacuum system (backup) mechanical pump, at inlet of torus tritium cleanup system, diagnostic equipment area, and test cell ventilation equipment area.
- Tritium monitors must detect tritium concentration in the range from 2×10^{-6} to 10^{-2} μ Ci/cm³ of air.

- Tritium monitors will have an adjustable set point in the range of 10^{-5} to $10^{-3} \mu\text{Ci}/\text{cm}^3$. When tripped, a signal will put the TFTR ventilation system into the recirculation mode and indicate the condition in the control room and safety office. The alarm will be manually reset from the control room.
- Tritium monitoring devices must differentiate between tritium activity and argon-41 activity when the latter concentration is as high as $30 \times 10^{-6} \mu\text{Ci}/\text{cm}^3$.
- Tritium monitors must function in a uniform external gamma radiation field as large as 1 mr/hr. Local shielding will be provided to limit radiation to this level.
- Provisions will be made for test and calibration of sensors at their installed positions.
- Beta, neutron, and gamma radiation for the TFTR facility will be monitored and recorded on a continuous basis.
- Continuous air particulate monitoring will be provided for all working areas, test cell, and at points of exhaust.

4.1.15 SYSTEM STUDIES

Many system or tradeoff analyses were performed as part of the conceptual design phase. A significant activity was the initiation of a system analysis model TCTSAM. Even though this model is far from complete, some useful information is already available. Other tradeoff studies include: locating the chillers with respect to the ballast tank, choosing the number of cooling turns per circuit for the TF Coil, and minimizing the cost of the electric power system and the MG building. The following subsections provide a brief summary of these studies.

Systems Study of Motor-Generator (MG) Set Loading

The Tokamak systems analysis model, TCTSAM was used in a parametric study to determine how the energy and power requirements on the MG sets, and the coil cooling requirements on the chiller system, vary with system parameters. The objective of this systems study, at this time, was to identify trends and evaluate alternates. The formulation has progressed to the weak compression case which is presented herein but is not the limiting case. Section 4.1.10 contains the latest information.

The baseline case was a 1-second flat-top, 2.5 MA plasma experiment, with nominal resistances, inductances and supply voltages. The TF current buildup was started at -11.5 seconds in order to just reach the nominal 73.2 kA current at time 0. The OH

current buildup was started at -3.0 seconds, with a constant rate of change such that the maximum 40 kA current would be reached at time 0. At time 0 the OH supply voltage was removed and the capacitor bank switched in, causing the OH current to decrease rapidly and reverse. The neutral beam power was turned on at +.3 and off at +.8 seconds. At the end of the flat-top period, +1.0 seconds, the TF supply voltage was reversed to -70% of nominal, and the OH and EF supply voltages were removed. The OH current decayed to zero through the capacitor bank in less than 0.2 seconds and the TF current decayed to zero in 5.5 seconds. From this time on there was no input in the coils but heat transfer to the coolant system is calculated for the remainder of the 300 second cycle. The electric motor speeds up the MG flywheel system back to the required speed. Table 4.1.15-1 shows the power loads predicted by TCTSAM. The peak powers all occur during or just before the flat-top period. The buildup of the TF field results in a peak power of 516 MW at time zero. The EF power goes through a rapid increase in the period in which the OH system is reversing through its capacitor bank. This results in a peak of 490 MW, with this being quite sensitive to the particular experiment.

TABLE 4.1.15-1. TFTR ELECTRIC POWER LOADS (MW) (FOR A ONE-SECOND FLAT-TOP, 2.5 MA PLASMA EXPERIMENT)

Time from Start of Flat-Top (sec)							
	-0	+0	0 to .13	.15	.3	.8	1
TF Power	453	361	361	361	361	361	361
OH Power	63	0	0	17	29	55	63
EF Power	0	0	129	57	51	30	23
NB Power	0	0	0	0	126	126	0
Total Power	516	361	490	445	567	572	447

The largest peak occurs while the neutral beam is on, and is calculated at 572 MW. For these calculations the TCTSAM model used nominal values of coil resistances, inductances, and supply voltages, and included both dc and ac power losses. The hybrid concept which has the OH and EF systems sharing coils was modeled and resulted in lowering the peak power by about 10 MW. These power requirements affect sizing of the generators. The integrated powers affect sizing of the motors and the inertias of the MG-flywheel systems.

Table 4.1.15-2 shows the energy requirements for the nominal parameters. The peak drain from the flywheel is 3832 MJ. These energy requirements change somewhat with system parameters. Table 4.1.15-3 shows the magnitude of the energy changes for

several parameter changes. The maximum energy requirement would increase some 78 MJ over the 3832 MJ if all coils had high resistance and inductance. The comparison of TF current buildup as rapidly as possible (the nominal case) vs. a conceived reference 12-second buildup shows clearly why the nominal procedure is used. Some 262 MJ to 499 MJ extra are used for the 13-second buildup depending on whether nominal or maximum TF parameters are used.

TABLE 4.1.15-2 TFTR ELECTRIC ENERGY LOADS (MJ)

	Up to Start of Flat-top (-11.5 to 0 sec)	To End of Flat-top (-11.5 to +1.)	To End of Current Decays (-11.5 to +7.)
TF Power Supply	3221	3585	2870
OH Power Supply	93	130	132
EF Power Supply	0	52	59
NB Power Supply	0	65	65
Total Power Supply	3314	3832	3126

**TABLE 4.1.15-3 TFTR ELECTRIC ENERGY LOAD VARIATIONS (MJ)
(ADDITIONS AS COMPARED TO NOMINAL)**

	Up to Start of Flat-top (-11.5 to 0)	To End of Flat-top (-11.5 to 1)	To End of Current Decays (-11.5 to 7.)
Maximum Resistance, Inductance, and Supply Voltage			
In TF Coil	54.	70.	-18.
In OH Coil	9.	5.	1.
In EF Coil	—	3.	2.
In All Three	63.	78.	-15.
13-second TF buildup, Nominal parameters	+262.		
Maximum parameters	+499.		

Cooling System Performance Studies

The effects of the ballast tank size, test cycle length, and flat-top time on the system temperatures and flywheel speed were studied using the systems analysis model TCTSAM. Some effects of changing the ballast tank volume are shown in Table 4.1.15-4. Decreasing the tank volume results in higher tank temperatures during the cycle but lower at the end of the cycle and small effect on the TF coil temperatures.

**TABLE 4.1.15-4 SYSTEM TEMPERATURES VS BALLAST TANK VOLUME
(1 SEC. FLAT-TOP, 300 SECOND CYCLE)**

<u>Ballast Tank Volume (m³)</u>	10.6	21.0	41.8
<u>Temperature (K)</u>			
Average TF Copper Temp.			
Maximum during cycle	302.3	302.0	301.7
At end of cycle	284.8	284.5	284.5
Ballast Tank Water Temp.			
Maximum during cycle	298.3	295.6	294.2
At end of cycle	287.7	289.0	290.7

Lengthening the test cycle obviously provides more time for cooldown of the system and for recovery of flywheel speed. Extending the flat-top requires more energy which results in higher temperatures and a larger droop in flywheel RPM. Table 4.1.15 5 shows the droop and recovery time for several flat-top times. The recovery times represent minimum permissible cycle times for the size motors selected, and several consecutive cycles below this minimum would cause excessive flywheel droop and failure of the experiment. This sharp go, no-go characteristic with regard to flywheel rpm is not true of TF temperature effects.

TABLE 4.1.15-5 FLYWHEEL DROOP AND RECOVERY TIME VS FLAT-TOP DURATION

*Minimum RPM (% of 90 hz RPM)	Flat-Top Duration (sec)		
	1	2	4
Recovery time	155 sec	171 sec	210 sec
*Note: Limit on minimum rpm is 67%, which would result in 60 hz output from the generator.			

Figure 4.1.15-1 shows the number of cycles the system can undergo (starting from a uniform 283 K) before the TF copper average temperature reaches the indicated values. For the system parameters used, temperature is not a constraint for flat-top durations of 2 seconds or less with cycle times greater than 200 seconds. These data are preliminary but indications are that considerable tradeoffs can be made between chiller size, flow rate, pipe size, ballast tank, etc. to arrive at an optimum system.

Effect of Locating Chillers Relative to Ballast Tank

A tradeoff study of the cooling water system was performed to determine the effect of the physical location of the chillers relative to the ballast tank in the coolant flow loop. The study was done using the TF coil cooling scheme of cooling the conductors one turn per circuit, even though the current cooling scheme is by two turns per circuit. The conclusion obtained from the older scheme also applies to the new scheme. Figure 4.1.15-2 shows the comparison of the chiller water inlet and outlet temperature with the two arrangements: chillers upstream of the ballast tank vs. chillers downstream of the tank. The figure shows that with the chillers located upstream of the ballast tank, the chiller inlet temperature is above its design point for a longer duration because the temperature variation is much greater than with the chillers located downstream of the ballast tank. Therefore the chillers for the cooling water system were located downstream of the ballast tank.

Effect of Number of TF Coil Cooling Turns per Circuit

A tradeoff study was performed to determine the effect of cooling the TF coils by one turn per circuit as compared to two turns per circuit. By cooling the TF coils with two turns per circuit and by maintaining the coolant velocity in the channel at 3.3 meters/sec, the total water flow rate was reduced from the previous rate of 0.46 m³/s to 0.23 m³/s (7200 gpm to 3600 gpm) and the ballast tank volume from 41.8 m³ to 21.0 m³ (11,000 gallons to 5500 gallons.) With the new water system capacity, transient thermal analyses of the TF coil cooling loop were performed. The comparison of the hottest coil temperature at the minimum cross section and the average coil temperature of hottest turn is shown in Figure 4.1.15-3. The coolant temperatures are shown in Figure 4.1.15-4. It can be seen that there is not a significant difference between the two cooling schemes. The additional heat generated in the two turns per circuit of cooling was eventually removed during the cool-down period. The scheme of cooling by two turns per circuit is therefore used for cooling the TF coils because less hardware is required.

Influence of Generator Reactance on Electric Power System Cost

The effect of generator reactance on the combined cost of the generator and the power supply thyristors was analyzed. Basically, the increased reactance would decrease the short circuit current, hence, decreasing the ratings and cost of the thyristors.

Figure 4.1.15-1. Average TF Temperature Reached Vs. Cycle, Cycle Duration and Flat-Top Time

Figure 4.1.15-2. Comparison of TF Coil Coolant Temperatures

Figure 4.1.15-3. Comparison of Coil Temperatures by Two Cooling Schemes

Figure 4.1.15-4. Comparison of Coolant Temperatures by Two Cooling Schemes

Because of the increased drop across the reactance, the MVA rating and cost of the generator would increase. Thus, it was anticipated that there may be a value of reactance which would minimize the combined cost.

Results of the qualitative analysis indicate that the total cost (generator and thyristor) decreases monotonically as the per unit source reactance increases, as shown on Figure 4.1.15-5. Discussion with some of the potential suppliers indicated that the per unit source reactance of $0.3 \approx 0.4$ is common and readily available. Therefore, a value of 0.35 is used for the reference conceptual design of the power supply and distribution system.

Motor Generator Building Cost Analysis

Analysis of the cost of four structurally adequate but different structural systems for the Motor Generator Building was made. The four alternative structural systems are shown in Figure 4.1.15-6. The basic building floor plan is 33.5 m x 61.0 m (110 ft by 200 ft). This study assumes that the cost of the metal siding and roofing and column footings will be roughly the same for all four alternates. The principal variation is in the high bay span of 33.5 m (110 ft), 22.6 m (74 ft), and 13.4 m (44 ft). Results of the study are summarized in Table 4.1.15-6. Alternative No. 3 is shown to be the least costly system which is functionally adequate and was therefore adopted as the conceptual design.

Figure 4.1.15-5. Power System Cost Vs. Reactance

4.2 TOKAMAK SYSTEM DESIGN

The conceptual design of the TFTR device is shown in Figure 4.2-1. This arrangement of components resulted from several design iterations which involved many tradeoffs to balance performance, accessibility, maintainability and cost. As shown, the vacuum vessel is a ring stiffened shell with bellows sections inserted at twelve locations to provide the necessary toroidal resistance. Twenty equally spaced toroidal field coils link the vacuum vessel to produce the principal component of the magnet field that confines the plasma. The toroidal field coils are water-cooled-copper, structurally reinforced with a "Nitronic 33" (a nitrogen strengthened austenitic stainless steel) case. Shear panels built into the coil subassemblies resist the overturning forces created by the various field interactions. The poloidal field coils which include ohmic heating field coil turns, equilibrium field coil turns, hybrid turns and adjustable field curvature turns are located outside the toroidal field coils.

There are two types of device shielding referred to as the vessel shield and enclosure shield. The vessel shield is located in the space between the vacuum vessel and the inside diameter of the toroidal field coils. The enclosure shield is located in the spaces between the toroidal field coils, around the outside perimeter of the device, below the device, and above the device. This shielding is supplemented by portable panels and localized equipment shielding where needed. The device substructure consists of two

ALTERNATIVE NO. 1 - One 110' High Bay With Bridge Crane (200 Ton)

ALTERNATE NO. 2 - One Low Bay 36', One High Bay 74' With 200 Ton Bridge Crane

ALTERNATIVE NO. 3 - 2 Low Bays, 36' and 30'; 1 High Bay 44' with 200 Ton Bridge Crane

ALTERNATIVE NO. 4 - Same as No. 3 but Using Gantry Crane (200 Ton)

Figure 4.1.15-6. Alternative Structural Systems for MG Building Cost Study

TOKAMAK FUSION TEST REACTOR

Figure 4.2-1. TFTR Device

**TABLE 4.1.15-6 COST COMPARISON FOR ALTERNATE STRUCTURAL SYSTEMS
MOTOR GENERATOR BUILDING**

Alt. No.	Wt. -Tons	Cost at \$850/ton (1000's)	Crane Rail Cost-\$ (1000's)	CraneRail Fdn-Cost \$ (1000's)	Crane Cost-\$ (1000's)	Total Cost-\$ (1000's)
1	254	215.90	9.12	N/A	750.00	975.02
2	197	167.45	9.12	N/A	600.00	776.57
3	204	173.40	9.12	N/A	500.00	682.52
4	154	130.90	16.80	74.13	600.00	821.83

platforms, one of which is supported on roller bogies to permit separation of the device for remote maintenance. The other platform is supported directly by the floor and remains fixed.

The major differences between this Tokamak and its predecessors result from its scale and from its planned operation with D-T plasmas. The toroidal field coils are so large that the mechanical properties of copper are inadequate for their construction; the copper is therefore supported by structural stainless steel which serves also to perform internally the functions of the external frames of previous machines. The machine will become radioactive and require remote maintenance; as a consequence of this, the machine and its support are designed in detail for remote disassembly and reassembly. Finally, the machine in operation is a source of intense gamma and neutron radiation; it is therefore surrounded to the greatest extent possible by a uniform layer of shielding.

As a matter of convenience, the components of the device system have been separated into four major subsystems, i. e., vacuum vessel assembly, structure and shielding, toroidal field coil assemblies and poloidal field coil assemblies. Each of the subsystems is discussed in detail in the following sections in a progressive manner; viz., a discussion of the functions and design requirements considered in developing the component (or system), a description of the conceptual design satisfying these requirements, and finally, design alternates which are still worthy of consideration.

4.2.1 VACUUM VESSEL ASSEMBLY

Design Considerations

The ultimate objective of the TFTR is to achieve a fusion reaction of tritium and deuterium in a magnetically confined toroidal plasma. This plasma will be initiated, heated,

manipulated, observed and measured, and quenched inside the vacuum vessel assembly. The vacuum vessel therefore, has two active functions:

- to exclude the atmosphere from the toroidal space in which the plasma exists
- to contain the reactant gases before initiation and after termination of the plasma

In addition to these, the vacuum vessel has five passive functions:

- to cause minimum contamination of the plasma
- to cause minimum distortion of or interference with the magnetic fields
- to provide access to the plasma for other systems
- to resist damage to itself by the operation of the TFTR
- to be amenable to maintenance, repair, modification, or replacement

In a design as complex as that of the vacuum vessel, no feature provides for any single function completely and exclusively. In a general sense, however, exclusion of the atmosphere and containment of the reactant gases are provided by the vessel being designed as an all-metal externally loaded pressure shell, fabricated by techniques proven in ultra high vacuum technology. Contamination of the plasma is minimized by selection and treatment of material, avoidance of virtual leaks, and use of only plastically deformed metal seals. Disturbance of the magnetic fields is minimized by the use of stable non-ferromagnetic material arranged geometrically to minimize induced electrical currents. Access to the plasma is provided through shell penetrations adequately reinforced. Resistance to damage is provided by the internal protective plates and limiters. The feature which provides for maintenance or replacement is the two circumferential closure joints which permit the torus to be separated into two halves and disengaged from the surrounding TF coils.

Design Description

The vacuum vessel assembly, which is shown in Figure 4.2.1-1, consists of the following components:

- Toroidal vacuum vessel shell
- Closures
- Penetrations
- Supports

- Neutral Beam Adapters
- Protective plates
- Limiters

These components will be described separately. All these components to the greatest extent possible are fabricated of AISI Type 305 L austenitic stainless steel, vacuum induction melted and vacuum arc re-melted. Exceptions are:

- Protective Plates - tungsten and molybdenum, required for their high melting temperature
- Limiter blades - tungsten, required for its high melting temperature
- Fasteners - Nitronic 33, required for its resistance to galling
- Gaskets - OFHC copper, required for its ductility
- Electrical Insulators - mica (if flat) or ceramic

The vacuum vessel dimensions were defined in terms of an external clearance envelope inside the toroidal field coils and an internal clear space for the plasma. The clearance envelope is a torus of approximately 280 cm major radius and 137.5 cm minor radius; all attachments to the vessel are provided with disconnections inside this envelope.

Figure 4.2.1-1. Vacuum Vessel

The clear space for the plasma is a torus of 265 cm major radius and 110 cm minor radius; the protective plates encroach on this volume 3 cm at the major equator and the outboard limiter blade can be withdrawn to a point where it encroaches only 5 cm on this volume.

Between these two envelopes is the vacuum vessel. It is a 30,000 kg welded stainless steel toroid made of twenty mitred ring-stiffened cylindrical segments; the number of segments is chosen to agree with the number of TF coils, and the ring-stiffened construction was chosen in preference to a uniformly thick shell to minimize horizontal eddy current loops which would increase the EF time constant. If the stiffening rings could be uniformly spaced and served no other function, there would be 20 rings having a 7.5 cm square section. However to permit 7.5 cm penetrations without local reinforcement the width of the rings has been increased to 15 cm. Two rings of equivalent cross section, modified to be closures which can be remotely separated and rewelded are located at the device parting plane. The shell plates which are welded to the rings at their 235 cm outside diameters are 0.8 cm thick. A recent change in the vessel design was to relocate these shell plates from the inside diameter of the rings to the outside diameter. This increases the volume to be pumped down by 12.6% but increases the internal area of the vessel, i.e., the outgassing surface, by only 1.2%. This change eliminates the need for external stiffening ribs in the toroidal direction to reduce bending stresses in the bellows end rings.

To increase the electrical resistance of the vacuum vessel in the toroidal direction, which must be large compared to the resistance of the plasma, the shell plates are interrupted at twelve locations around the torus by the inclusion of welded metal bellows (see Figure 4.2.1-2). There are 46 bellows convolutions at each location. The bellows rings are 10 cm in the radial dimension and 0.10 cm thick. The ends of each group of bellows convolutions are attached to rings of L-shaped cross-section; the legs of the Ls parallel to the major circumference of the torus are inside the bellows and protect them from the plasma. The other legs of the Ls are welded at their outer circumference to the shell plates. (These rings and the closure rings were not used as stiffeners in the analysis of the vessel, although they do contribute to its strength.) Electrically insulated stainless steel cover plates span the gaps between bellows end rings and carry the atmospheric load across the bellows. The space between the bellows and the cover plate is evacuated by the fore-pumps of the torus vacuum pumping system to minimize flexing of the bellows and to hold the convolutions open for better out-gassing.

At two locations, midway between normal stiffening rings and 180° apart, are closures to permit final assembly and later disassembly of the vessel. Welding and cutting for separation are performed inside the vessel. Access for the welding and cutting machines is through a 63 cm diameter nozzle adjacent to the closure. The detail design of the closure will be developed in conjunction with the manufacturer of the welding and cutting machines; potential suppliers consider this closure to be within the state of the art.

Figure 4.2.1-2. Welded Metal Bellows

There are a total of 240 penetrations into the interior of the vacuum vessel. Of these, 195 are 7.5 cm inside diameter and are made directly through the stiffening rings without reinforcing. All other penetrations are reinforced. The largest of these are six rectangular nozzles, 43.8 cm wide and 72.4 cm high, inclined approximately 45° to the surface of the vessel, opening on the vessel equator midway between stiffening rings. These nozzles are for the neutral beam injectors and permit injection of beams tangentially into the plasma; three nozzles are inclined to permit clockwise injection and three to permit counterclockwise injection. At each of the stiffening rings, except the six closest to the neutral beam nozzle entrances, a 63 cm nozzle projects radially from the vessel equator. Of these 14 nozzles, seven are required for mounting the internal protective plates, two are required for closure access, and one is dedicated to the limiter. The four remaining 63 cm nozzles are solely for diagnostic instrumentation access; nine others (excluding the one dedicated to the limiter) are available for instrumentation that does not interfere with their primary functions. At four equally spaced locations, pairs of adjacent stiffening rings are fitted with 23 cm nozzles at the top and bottom of the ring and at two of these locations (rings which do not have 63 cm nozzles) rectangular nozzles 15 cm wide by 61 cm high on the vessel equator; to avoid interference with the adjacent neutral beam nozzle, these rectangular nozzles are located 2.85° on the circumference of the torus (19 cm) from the radial plane of the center of the stiffening ring.

At one location a stiffening ring has 15 cm vertical nozzles top and bottom and a 15 cm radial nozzle. Midway between the ring and the next, a similar set of nozzles is located in the shell plate; these, the neutral beam nozzles, and the two 15 cm by 61 cm rectangular nozzles are the only penetrations that are not through stiffening rings.

The vacuum vessel is supported by twenty vertical struts, one extending downward from each stiffening ring. A continuous semi-circular horizontal rail is bolted to the ten struts on each half of the vessel. This rail is electrically insulated from the vessel. The rail is entirely within the clearance envelope of the vessel and rests on horizontal rollers which project upward into that space.

The neutral beam adapters are mechanically sealed to and removable from the neutral beam nozzles, and must be installed after the vacuum vessel is inside the toroidal field coils. They provide a clear passage as large as the neutral beam nozzles through a transition from a 91 cm ID circular flange, which is the interface with the neutral beam system. Each neutral beam adapter contains a vertical pivot sealed by a rectangular bellows section which permits the neutral beam to be injected tangentially into plasmas ranging from 220 cm to 270 cm major diameter. Two of the four neutral beam adapters also contain a nozzle for a 24 inch (61 cm) mercury diffusion pump.

Under some conditions the neutral particle beams might impinge directly on the vacuum vessel shell. At locations where this is geometrically possible, the vacuum vessel is protected internally by refractory metal plates. The plates form a continuous band extending 126° around the major circumference and 15° above and below the equator. The surface of each plate is composed of 1 cm cubes of tungsten, diffusion bonded to a 0.3 cm thick molybdenum plate. The cubes are positioned with one face diagonal running in the toroidal direction; with this alignment the neutral beams, which strike approximately at a 45° angle to the surface, cannot directly strike the molybdenum plate through the 0.1 cm gaps between cubes. An unattenuated strike of a neutral particle beam delivering 7 kW/cm^2 will cause the temperature of the exposed tungsten surface to rise approximately to 3300°C . The protective plates are therefore mechanically attached to molybdenum struts approximately 60 cm long which in turn are mechanically attached to the stainless steel vacuum vessel structure. The temperature drops through the two mechanical joints and the temperature gradient along the length of the struts protect the stainless steel from thermal damage. There are 21 protective plates installed in sets of three through seven adjacent 63 cm nozzles. Each set consists of two side plates, one on each side of the nozzle, and a center plate mounted on the blind flange which closes the nozzle.

At a single location, a set of limiters is installed in the vessel. The upper, lower and inner limiter bars are fixed, but are removable and replaceable through a 63 cm nozzle; the outer limiter is movable radially, being mounted on a bellows-sealed ram which in turn is mounted on the blind flange which closes the nozzle. The inner limiter is a

vertical bar positioned in the torus with its surface exposed to the plasma at 163 cm major radius. The upper and lower limiter bars are curved, each extending approximately 90° around the minor circumference of the vacuum vessel. The outer movable limiter is also a vertical bar, 62 cm in the vertical direction to enter the vacuum vessel through the 63 cm diameter nozzle; the radial stroke of this limiter is 105 cm. The limiters included in the conceptual design are massive uncooled tungsten bars; unless a supplier of larger ingots can be found, the bars will be made of cold-worked plates diffusion-bonded together at a temperature below their recrystallization temperature.

Design Alternatives

Many alternative design features were considered in the development of this conceptual design. Some have desirable characteristics in comparison with those incorporated in the conceptual design but were not included because of uncertainties concerning other characteristics. Typical alternatives, which could be incorporated into the vacuum vessel design as soon as the existing uncertainties are successfully resolved, are discussed below:

- Material - a structural material having higher yield strength than AISI 305, such as Nitronic 33, could reduce the weight and cost of the vessel slightly or, with its equal strength at higher temperature, permit a higher operating and vacuum bake-out temperature. An alloy of higher electrical resistivity, for example Hastelloy C-276, would permit elimination of half the bellows convolutions presently required, and would reduce the out-gassing of the vessel by 42%. Such a change might also result in a significant cost reduction.
- Structure - ceramic or graded Kovar-glass electrical insulators breaking the conducting paths along the major circumferences of the torus would permit elimination of all the bellows convolutions. This would reduce the internal area of the vessel, and probably its outgassing rate, by 84% and most likely would result in a significant cost reduction. If such insulators are demonstrated on another machine, for example the alumina rings for PLT, as many as needed to distribute the resistance could be substituted for the bellows assemblies, of which there are presently twelve, without disrupting the present configuration of the vessel.
- Limiters - the conceptual design includes a single set of simple limiters. All the limiters are replaceable but only the outer limiter can be repositioned, and only between pulses. As more information is developed concerning the functions of the limiters (on PLT and other machines), more sophisticated limiters may be developed. The most elaborate mechanical limiters so far proposed consist of a fixed inner limiter bar, upper and lower limiters each consisting of two bars each mounted on two actuators, so that both elevation and inclination can be controlled, and a fast-acting out limiter that can guide or follow the plasma during its lifetime.

- Ultra high vacuum seals - at present the only fully acceptable UHV seals are welds and plastically deformed metal gaskets; a double seal consisting of an inner metal seal elastically deformed and an elastomeric atmospheric seal, with the space between the seals continuously pumped down, is marginal. The available plastically deformed metal gaskets, either gold wire or OFHC copper sheet, are loaded by bolted flanges and are difficult to manipulate remotely. A new concept in plastically deformed metal seals depends on the unique properties of NITINOL, a Ni-Ti alloy in which dimensional changes up to 5% accompanied by stresses up to 345 MPa (50,000 psi) result from a combination of cold straining followed by an energetic phase transformation at a transformation temperature below room temperature. The force associated with this stress can be reacted to plastically deform a metal seal. A small aircraft hydraulic seal based on this principle (Cryofit fitting) has been helium leak tested and proven to be an acceptable UHV seal. The size range through which such seals can be used may even include the 2.35 meter torus seal. Development of a full range of seals of this type would greatly simplify maintenance of TFTR.

4.2.2 STRUCTURE AND SHIELDING

The structure and shielding system provides the structural support for the TFTR components, radiological shielding for operation of the device with a tritium plasma, and the coolant headers and manifolds for water cooling the device field coil systems and their related buswork.

Design Considerations

The device structure is required to support and maintain relative position of the major TFTR components during operation. These components, identified in Figure 4.2-1, include the toroidal vacuum vessel, toroidal field coils, ohmic heating field coils, equilibrium field coils, device shielding, coolant headers, and related hardware. The total weight of these components is in excess of 800,000 Kg. In addition to supporting these components during operation, the structure provides for separation of the device for maintenance.

Early in the conceptual design phase the support structure was defined as two massive slabs of reinforced concrete fitted with air casters to provide the mobility for separating the device. As the design developed, three serious concerns surfaced relative to a system of this type. First, air casters require an initial lift of approximately three quarters of an inch to establish the air cushion. This would require lifting both halves of the device simultaneously and uniformly to avoid shearing the connectors which bridge the device parting plane. Second, a relatively flat clear floor is necessary for smooth movement. With all of the service lines in and around the device it is not reasonable to expect that the floor can be kept clear over such a large area. Third, the size and weight of the concrete slabs is such that they must be cast in the TFTR

test cell. This leaves little or no flexibility for later changes and creates scheduling problems and logistics problems for other components of the device. For these reasons other approaches were considered. The one that appeared to offer the best compromise was two identical platforms formed by a grid of austenitic stainless beams supported on hardened steel rollers which ride on rails built into the test cell floor. The primary advantage of the system is the provision for lateral movement of the device halves while maintaining a fixed vertical elevation. Other advantages of this arrangement are that the radiation shielding, which consists of small concrete blocks supported by the flanges of the I-beams, can be changed or modified relatively easily and that the entire structure can be assembled and dismantled in the field. It also makes it possible to leave the shielding out for better access during the non-tritium phase of the experiment.

Subsequently, in the interest of cost and simplicity the design was changed to provide movement for only one half of the structure, leaving the other side fixed during maintenance. The diagnostic instrumentation protruding from the vacuum vessel into the basement was concentrated on the fixed side to take advantage of this change, thus eliminating the need to cut and re-weld the instrumentation tubes during separation of the device.

All previous versions of device structure included a permanent center column. The desire to make provisions for inserting an iron core in the center of the device for later experiments led to a design which eliminates the permanent center column. Previously, the center column was used to locate and support the inside sections of the EF/OH field coils and the top shield. In the present design these functions have been transferred to a hollow central cylinder which is an integral part of the EF/OH field coil support structure.

In addition to supporting and locating the TFTR device, the structure and shielding system provides most of the shielding necessary for attenuation of radiation produced by the device. During operation of the TFTR device, the tritium plasma is the major source of radiation, mostly neutrons. At other times the major source of radiation, primarily gamma, is material which has been activated by the neutrons. The vacuum shell obviously cannot be shielded and will become highly activated. The primary shield therefore must be a material effective in absorbing both high energy neutrons and high energy gamma radiation. The material should have low magnetic permeability and high electrical resistance to minimize electromagnetic interaction with the equilibrium field, which could delay field changes necessary for fast compression of the plasma. The material must be easily fabricated into irregular shapes by machining or casting and must have reasonably good mechanical properties. In addition, the shielding material must be non-flammable and should not give off gases at the radiation exposure levels anticipated in the TFTR device. Various shielding materials were evaluated and the materials considered to exhibit the best overall capabilities are lead-loaded boronated mortars.

Ideally, it is desirable to locate all of the shielding in the space just outside the vacuum vessel. This would keep the activation of other components outside this area to a minimum, thereby reducing the waiting time after shutdown for access to the device for maintenance. Since this space is limited, it is not possible to locate all of the shielding in this area and, in fact, only a small percentage of the total shield can be located in this area. The bulk of the device shielding is placed above, around and below the device to form a continuous enclosure. Due to the many penetrations in the shield enclosure for neutral beam access and diagnostic instrumentation, it is necessary to supplement these shields with portable panels and other types of flexible shielding to eliminate streaming paths.

The cooling water headers, manifolds, and related hardware provide for the distribution of cooling water to the toroidal, ohmic heating, and equilibrium field coils, and to their buswork at flow rates and pressure differentials that are adequate to satisfy the coil design requirements. Various locations were considered for positioning the cooling water headers including a location below the structural floor of the device; however, significant design constraints were encountered in order to accommodate separation of the device and to provide for accessibility during remote maintenance.

The location that best satisfies all requirements appears to be above the device with the headers supported from the top of the toroidal field coils. The primary advantage of the location is accessibility for maintenance. The disadvantage relates to leakage spreading down over the components of the device.

Structure Design Description

The device structure, Figure 4.2.2-1, consists of a fixed and a movable rectangular platform, each approximately 12 x 6 meters (40' x 20') in area and 76 cm (30") thick which, when brought together, form a 12 meter square platform; the corners of the square platform, which have no function, are removed to reduce weight and cost. A semi-circular cut-out in each mating side forms a 6.1 meter (20') diameter opening in the center of the assembled square platform. The movable platform is supported on roller bogies which ride on nine hardened steel round ways set in trenches to make them flush with the floor of the Test Cell; these ways run perpendicular to the mating plane of the platforms. The fixed rectangular platform, which is symmetrical to the movable one, is supported directly by the floor on pads in the same locations as the bogies on the movable platform.

The platform structure, which must transfer forces from the points of load application in the device to the points of support (on the ways or on the floor) with minimum deflection consists of an upper layer of closely spaced I-beams, 46 cm deep, resting on widely spaced I-beams, 30 cm deep. The I-beams are austenitic stainless steel and are held together by electrically insulated bolted joints. The floor of the platform consists of a 1.27 cm (0.5") thick layer of commercially available glass reinforced sheet insulation. The two layers of I-beams are designed to accommodate installation

Figure 4.2.2-1. Device Structure

of shielding sections which provide approximately 76 centimeters of shielding under the device. This shielding supplements the 122 cm (48") thick test cell floor to protect diagnostic equipment in the basement.

The rollers in each bogey are linked together to form a recirculating chain of rollers under the load. Each bogey on the center way consists of two parallel chains of rollers straddling the way to provide guidance for the platform; each double chain bogey is capable of supporting 32,000 kg. The bogies on the other ways consist of single roller chains riding directly on top of the ways to eliminate lateral forces on the platforms; each single chain bogey is capable of supporting 22,000 kg.

The coefficient of friction of roller bogies is stated by the manufacturer to be 0.02 to 0.05. For the higher value, which represents little or no lubrication, a horizontal force of 180,000 N is required to overcome friction on one-half of the support structure; acceleration and disconnecting forces are small compared to the friction force. The necessary forces are generated by hydraulic cylinders working on a 7 MPa system. Two cylinders are attached to the movable platform at floor level at each side. With the device either closed or fully separated, each pair of cylinders consists of one cylinder fully extended and one cylinder fully collapsed. With this arrangement, the platform can be moved in either direction with the piston rods always in tension; the rods are 6.5 cm dia., the cylinder bore is 21 cm, and the stroke is 9 meters. Each cylinder is capable of moving the entire load at 5.7 MPa pressure; the expected operating pressure is 2.9 MPa. The end connections of all cylinders are always accessible from above so that each cylinder is removable for maintenance.

During operation the movable half of the device is supported on its roller bogies. The fixed half rests on the floor acting as an anchor for the entire device. Final alignment of the device halves as they come together is provided by guide pins and dowels rather than by the roller track ways.

Device Shielding Design Description

The Device Shielding (Figure 4.2.2-2) consists of a vacuum vessel shield functioning as the primary shield and an overall enclosure shield which supplements the vessel shield. The vessel shield is located in the space between the vacuum vessel and the inside diameter of the toroidal field coils. The vessel shield consists of individual segmented shield sectors fitted closely around the outside diameter of the vacuum vessel. This shield is supported by the vessel and moves with it when the vessel is removed from the TF coils for maintenance. The shield is contoured to be as thick as possible while still avoiding interference with the TF coils, and has penetrations wherever necessary for access of neutral beams, instrumentation, etc., to the vacuum vessel. The individual segments have attachments and lifting lugs suitable for removal using remote handling equipment in the hot cell. The vessel shield material is lead-loaded boronated mortar encased in fiberglass reinforced plastic for containment and strength.

Figure 4.2.2-2. Device Shielding

The overall enclosure shield, which supplements the vessel shield, consists of:

- A top shield formed by a cylindrical pancake located at the center of the device surrounded by radially segmented shield blocks forming a large circular shield disk. The top shield is supported by the central support structure for the poloidal field coils.
- The upper peripheral shield, which consists of shaped blocks inserted between the TF coils above the horizontal centerline of the device. These shield blocks are supported by the TF coil structure.
- The lower peripheral shield, which consists of twenty pedestals of shield material supported from the structural support deck up to the lowest of the upper outer turns of the EF and OH coils. These pedestals are outside the lower outer turns of the EF and OH coils and form a continuous circular stockade of shielding material. At the top of each pedestal is an inward projection which fills to the greatest extent possible the gaps between adjacent TF coils.
- The base shield, which consists of shield blocks fitted into the lattice of the structural support grid and radially segmented shield blocks located on the floor filling the circular cutout in the center of the platform.

- The concrete floor slab, which is not considered to be part of this system for any other purpose, is the final element necessary to complete the shield enclosure around the TFTR device.

All the shield components (including the Test Cell floor) are modified locally to provide access penetrations to the device and to prevent radiation streaming through the penetrations. The enclosure shield material is lead-loaded boronated mortar encased in fiberglass reinforced plastic to improve handling characteristics. The individual shield blocks are fitted with positioning pins, lifting devices, etc., compatible with the remote handling equipment in the Test Cell.

The roof and outer walls of the Test Cell constitute the biological shield which completes the protection of the environment from the radiation produced by the TFTR device.

Coolant Headers Design Description

The coolant water headers (Figure 4.2.2-3) consist of 45.7 cm (18") dia. pipes fabricated to form four 333 cm (131") major radius semicircular halves supported from the TF coils at the top of the device. One header set provides inlet flow and the other accommodates outlet flow for the toroidal field coils and several turns of the EF and OH field coils. The headers are split at the device parting plane to accommodate separation of the device during maintenance operations. Individual manifolds for each TF coil protrude radially outward from the headers and provide for installation

Figure 4.2.2-3. Water Cooling Headers

of flexible insulating hoses which carry the cooling water from the manifolds to the field coils. The manifolds are attached to the headers with quick disconnect flanges which are removable by remote handling equipment to accommodate coil replacement without removing individual cooling hoses. The individual hoses are also replaceable and contain valved, quick disconnect fittings. The headers are fabricated from austenitic stainless steel.

Design Alternatives

Many design alternatives were identified during the course of the development of the structure and shielding conceptual design. A discussion of the most promising of these alternatives deserving further evaluation and consideration follows.

The movable half of the device structure is supported on roller bogies which ride on hardened steel round ways. Alternate methods of support and translation of the movable structure were considered including standard steel wheels and rails and strategically located HILMAN (or equivalent) type supports similar to those used to move large machinery. These methods could result in a cost savings and improve serviceability.

The fixed half of the device structure is permanently installed on the Test Cell floor. Consideration was given to providing a portable means for moving the fixed structure in the event it is necessary, due to an unanticipated modification or adjustment. The approach would involve providing trenches in the floor under the fixed half of the device. To transport the fixed half, portable HILMAN type supports would be slipped under the device via the trenches.

Shielding is provided between the TF coils by individual shielding blocks located in the space between the structural shear panels. Combining the shielding and shear panels into an integral structural segment was considered as an alternate method of providing this shielding. This structural segment could be installed between V-shaped TF coils during their fabrication. The potential advantages are higher structural capability to resist coil loads and deflection, more effective shielding, and fewer components which could simplify remote handling operation.

In addition, arrangements which locate this shielding outside the toroidal field coils were also considered. These are shown in Figures 4.2.2-4 and 4.2.2-5. The advantage of these arrangements is that the shielding is decoupled from the TF coils, thus providing more uniform shielding and easier access for remote handling.

The device structure is designed to accommodate installation of individual shielding blocks between the I-beams of the platform. An alternate design in which the blocks on the lower grid are supported from the floor of the test cell was considered. This approach reduces the weight which must be handled by the roller bogies. The disadvantage is that more clearance must be provided between the blocks and the structure to assure that no interference occurs when the structure is moved. This, of course, means less effective shielding.

Figure 4.2.2-4. Alternate Shielding Arrangement

Figure 4.2.2-5. Alternate Shielding Arrangement

4.2.3 TOROIDAL FIELD COIL ASSEMBLIES

Design Considerations

The toroidal field coil assemblies produce the principal component of the field that confines the plasma. To produce the confining magnetic field, the coils will be cycled in a pulsed mode where each cycle will consist of ramping dc current up to a desired flat-top value, holding and then decreasing the current to zero. This cycle may be repeated as often as every 5 minutes.

Because it is necessary to distribute the toroidal field windings into a discrete number of coils located around the torus to provide access to the vacuum vessel for injection of neutral particles, vacuum lines, and diagnostic instrumentation, consideration must be given to the resulting field ripple. Consistent with providing the necessary access, the field ripple must be maintained below 1% in the plasma region of primary interest and should not exceed 5% in the region where the plasma is initially formed. One must also consider the structural implications of the coils themselves since the forces generated by the high field (9 to 10 tesla) are very large. The coil assemblies must also resist the loads imposed by interactions with the poloidal coil systems and other components of the device which they support. It is necessary that these loads be resisted with minimal deflection since large deflections can distort the field shape.

Several coil configurations were considered before arriving at the present design. These included a "D" shaped coil and various methods of structurally reinforcing the copper windings to satisfy the deflection requirements. The "D" shaped coil was dropped primarily for economic reasons. With a circular plasma and the various compression scenarios, a circular coil could be utilized to minimize the quantity of copper and the resistance of the coil.

Although the in-plane bending stresses are slightly lower for the "D" shaped coil the reduction is not sufficient to eliminate the need for significant structural reinforcement.

The initial TF coil design consisted of 20 coils structurally reinforced with an integral Inconel 718-I section. Torque due to interaction of poloidal fields with the current in the TF coils was resisted by an octagonal austenitic stainless steel torque frame which surrounded the device.

Later the components of the device were rearranged to locate the equilibrium field coils outside the TF coils. This permitted the TF coil size to be reduced, and consideration was given to elimination of the torque frame. In order to eliminate the torque frame, it was necessary to incorporate efficient shear panels between the coils. The toroidal field coil structural support was modified extensively to implement the shear panels. A coil assembly now consists of a section taken between the central planes of what formerly constituted two neighboring coils (see Figure 4.2.3-1).

Figure 4.2.3-1. Toroidal Field Coil Assembly

Reference Design Description

The toroidal field coil assembly shown in Figure 4.2.3-1 consists of two water-cooled copper windings, structurally reinforced with back-to-back Nitronic 33 channel sections. The channel sections are joined at the coil nose to form a "V" shaped assembly. Twenty equally spaced TF coil assemblies encompass the toroidal vacuum vessel. Each TF coil assembly mates with adjacent coil assemblies on a radial plane so that a slit box section is formed around the copper windings. Closed loops for induced currents are minimized by insulating the mating surfaces with plasma sprayed aluminum oxide coatings. Fitted shear bolts which pass through tabs at the coil mating interfaces clamp the structures except at the device parting plane. At this interface, shear pins are used in the tabs to facilitate parting of the device when maintenance is required.

A winding consists of two copper pancake coils, each having twenty-two turns connected in parallel. The two windings are wired in series to provide forty-four turns for each toroidal field coil assembly. The copper conductors for the coil that mates with the inside of the case are contoured before winding and insulated to provide the wedge shape necessary at the coil nose (see Figure 4.2.3-2). The other coil is a flat pancake which requires no contouring.

The coils are fabricated from extruded copper conductors which have an integral coolant hole. Each turn is wrapped with two layers of 0.05 cm (0.020 inch) thick B-stage insulating tape before the coil is wound. After winding, the two coils are joined

Figure 4.2.3-2. TF Coil Nose Cross Section

together and overwrapped with two additional layers of 0.05-cm (0.020 inch) thick B-stage ground insulation. This is supplemented by fifteen layers of .005 cm (.002 inch) thick mylar ground insulation. The entire assembly is placed in the Nitronic 33 structural channel and impregnated with epoxy to form a monolithic coil assembly.

Water cooling for the copper conductors is supplied from manifolds located above the TFTR device. These manifolds are supported from the tops of the toroidal field coils. Water is routed to the coils via electrically non-conducting plastic hoses. These hoses are connected to copper tubes which are brazed into the conductors and penetrate the sides and back of the coil case. There are two parallel cooling circuits per turn. Coolant enters the coil and makes two complete turns and then exits to the outlet manifold. In all, there are forty-four coolant circuits for each toroidal field coil or a total of 880 circuits for the twenty coils. Each circuit contains a flow indicator to verify that no blockage has occurred. Table 4.1.6-2 provides a summary of the toroidal field coil thermal design.

The centering forces resulting from the toroidal field are resisted by the wedging action of the steel box sections at the nose of the coil. The torque due to interaction of the poloidal field and/or equilibrium field is resisted by shear panels bolted into the coil assemblies. The shear bolts which attach these panels require a close fit which is accomplished by line drilling and reaming at assembly.

The dead weight of the coils is supported by the device substructure. Ten coils are supported on each half of the substructure. When the device is separated for maintenance, the toroidal field coils travel with the substructure with all electrical and water cooling supply lines intact. After the vacuum vessel is removed an individual coil can be disconnected and removed for maintenance or replacement.

Detailed specifications for the toroidal field coils are listed in Table 4.2.3-1.

Status of Design Alternates

The alternative design features still under consideration for the toroidal field coils center around the shear panels. Alternate means of attaching the shear panels to the coil structure, such as shear keys, taper bolts, etc. should be investigated. In addition, alternate locations for the panels should be considered. The primary candidate in this regard would be panels which bridge the coils to form a shell type enclosure. These panels would be bolted to the backs rather than the sides of the coil cases. Radiation shielding between the coils could be supported directly from the panels.

Other areas deserving further consideration are the method of winding the copper coil, the insulation system and the water coolant entrance and exit arrangement.

TABLE 4.2.3-1 TOROIDAL FIELD COIL SPECIFICATIONS

Parameter	Units	Value
Ampere	MA _t	64.4
Number of coils		20
Turns/coil		44
Ampere turns/coil	MA _t	3.22
Coil Current	kA	73.2
Total turns		880
Coil major radius, R ₀	cm	280
Inside radius of copper	cm	145
Outside radius of copper	cm	189
Field at coil center	T	4.6
Field at vacuum vessel center	T	4.86
Maximum field in coil	T	9.54
I ² R	MW	340
1/2 LI ²	MJ	1360
Equivalent squarewave pulse length	s	7.47
Total dissipated energy per pulse	MJ	2538
Inductance of T. F. Coil System	H	.510
Resistance of T. F. Coil System @ 328°K (55°C)	Ω	.0634
Coil time constant	s	7.49
Coil centering force per coil	MN	26.2
Mass of copper (one coil)	kg	12800
Mass of "Nitronic 33" stainless steel	kg	4900
Total mass of finished coil (one)	kg	17700
Max. current density	kA/cm ²	3.32
Insulation test voltage	kV	30

4.2.4 POLOIDAL FIELD COIL ASSEMBLIES

Design Considerations

The primary functions of the poloidal field coil assemblies are to stabilize, heat, and compress the plasma. To produce the required magnetic fields to accomplish these functions, the coil assemblies are operated in a pulsed mode where each cycle consists of ramping dc current up to a desired value and then rapidly reversing and/or increasing the current.

The initial poloidal field coil assembly concept consisted of separate ohmic heating field (OH) and equilibrium field (EF) windings. The OH coils were located outside the toroidal field (TF) coils and the EF coils were located in the space between the vacuum vessel and the bore of the TF coils. The OH coils were supported from the device central column, while the EF coils were supported by a structural frame built upon the rail system which was common with the vacuum vessel.

With this arrangement, an electrical connector was required for each EF coil turn for assembly/disassembly of the device. As the conceptual design evolved, the EF coils were moved outside the TF coils which eliminated the need for electrical connectors on the coil turns supported by the central support structure, since they remain fixed when the device is parted for maintenance. This relocation improved access for disconnecting the turns outside the TF coil vertical centerline which were now supported by the TF coil structure.

Consideration was given to eliminating the electrical connectors in these outside turns by making provisions for removing them as complete hoops when the device is to be parted. This approach is not unreasonable for the turns above the horizontal centerline of the device since they could be released from their support brackets and lifted vertically upward. Of course, other features of the device would have to be changed to open a corridor for the hoops to pass through; however, this could be accomplished relatively easily for these upper turns. Once the turns are stripped off, the device accommodations for handling and storing these large diameter hoops in the Test Cell must also be considered.

The turns below the horizontal centerline of the device present a considerably more difficult problem. To drop these turns down in order to allow the device to pass over them, requires that the support structure size be reduced so that it lies within the diameter of the smallest hoop. This reduces the support structure to nothing more than a narrow ring. In addition, the rails that support the movable side of the structure must contain a removable section which can be taken out and replaced after the coils pass through them. Although all of these features could be incorporated in the device design with some compromises in other areas, the fact that ultimately numerous operations in very restricted areas must be performed remotely makes this approach undesirable.

Magnetic field analyses of the EF-OH coils revealed that there could be advantages in using some of the same turns for both EF and OH functions. This was especially true for the turns in the center of the device where space is limited. These turns that serve both EF and OH functions are designated hybrid (HY) turns and during the flat-top portion of the pulse, operate at a lower net current since the EF and OH currents are in opposite directions. The only disadvantage of this arrangement is the need for additional electrical connections for the hybrid turns.

Reference Design Description

The poloidal field coil assemblies are shown trimetrically in Figure 4.2.4-1. The assembly consists of four different windings as tabulated below:

- Ohmic heating
- Equilibrium
- Hybrid
- Adjustable field curvature

Figure 4.2.4-1 shows the location of the turns for these windings relative to the device center and the TF coils. Turns which lie inside the TF coil centerline are supported by a central cylinder; those outside the TF coil centerline are supported by the TF coil structures. The central support structure is a hollow cylinder which allows for insertion of an iron core should it become desirable for later experiments.

The turns which make up the center portion of the coil are preloaded by clamping them between plates with tie bolts. The remaining turns are clamped to the support arms in groups containing various numbers of conductors. There are twenty upper and twenty lower support arms each located in the plane of a TF coil. The support arms are bolted to metal inserts in the central cylinder. When the device is separated for maintenance, these windings remain fixed, supported from the stationary central cylinder.

The turns which lie outside of the TF coil centerline are supported by brackets attached to the TF coil structure. These turns are backed up with fiberglass reinforced split rings which in turn are backed up with a tension band which has tensioning devices (Figure 4.2.4-2) at the device parting plane. The tensioning devices provide a means of preloading the coil to react coil loads induced by the magnetic fields.

"Multilam" electrical connectors (Figure 4.2.4-3) are used to connect the half turns at the parting plane of the device. These connectors are designed to break and remake automatically when the device is separated or closed. Close alignment of mating turns at the parting plane will be established during the initial build by shimming, since the multilam connector can accommodate only a very limited misalignment.

Figure 4.2.4-1. Poloidal Field Coil Assemblies

Figure 4.2.4-2. Band Tensioning Device for Poloidal Field Coils

Figure 4.2.4-3. Multilam Connector

Detailed specifications for the ohmic heating field coil and the equilibrium field coil are listed in Tables 4.2.4-1 and 4.2.4-2, respectively.

Cooling water to the coil conductors located inside the TF coil centerline is from sub-headers supported by the coil support arms. These sub-headers are connected to the main risers via a common tie line; with this arrangement a minimum number of disconnects will have to be made if it is necessary to remove sections of the coil assembly.

Cooling water to the coil conductors located outside the TF coil centerline is routed from the main cooling water headers since these lines will remain with the device half when the device is parted for maintenance.

TABLE 4.2.4-1 OHMIC HEATING COIL SPECIFICATIONS

Parameter	Units	Value
Ampere turns (for 2.5 MA plasma current)	MA _t	8.24
Total turns		206
Max. design current (2.5 MA plasma current)	kA	40
Max. current density (2.5 MA plasma current)	kA/cm ²	3.48
Inductance	H	.027
Mutual inductance to plasma (2.5 MA)	μH	136
Ohmic-to-equilibrium field mutual inductance	mH	-.144
Resistance @ 328°K (55°C); Leads not included	mΩ	25
Insulation test voltage	kV	65
Total copper mass (based on density of 8.93 cm ⁻³ ; includes Hybrid turns)		14,750

TABLE 4.2.4-2 EQUILIBRIUM FIELD COIL SPECIFICATIONS

Parameter	Units	Value
Max. ampere turns (2.5 MA plasma current)	MA _t	4.48
Turns		56
Max. design current (2.5 MA plasma current)	kA	80
Max. current density (2.5 MA plasma current)	kA/cm ²	3.19
Inductance	mH	6.25
Field per unit coil current	mT/kA	3.75
Mutual inductance to plasma (2.5 MA plasma current)	μH	-53
Ohmic-to-equilibrium field mutual inductance	mH	-.144
Resistance @ 328°K (55°C); Leads not included	mΩ	8.6
Peak transient coil voltage	kV	20
Insulation test voltage	kV	65
Total copper mass (based on density of 8.93 g cm ⁻³ ; does not include hybrid turns)		23,850
Note: Eight turns for changing index of curvature not included.		

Design Alternates

The reference design parting plane electrical connectors (Multilam) have been subjected to preliminary cyclic testing. While this design looks promising, further testing is required to assure that it is suitable for this application.

Other connector concepts should be evaluated during the next phase of the program. Among these are a simple mechanical (bolted) connector, a pressurized connector, and a connector which utilizes a "Nitinol" actuated clamping device.

The steel tension band around the outer OH and EF coils may not be acceptable due to the potential for circulating eddy currents. If this is the case, consideration should be given to a fiber reinforced non-metallic band.

Since the central column of the device has been eliminated to provide space for insertion of an iron core, it may be possible to take advantage of the void created in the center of the device for routing coolant to the central section of the poloidal coils. The primary advantage of routing coolant lines through this area is the added capability for maintenance from the basement. This arrangement would require a hydraulically operated floor plug in the center of the device, which could be withdrawn to provide access from below.

4.3 ELECTRICAL SYSTEMS

The electrical systems for TFTR take power from the commercial 138 kV transmission line, transform and distribute it, and store and convert it to meet the tokamak requirements. The total electrical facilities are divided into five systems:

- The Energy Supply and Distribution System converts the incoming 138 kV power to 13.8 kV and 4.15 kV and distributes the power over the site. This system provides the energy storage by means of motor-generator-flywheel sets which permit averaging the tokamak pulsed loads over the five-minute operating cycle;
- The Toroidal Field Pulsed Energy Conversion System converts the ac output of the generators into controlled dc at the current level required for the experiment at the TF coils;
- The Ohmic Heating Pulsed Energy Conversion System converts the ac output of the generators into controlled dc at the OH coils. It also provides for a high reverse voltage for plasma initiation, rapid current reversal to build the plasma current and temperature, and rapid shut-down for vacuum vessel protection;

- The Equilibrium Field Pulsed Energy Conversion System converts the ac output of the generators into controlled dc at the EF coils, and provides the ability to ramp the EF coil current to follow the plasma current build-up and maintain the plasma position. It also provides the capability to provide a large current increase over 20 to 30 milliseconds to compress the plasma;
- The Central Instrumentation and Control System provides the centralized control functions, the control room monitoring provisions, and the data processing and display facilities for the experiments.

4.3.1 ENERGY SUPPLY AND DISTRIBUTION SYSTEM

Energy requirements for the operation of the TFTR are met with electric power from the local public utility system. The Energy Supply and Distribution System (ESDS) provides the interface with the public utility and distributes power at the proper level and in the required quantity to accommodate the load schedule of the TFTR.

Design Considerations

Electric power for the operation of the TFTR is supplied at 138 kV, 60 Hz. The ESDS is required to receive power at this level, transform it to the proper levels for use, and distribute it to the using locations. Since the major loads are pulsed in nature, the ESDS is also required to provide for energy storage and release, while maintaining an essentially steady load on the utility system. The ESDS must incorporate provisions to ensure the safety of personnel and the protection of equipment from damage due to off-normal conditions, and include a continuity of service during intervals of loss of power from the utility supply.

Considering the cost and schedule objectives of the TFTR project, it was determined that maximum use should be made of utility-grade commercial equipment, making the selection of ratings, sizes, capacities, etc., in accord with commercial standards, whenever this could be done without involving compromise of TFTR performance requirements. The use of standard voltages and frequencies would permit the widest possible use of utility power directly, and the substitution of utility power for locally generated power when the generators are not in use.

The preference for commercial ratings influenced the choice of equipment for meeting the energy storage requirement. While other means of short-time energy storage are possible, storage in mechanical rotation is recognized as being readily achieved and flexible in application. By means of a motor driven flywheel-generator combination, energy can be drawn at a constant rate from the utility to accelerate the flywheel, and then withdrawn in a brief interval at a high rate by allowing the flywheel to drive the generator, without reflecting the power peak into the utility system. Numerous prior installations verify the feasibility of this energy storage method.

Several generator-flywheel configurations were examined. Early consideration was given to homopolar generators, high-speed horizontal generators, and both high-speed and low-speed vertical generators, all with associated flywheels for supplying the pulsed loads of the TFTR. Homopolar generators of the required rating were not available commercially and would require considerable development time and expense. High current and high dc voltage ratings require large numbers of brush collectors in series and, in operation, they would require constant attention to prevent possible flashovers. For those reasons, the homopolar generator was not considered feasible for the TFTR. High-speed horizontal generators could be obtained commercially for the TFTR application, but the stored energy requirements could be met only by having several horizontal flywheels with associated bearings connected to the generators. In addition to the long length and consequent increased building space requirements, a protective housing would be required to absorb missiles from the high-speed flywheels in the event of a catastrophic failure. The high-speed vertical generator and flywheel assembly was also considered for the TFTR project, but would require additional development time and costs to design a unit for the large pulsed load and stored energy requirements of the TFTR.

The low-speed vertical generator with laminated steel-rim flywheel design overcomes the disadvantages of the preceding systems by making use of present day commercial design technology and components used in the manufacture of large water-wheel generators. While more steel of lower strength is used than in the high-speed generator and flywheel designs, the overall cost is lower. Use of steel laminations to construct the flywheel minimizes the possibility of a minor flaw propagating throughout the whole assembly, since it is confined to one lamination. In addition, locating the generator-flywheel assembly vertically in a containment pit greatly minimizes the hazard from missiles in the unlikely event of a catastrophic failure.

Early studies also indicated that optimized costs of the motor-generator-flywheel-rectifier combinations used in the TFTR energy supply system resulted if the speed range of operation was 100% to 70% during application of the pulsed loads, and if the corresponding generator output frequency varied from about 90 to 60 Hz. Commercially available equipment designed for 60 Hz operation could then be optimally used, since the maximum pulsed load occurs at 70% speed corresponding to 60 Hz operation.

The low-speed vertical motor-generator-flywheel sets are similar in construction to hydraulic turbine-driven generating sets and thus utilize design, manufacturing, and assembly-at-site techniques which have been long established. Commercial designs have inherent overspeed capability, enabling the attainment of the 90 Hz speed without the compromise of conservative safety factors. The use of generators of conventional design enhances their adaptability to the needs of other loads in the long range future.

Several configurations of the motor-generator subsystem were considered in the course of arriving at the two-generator arrangement. Investigation of the several configurations in light of: a) capital equipment costs, b) desired delivery schedule, and c) application of equipment of proven design, weighted the selection positively in favor of

two generators. The two-generator arrangement has less flexibility than would a four- or six-generator arrangement; however, the operating advantage foreseen with the added flexibility is insufficient to justify the cost increase involved. The power requirements necessitate the use of 13.8 kV in order to hold switchgear components within the range of established commercial designs. The speed range of the generating sets produces an output frequency in the range of 90 to 60 Hz which commercial circuit breakers and transformers can accept, minimizing the need for any design development effort on these items.

A summary of the large pulsed loads supplied from the generator is given in Table 4.1.10-1. The peak power requirement occurs during the 2.5 MA plasma experiment "flat top" interval. To these pulsing loads, approximately 10 MW for the auxiliary loads powered from the 4.16 kV system, and 6 MW for system losses must be added to obtain the total power requirement for TFTR. The major auxiliary loads are identified in Table 4.3.1-1.

Reference Design Description

The ESDS interfaces with the utility power system at an existing substation near the TFTR area. Available power capacity is adequate to accept up to 50 MVA peak demand from the TFTR project. The total energy input is divided between two distribution systems, one serving the pulse-type loads and the other the more constant auxiliary loads. The existing 138 kV oil circuit breaker will be used for fault interruption between the utility and TFTR systems.

For location in the substation, the ESDS provides a 138 kV line terminal, a dual secondary power transformer and switchgear equipment for protection and control of the medium voltage distribution circuits. The 13.8 kV transformer secondary provides power for the pulsed loads and the auxiliary loads are served from the 4.16 kV secondary winding. Capacity at 13.8 kV is approximately twice the 4.16 kV rating. Figure 4.3.1-1 shows a single-line diagram of the ESDS. The transformer 138 kV winding is delta connected and is protected by lightning arrestors from surges incoming from the utility transmission system. The 13.8 kV secondary winding is wye connected, and its neutral is grounded through a resistance to provide ground fault current for relaying and to minimize transient overvoltages during arcing ground faults. The 4.16 kV winding is also wye connected with similar ground fault protection. The secondary windings feed the TFTR facility through metal clad bus and 15 kV class outdoor switchgear having 500 MVA interrupting capacity circuit breakers for protection of the transformer.

At the motor-generator building, the incoming 13.8 kV power is divided between two sets of indoor metal clad switchgear and motor control equipment, each of which is associated with the motor portion of a motor-generator set. In each switchgear assembly, incoming power is applied through a totalizing circuit breaker to a 2000 ampere bus; the bus distributes power to the other sections of the assembly. Circuits for the distribution of power for starting, accelerating and pulsing operation of the motor are provided. The power for generator excitation is also provided.

TABLE 4.3.1-1 MAJOR AUXILIARY ELECTRICAL LOADS

	Total-HP	Total kVA (Including Appropriate Power Factor)	Maximum Simultaneous kW Demand
<u>TF, OH, and EF Field Coil Cooling Water</u>			
Main Circulating Pumps	400	400	400
Chiller Compressors	4000	4000	3200
Bypass Heater	2800	2800	
Bypass Pump	15	15	15
SCR Cooling Pump	50	60	50
<u>Neutral Beam Injector Cooler</u>			
Chiller Compressor	10	10	8
Circulating Pump	5	5	4
<u>Tritium and Non-Tritium Gas Delivery</u>	3	3	3
<u>Tritium Cleanup Systems</u>			
Fuel Vault		3	3
Torus		15	
<u>Zirconium Cannister Activation</u>		10	
<u>Computer Complex</u>		90	85
<u>Facility Cooling Water System</u>			
Supply Pump	1100	1100	911
Recirculating Pump	680	680	564
Return Booster Pump	500	500	414
<u>Liquid Waste Disposal System</u>	13	13	11
<u>TFTR Cell Ventilation</u>			
Air Conditioner Chiller	250	250	
Room Heater and Humidifier (Air Heater)		1600	
Exhaust Blower	18	18	15
Inlet Fan	18	18	15
<u>Transfer Bay Ventilation</u>			
Air Conditioner	100	100	
Room Heater (Air Heater)		700	
Blowers	15	15	12
<u>Hot Cell Ventilation</u>	14	14	12
<u>Building Load</u>			
Air Conditioning		1900	1900
Lighting	1300		1300
<u>Uninterruptible Power Supply</u>		150	150
TOTAL SIMULTANEOUS DEMAND			9072

Figure 4.3.1-1 Single Line Diagram. ESDS

The wound-rotor driving motor for the motor-generator set is controlled by a slip regulator composed of a current controller and a liquid rheostat. The slip regulator limits the starting inrush current and maintains motor current at a steady value during acceleration cycles. The current controller prevents any rise in motor load during a power pulse. Rotor circuit power losses are transferred to the facility cooling water system via a heat exchanger in the liquid rheostat.

Each motor-generator-flywheel combination provides energy storage of 4500 MJ.

Motor torque is effectively limited at its nominal 100% rating and, when a power pulse is required, all the energy required in excess of the motor rating is drawn from the flywheel by allowing the MG set speed to decrease. The generator exciter has adequate capacity to maintain generator voltage at 100% for MG set speeds of 70% or higher. After the power pulse, the torque produced by the motor causes the MG set speed to increase, so that it returns to nominal 100% speed in approximately 4-1/2 minutes.

Power delivered by each generator is brought to independent switchgear assemblies for distribution to the TF, EF, and OH coil circuits, and to the neutral beam supply system. Feeder circuit breakers of adequate short circuit interrupting capacity are provided for each coil circuit to permit operating flexibility and to obtain isolation of faults. An input from the 13.8 kV utility power is also brought to this switchgear, to provide (alternatively) for energizing the neutral beam system from utility power when desired. Interlocking prevents parallel connections between the generator and utility buses.

While operating the generators on a common bus has been contemplated, the resultant short circuit capacity would exceed the rating of presently available circuit breakers. An equalizing bus could be provided in the future for paralleling generators.

Control of the pulse power equipment is exercised from a benchboard in the motor-generator building control room. The benchboard will contain status displays, pilot control devices and alarms by which the operator will be able to monitor performance of the ESDS and control its major functions. Devices will be arranged along the face of the board in relation to representation of equipment in mimic bus depicting the ESDS. Instruments and meters on the board will permit the operator to be aware of loading conditions in principal circuits and, in conjunction with the alarm annunciators, to be aware of overloads or other off-normal conditions should they occur.

The arrangement of equipment in the motor-generator building is shown in Figure 4.3.1-2.

The ESDS distributes power at 4.16 kV to power centers at several locations within the TFTR area, for operation of auxiliary equipment and for usual building loads such as lighting and air conditioning. The auxiliary loads consist of chillers, pumps, compressors, heating, etc., functions which must be supplied on a continuous basis to support operation of the facility. The power centers contain transformers to reduce power to 480 V, and feeder breakers for circuit protection and control.

1. Motor Control Switchgear
2. M-G Set
3. Generator Output & Load
Distribution Switchgear
4. Control
5. Uninterruptible Power
6. Standby Power
7. Generator Exciter
8. Motor Speed Control
- F. Future

Figure 4.3.1-2. TFTR Motor Generator Power Supply Complex

Standby power is provided by a diesel-driven, 1500 kW, 4.16 kV generator, which starts automatically upon loss of utility power. Output of this set is distributed through the 4.16 kV system so that loads which must remain in operation during the utility outage can be supplied for an extended period. Fuel for the engine is stored locally in sufficient quantity to cover continuous operation at nominal rated full load for 1 week.

Power requirements for safety circuits, critical instrumentation and control functions, requiring the highest service continuity, are served by an uninterruptible power supply system. This system has dual power sources - the utility system and a 125 V storage battery. The battery will supply power automatically without any interruption to the system output when a disappearance of utility power occurs. The battery is sized to furnish 150 kW for 5 minutes on the basis that either utility power will have returned or the standby generator will pick up the critical loads before expiration of the 5-minute interval.

Ratings of ESDS equipment are given in Table 4.3.1-2.

TABLE 4.3.1-2 ESDS EQUIPMENT RATINGS

Main 138 kV Power Transformer (1)	40 MW, 50 MVA Total
Motor-Generator-Flywheel Sets (2)	16,500 HP, Each Motor 330 MW, 475 MVA, Each Generator 4500 MJ Total, Each Flywheel 2250 MJ Useful
Diesel Generator (1)	1500 kW, 1875 kVA, 60 Hz, 4.16 kV
Uninterruptible Power Supply (1)	150 kW, 480 V, 60 Hz for 5 Minutes
Breakdown Oscillator (1)	100 kW, 480 V, 100 kHz
Generator Exciters (2)	4200 kW Each Generator 13.8 kV, 60 Hz
Neutral Beam Filaments, arc	5170 kW at 13.8 kV

Status of Design Alternatives

The conceptual design of the energy supply and distribution system contains equipment with ratings that presently meet envisioned TFTR load cycle requirements with economy and reliability. However, it is recognized that the application of commercially available

equipment to unusual load cycles may result in unforeseen stresses and wear on the equipment that might reduce its reliability and availability below that anticipated. In addition, while the TFTR has been designed to produce plasma and neutral beam injection conditions favorable to producing fusion when supplied by specified pulsed power from the motor-generator-flywheel system, it may be found desirable in the future to increase the power to extend the fusion reaction time. Therefore, consideration must be given to design alternatives that can provide some additional margin of operational reliability and pulsed loading capability.

One of the major considerations is the motor-generator-flywheel arrangement consisting of two sets, both of which are required to operate the Tokamak device at its design rating. These two units should provide the specified electrical output with the high reliability typical of similar units in service on electric utility power systems. While it is anticipated that these units will operate reliably, the system design provides space in the motor-generator building for a third identical motor-generator-flywheel set and its associated equipment, in the event improved reliability and increased capacity are needed.

Space has also been provided for additional circuit breakers to permit switching the spare motor-generator-flywheel set to either of the two load buses when one motor-generator-flywheel set is taken out of service for maintenance or repair.

Looking to the future, when more pulsed power may be needed for the present Tokamak device for extended operation, or for a future advanced device that may require more pulsed power, the space provided for the third identical set and its associated switchgear would permit it to be easily added. A common bus for all three machines could be established, using reactors between generators, to keep short-circuit current within the rated interrupting capacity of the existing breakers.

4.3.2 TOROIDAL FIELD PULSED ENERGY CONVERSION SYSTEM

Design Considerations

In each experiment the greatest energy is supplied to the toroidal field (TF) coil to generate the magnetic field which confines the plasma. The rated peak value of the TF current is 73.2 kA, and the nominal TF coil inductance is 0.475 H. Thus, the energy which must be stored in the coil is $0.5 \times 0.475 \times (73.2)^2 = 1273$ MJ. Even more energy is dissipated as heat by the resistance of the TF coil during the times when current is being built up, regulated for the duration of the experiment, and then decreased to zero. This dissipated energy is strongly dependent on the speed with which the current can be increased and decreased, which in turn depends on the amount by which the available voltage exceeds the steady-state IR drop ($73.2 \text{ kA} \times 0.063 \text{ ohm} = 4.6 \text{ kV}$).

The Toroidal Field Pulsed Energy Conversion System (TFCS) rectifies the ac from the flywheel generators and controls the TF coil current. A tradeoff study between the costs of rectifier voltage rating and flywheel energy rating indicated that the appropriate dc operating voltage is around 6 kV. The corresponding time to build the current up to its rated value is approximately 12 seconds, and the total energy is approximately 3700 MJ. Twelve-pulse rectifiers were selected to produce an acceptable current ripple in the TF coil with no additional filtering.

Because of the large power demand of the TFCS relative to the generator ratings, the harmonic currents which it draws will cause considerable distortion of the voltage waveform at the generator terminals. This distortion is of particular concern because of its possible effects on the other rectifiers which are supplied from the same generators. Topical Report WFPS-TME-006, "Analysis of the TFTR Toroidal Field Power Supply and its Interactions with Other Loads", contains a more detailed discussion of this subject.

Reference Design Description

The configuration of the reference design for the TFCS is shown in Figure 4.3.2-1, and key parameters are summarized in Table 4.3.2-1. Each generator supplies four 12-pulse transformer/rectifier modules, each having nominal ratings of 750 V and 73.2 kA dc. The series connection of these modules on the dc side facilitates full-power testing of TFCS. If the load is deliberately shorted, power can be exchanged between the two generators by operating one 4-module group as a rectifier while the other operates as an inverter.

Upon command from the Central Instrumentation and Control System (CICS) some 12 seconds before plasma initiation, the TFCS will apply full voltage to the TF coil to increase the current from zero to a preset value. When this value is reached, the thyristor firing angles are controlled by a closed-loop current regulator to maintain the current constant for the remainder of the experiment. If all of the rectifier modules are phased back by the same amount during this interval, the power factor will be less than $4.6 \text{ kV} / 6 \text{ kV} = 77\%^*$. This poor power factor, together with the simultaneous loading by the other three pulsed energy conversion systems, causes an excessive MVA demand on the generators.

The MVA demand of the TFCS can be reduced considerably by phasing back only one rectifier module on each generator, a process which is known as sequential control. Thus the thyristor firing circuits for the rectifier modules will be designed to be individually controllable, and sequential control or its equivalent will be used to maximize the power factor and minimize the MVA demand from the generators.

* Transformer and ac cable reactance cause a further reduction in the power factor seen by the generators.

Figure 4.3.2-1. Reference Design of the TF Pulsed Energy Conversion System (SD-4D)

**TABLE 4.3.2-1 TOROIDAL FIELD PULSED ENERGY CONVERSION SYSTEM
SUMMARY OF PARAMETERS**

Rated peak dc current	73.2 kA
Rated dc voltage under load	6 kV
Rated dc power to load	440 MW
Maximum apparent power from generators	611 MVA
Maximum energy from flywheels per 5-minute interval	3640 MJ
Nominal ac input voltage (rms)	13.8 kV
Maximum ac input voltage (crest)	32 kV
Converter configuration	
8 series 12-pulse modules, shared equally by two generators	
Reversible voltage for de-energizing coil	
Sequential control or equivalent to maximize power factor	
Grounded through resistance at center tap of TF coil	
Transient dc voltage rating of rectifier	20 kV
Spark gap setting	16 kV
Hi-pot test to ground	30 kV
TF coil resistance (nominal)	63 mΩ
TF coil inductance (nominal)	*475 mH
Maximum current ripple (peak-to-peak)	0.1%
Current regulation	0.5%
Maximum ac cable length (transformer to rectifier)	75 ft
Maximum dc bus length (rectifier to TF coil)	500 ft
<p>*NOTE</p> <p>Recent calculations of final conceptual design configuration yield results as high as 510 mH. Power supply design allows margin for up to 522 mH.</p>	

At the conclusion of each experiment, the TFCS will receive a command from the CICS to begin operation as an inverter. In this mode, as much as possible of the energy stored in the TF inductance will be returned to the flywheels via the generators (which act as motors). In this way, it is anticipated that some 600 MJ or more can be recovered from each experiment, thereby reducing the heat load on the cooling system and also reducing the net input power demand.

The TF coil is grounded through a resistor at its center tap to provide balanced operation and to limit ground fault currents to a low value. A spark gap set for 16 kV provides over-voltage protection for both the rectifier and the TF coil, which are each designed to withstand at least 20 kV between the dc terminals. A crowbar and its associated resistor permit the energy stored in the TF coil to be dumped rapidly and safely. Any fault which necessitates rapid shutdown will cause the crowbar to be triggered.

Although the TFCS will normally operate for about 18 seconds out of each 5 minutes during experimental runs, it will be designed to allow continuous operation at 20% of rated peak current for discharge cleaning of the vacuum vessel.

Status of Design Alternatives

The studies which have been performed to date indicate that the present reference designs for the generators and the TFCS are not likely to cause serious interactions with the other rectifiers. However, if such interactions become evident at a later date, balancing reactors or shunt capacitors across the ac lines provide possible solutions that could be implemented. Shunt capacitors have the additional advantage of partially compensating for the lagging power factors. Normal designs for tuned harmonic filters cannot be used because of the variable source frequency, which causes the harmonic frequencies to vary correspondingly.

The reference design for the TF crowbar uses thyristors. However, a hybrid crowbar using a mechanical contactor together with a thyristor switch is a design alternative.

4.3.3 OHMIC HEATING FIELD PULSED ENERGY CONVERSION SYSTEM

Design Considerations

Plasma initiation and ohmic heating are achieved by the Ohmic Heating Pulsed Energy Conversion System (OHCS) which supplies a controlled current to the OH coil. This coil acts as the primary of a transformer of which the plasma is the secondary. The current in the OH coil is raised to a peak rated value of 40 kA by the OH rectifier during the several seconds immediately preceding each experiment. The current is then interrupted from the rectifier by a switch and commutating circuit which allow the OH coil voltage to rise rapidly to approximately 31 kV, thereby inducing a plasma initiation voltage of 150 V. At this voltage, the coil current is transferred to a

reversing capacitor which resonantly reverses the current polarity in approximately 0.125 second. In the meantime, a reversing switch has operated to reverse the polarity of the rectifier voltage and allow the rectifier to maintain the reverse current until the end of the pulse.

At the end of the pulse, the commutation process is reversed to induce a reverse voltage into the plasma and rapidly reduce the plasma current.

Since the OH and EF coils share common turns and are inductively coupled with each other and with the plasma, there is considerable interaction between these three current loops. This interaction must be taken into account in analyzing the current responses and in designing the OHCS and EFCS control circuits.

Although the OHCS has the smallest power rating of the three field coil power supplies, it is by far the most complex because of the combined high voltage and high current requirements involved in its switching operations. In theory, the OH rectifier could be designed as a dual converter to provide both polarities of current and rated to withstand the 31 kV voltage transient. The corresponding rectifier rating would be $2 \times 40 \text{ kA} \times 31 \text{ kV} = 2480 \text{ MVA}$, compared to the actual need for only $40 \text{ kA} \times 1.5 \text{ kV} = 60 \text{ MVA}$ to power the coil.

Provision of a reversing switch allows a standard, unidirectional-current rectifier to supply the OH coil current both before and after reversal. Furthermore, the interrupting switch can be designed to isolate the rectifier from the OH coil circuit before the high-voltage transient for plasma initiation occurs. However, a switch which can interrupt 40 kA and block 31 kV within 1 ms does not exist and must be developed.

Reference Design Description

The configuration of the reference design for the OHCS is shown in Figure 4.3.3-1, and key parameters are summarized in Table 4.3.3-1. Each generator supplies one 12-pulse transformer/rectifier module having a nominal rating of 750 V and 40 kA dc. The series connection of these modules on the dc side facilitates full-power testing of the OHCS. If the load is deliberately shorted, power can be exchanged between the two generators by operating one module as a rectifier while the other operates as an inverter.

Once the desired initial current is established in the OH coil, the coil voltage is clamped at a low value by "freewheeling" the rectifier. The reference design utilizes an array of vacuum contacts for the interrupting switch. These contacts are then opened. The resulting arc is extinguished by a commutating circuit which causes the arc current to decrease to zero. A saturable reactor slows the final di/dt in the contacts to allow them to recover voltage-blocking capability.

Figure 4.3.3-1. Reference Design of OH Pulsed Energy Conversion System (SD-4C)

TABLE 4.3.3-1 OHMIC HEATING PULSED ENERGY CONVERSION SYSTEM
SUMMARY OF PARAMETERS

Rated peak dc current	40 kA
Rated dc voltage under load	1.5 kV
Rated dc power to load	60 MW
Maximum apparent power from generators	85 MVA
Maximum energy from flywheels per 5-minute interval	106 MJ
Nominal ac input voltage (rms)	13.8 kV
Maximum ac input voltage (crest)	32 kV
Converter configuration	
2 series 12-pulse modules, shared equally by two generators	
Reversible voltage for deenergizing coil	
Grounded through resistance at center tap of OH/EF coil	
Transient dc voltage rating of rectifier	6 kV
Spark gap settings	
Across rectifier	3.5 kV
Across OH coil	40 kV
Hi-pot test to ground	65 kV
Current-reversing capacitor	7.5 MJ
Plasma initiation voltage for 3-5 ms	160 V
Rise time of initiation voltage	1 ms
Time for current to reach zero during reversal	50 ms
OH coil resistance (nominal)	21 mΩ
OH coil inductance (nominal)	27 mH
Maximum voltage ripple (peak-to-peak)	10%
Current regulation	0.5%
Maximum ac cable length (transformer to rectifier)	75 ft
Maximum dc bus length (rectifier to OH coil)	500 ft

While the coil current is flowing in the commutating circuit, the voltage rises rapidly to -31 kV for plasma initiation. At this point, a switch is triggered to transfer the coil current from the commutating circuit to the reversing capacitor. The resistance in series with this capacitor permits the plasma initiation voltage to be maintained for several milliseconds. Then part of this resistance is shorted by triggering a switch until the coil current drops to zero. At this instant the energy stored in the coil is zero and the energy stored in the reversing capacitor is at its maximum value.

The inductive energy stored in the OH coil at rated peak current is approximately 21.6 MJ. For reasons of economy, the energy stored in the capacitor in the reference design has been limited to 7.5 MJ. Therefore, the resistors must be designed to dissipate part of the stored energy during the interval of approximately 50 ms while the current decreases to zero.

A third switch is triggered to transfer efficiently the energy in the reversing capacitor to the OH coil and establish a reverse current of approximately -20 kA. Then the alternate poles of the reversing switch are triggered and the coil current transfers once again to the rectifier. Thereafter the rectifier can be controlled to maintain the current at a constant level or increase it to -40 kA during the remainder of the pulse.

At the end of the pulse, the current-reversal process is repeated with a second commutation circuit in order to rapidly decrease the plasma current.

The OH rectifier in the reference design has a transient voltage rating of 6 kV and is protected by a spark gap which is set for 3.5 kV. A second spark gap is set for 40 kV to protect the OH coil insulation.

Crowbars across the OH coil and the reversing capacitor permit the stored energy to be dumped rapidly and safely. Any fault which necessitates rapid shutdown will cause the crowbars to be triggered.

A filter in the dc circuit will limit the peak-to-peak ripple voltage to 10% of the rated voltage.

Single point grounding will be through a resistor to limit ground fault current.

In the reference design, the switches in the reversing capacitor circuit all utilize ignitrons. This choice was made because ignitrons are available with high voltage ratings and can carry high pulse currents for the short intervals involved in the experiment. Therefore, they are expected to be somewhat more economical than alternate means of satisfying the switching requirements.

Status of Design Alternatives

The switches involved in the reference design of the OHCS have been selected to be as economical as possible with reasonable certainty of feasibility. In some cases, these choices involve tradeoffs between initial costs and costs of replacement and maintenance. For instance, the vacuum contacts in the interrupting switch will undergo erosion from arcing, and the operating mechanism will suffer mechanical wear from repeated use. Therefore, the initial stages of the development program for this switch should seek to establish not only the feasibility of the approach but also the replacement rate and costs. In the reversing capacitor switches, it may be advantageous to utilize hybrid switches with ignitrons or thyristors to switch rapidly and mechanical contacts to carry the subsequent current.

These possibilities provide alternative approaches should the reference design approaches encounter unforeseen difficulties.

A potential requirement for increased plasma initiation voltage has been identified. This may require doubling the OH coil reversing voltage. If this requirement is confirmed, a significant redesign may be required of both power supplies and coils.

4.3.4 EQUILIBRIUM FIELD PULSED ENERGY CONVERSION SYSTEM

Design Considerations

The radial position of the plasma within the torus is determined by a poloidal magnetic field. For a fixed plasma position, this field must be proportional to the plasma current. The Equilibrium Field Pulsed Energy Conversion System (EFCS) supplies the direct current to the EF coil to generate this field and control it in accordance with variations in the plasma current. Because of the rapid initial rise in plasma current, the dc voltage which is available from the EFCS must be considerably greater than the steady-state IR drop in the EF coil.

The EFCS also contains an energy-storage capacitor which permits the EF current to be changed even more rapidly when desired for plasma compression experiments.

Since the EF and OH coils share common turns and are inductively coupled with each other and with the plasma, there is considerable interaction between these three current loops. This interaction must be taken into account in analyzing the current responses and in designing the EFCS and OHCS control circuits.

Reference Design Description

The reference design of the EFCS will provide a dc voltage up to 3 kV, which will raise the current in the EF coil from zero to its rated peak value of 80 kA in less than 0.2 second. The configuration of the EFCS is shown in Figure 4.3.4-1 and key parameters are summarized in Table 4.3.4-1. Each generator supplies two 12-pulse transformer/rectifier modules, each having nominal ratings of 750 V and 80 kA dc. The

Figure 4.3.4-1. Reference Design of the EF Pulsed Energy Conversion System (SD-4E)

TABLE 4.3.4-1 EQUILIBRIUM FIELD PULSED ENERGY CONVERSION SYSTEM SUMMARY OF PARAMETERS

Rated peak dc current	80 kA
Rated dc voltage under load	3 kV
Rated dc power to load	240 MW
Maximum apparent power from generators	318 MVA
Maximum energy from flywheels per 5-minute interval	98 MJ
Nominal ac input voltage (rms)	13.8 kV
Maximum ac input voltage (crest)	32 kV
Converter configuration	
4 series 12-pulse modules, shared equally by two generators	
Reversible voltage for deenergizing coil	
Sequential control or equivalent to maximize power factor	
Grounded through resistance at center tap of EF/OH coil	
Transient dc voltage rating of rectifier	12 kV
Spark gap settings	
Across rectifier	7 kV
Across EF Coil	25 kV
Hi-pot test to ground	65 kV
Compression capacitor (connected in series with rectifier to give 20 kV across coil)	7.5 MJ
EF coil resistance (nominal)	7.4 mΩ
EF coil inductance (nominal)	6 mH
Maximum voltage ripple (peak-to-peak)	10%
Current regulation	0.5%
Maximum ac cable length (transformer to rectifier)	75 ft
Maximum dc bus length (rectifier to EF coil)	500 ft

series connection of these modules on the dc side facilitates full-power testing of the EFCS. If the load is deliberately shorted, power can be exchanged between the two generators by operating one 2-module group as a rectifier while the other operating one 2-module group as a rectifier while the other operates as an inverter.

Once the plasma is initiated by the breakdown oscillator and the high initial voltage on the ohmic heating coil, the EF coil current must increase from zero to track the increasing plasma current. The EF current reference signal is provided by the Central Instrumentation and Control System either as a preprogrammed ramp or from measurements of the plasma current. Because of the coupling between the EF current and the OH and plasma currents, the EF voltage must vary rapidly over a wide range during this interval. Thus both the thyristor firing control circuits and the inherent transient behavior of rectifiers play an important role in determining the dynamic behavior of the EFCS.

When the plasma current reaches the desired level, the EFCS must thereafter regulate the EF current at a nearly constant value. Since the IR voltage required to maintain this constant current is at most $80 \text{ kA} \times 7.4 \text{ m}\Omega = 600 \text{ V}$ compared to the available 3 kV, the EF rectifier would normally operate with a very poor power factor of less than $600/3000 = 20\%$ during this interval. Although the MW demand is not excessive, the generators have not been rated to tolerate the corresponding MVA demand from the EFCS while the other three pulsed energy conversion systems are also active. This peak MVA loading is of particular concern because its duration is too short to allow the generator regulator to compensate for it. The thyristor firing circuits for the EF rectifier modules will therefore be designed to be individually controllable. Sequential control or its equivalent will be used to maximize the power factor and minimize the MVA demand from the generators during reduced-voltage operation.

For experiments involving plasma compression, the EFCS contains a compression capacitor which can be precharged with 7.5 MJ of energy. Following the establishment of an initial plasma current, this capacitor can be momentarily inserted in series with the EF rectifier to provide a total of up to 20 kV across the EF coil. Both the rectifier and the capacitor supply energy to rapidly increase the coil current and shift the plasma position. When the energy of the capacitor is fully used, it is removed from the circuit by the blocking diodes.

A filter in the dc circuit will limit the peak-to-peak ripple voltage to 10% of the rated voltage.

A spark gap set for 25 kV provides overvoltage protection for the coil. A second spark gap is set for 7 kV to protect the rectifier, which is rated to withstand transient voltages of 12 kV. Crowbars across the EF coil and the compression capacitor permit the stored energy to be dumped rapidly and safely.

Although the EFCS will normally operate for less than 2 seconds out of each 5 minutes during normal experimental runs, it will be designed to operate with more rapid pulsing at reduced current levels for discharge cleaning of the vacuum vessel.

The reference design of the EFCS utilizes ignitrons in the crowbars and in the compression capacitor switch because it is anticipated that they will be more economical than other switches with comparable ratings.

Status of Design Alternatives

Switches using thyristors or hybrid switches using mechanical contacts provide possible design alternatives for the EF crowbars and/or compression capacitor switch. These alternatives could be pursued if difficulties are encountered with the ignitron switches.

4.3.5 CENTRAL INSTRUMENTATION AND CONTROL SYSTEM

The Central Instrumentation and Control System (CICS) provides central control, timing and monitoring, safety and protection, data handling and display, and radiation and tritium monitors. The CICS provides capability for operation of the TFTR facility from a central control station, as depicted in Figures 4.3.5-1 and 4.3.5-2.

Reference Design Description

a. Central Control and Monitor

The Central Control and Monitor System (CCMS) is a digital computer-based system operated from a central control location to provide remote control and monitoring functions for the numerous TFTR systems located throughout the TFTR facility. The CCMS is shown in the functional block diagrams 4.3.5-3, 4.3.5-4 and 4.3.5-5. These diagrams indicate the functional interfaces between the CCMS and the other TFTR systems with physical separations also indicated. Central Control and Monitoring functions include the following:

- Electrical Power Control and Monitor
- Neutral Beam Control and Monitor
- Tritium Gas Control and Monitor
- Non-Tritium Gas Control and Monitor
- Tritium Cleanup Control and Monitor
- Torus Vacuum Control and Monitor
- Machine Area Water Cooling Control and Monitor
- Auxiliary Facility Services Control and Monitor

Control modes and system operating parameters are selected by the operator(s) at the control consoles in the central control room. The CCMS may operate in an automatic mode to provide the desired control and timing demand signals to each of the

DISPLAY CONSOLE NOMENCLATURE:
 4011 - TEXTRONIX GRAPHIC DISPLAY TERMINAL
 PDS-4 - IMLAC INTERACTIVE GRAPHIC DISPLAY TERMINAL
 4610 - TEXTRONIX HARD COPY

Figure 4.3.5-1. Control Room

Figure 4.3.5-3. Central Control & Monitor System

Figure 4.3.5-4. Central Control & Monitor System
 (For Gas & Tritium Cleanup Systems)

Figure 4.3.5-5. Central Control & Monitor System (For Torus Vacuum, Machine Area Water Cooling, & Aux. Facility Systems)

TFTR systems as preprogrammed functions of the operator inputs and the measured parameters and status signals. Control and data channels are provided for routing these signals between the central control station and the local stations. A central/local control and monitor interface is located at each of the TFTR system local stations.

b. Timing System

The high resolution timer shown in Figure 4.3.5-6 operates as a peripheral to the minicomputer system and includes comw special hardware. The timer settings are loaded into the timing system which produces the required timing signals without further intervention once the start signal is given. The timing system provides 64 timer channels with a resolution of 10 microseconds and a jitter not exceeding ± 0.1 microsecond. In the extreme case the set time ranges from 49 seconds before Reference Time (T_R) to 49 seconds after T_R . These channels produce pulse outputs suitable for synchronization and control of other equipment.

The timing system also provides 64 channels with a resolution of 10 milliseconds. In the extreme case the set time ranges from 49.9 seconds before T_R to 149.9 seconds after T_R . These channels provide relay outputs suitable for control of other equipment.

Figure 4.3.5-6. High Resolution Timer

The basic timing is obtained from a 1 MHz oscillator and timing chain. The individual timer channels utilize a content addressable memory to generate a pulse at the time stored in the memory. Although some of the hardware must be special, a large part is available as standard modules.

When a time setting is entered into the system, the value entered may be checked against limit values and any calculations necessary to determine the time setting may be made. Thus, when a time is set with reference to a value other than T_R , the computer determines the appropriate timer channel setting. The computer then loads the proper content addressable memory and the channel is ready for operation.

The high resolution timer hardware may be expanded in a modular fashion up to 128 channels.

c. Plasma Position Control System

Initial operation of the TFTR is planned to utilize programming of OH and EF currents. Provisions will be available to permit the addition of automatic feedback controls when the requirements and methods are developed. Proposed control circuits for both horizontal and vertical plasma position controls are described in the system description.

d. Safety and Protection System

The Safety and Protection System (SPS), as shown in the hard-wired logic diagram in Figure 4.3.5-7, is designed to minimize situations which are hazardous to personnel or equipment. The SPS monitors the conditions which exist, and in the case of undesired conditions, causes the equipment to revert to a safe condition and sounds an alarm.

e. Data-Handling System

Data Acquisition and Storage System (DAS)

The DAS is a computer-based system used to collect data during the course of the experiment and store it for later analysis and display. The DAS is comprised of four standard minicomputers and their associated peripherals as summarized in Table 4.3.5-1 and includes all required interfaces to the main data-reduction computer.

f. High-Speed Data Acquisition

One requirement which has been given particular attention is the one for rapid collection of data. A schematic for a special high-speed data-acquisition channel is shown in Figure 4.3.5-8. Six high-speed analog-to-digital converters using multiplexing permit up to 96 analog inputs to be sampled in a programmed random access manner

Figure 4.3.5-7. Safety and Protection System

TABLE 4.3.5-1 DAS COMPUTER SYSTEM SUMMARY

Computer System	Features	Function
PDP-11/40* (Or Equivalent)	PDP-11/40 Central Processor 32K Core Memory DEC Writer II Terminal Extended Arithmetic Option Floating Point Unit Disk Memory (1.2 million word) Dual DEC Cassette Transport General Purpose DMA Interface DEC System II Software System Basic, Fortran	Fast Analog Data Acquisition
PDP-11/40* (Or Equivalent)	PDP-11/40 Central Processor 32K Core Memory DEC Writer II Terminal Extended Arithmetic Option Floating Point Unit Disk Memory (1.2 million word) Dual DEC Cassette Transport Lab Peripheral System Panel and Power Supply 12-Bit A/D Converter and Multiplexer Direct Memory Access Channel 2 11-Bit D/A Converter DEC System II Software System Basic, Fortran	Gather Preprocessed DATA
PDP-11/40* (Or Equivalent)	PDP-11/40 Central Processor 32K Core Memory DEC Writer Terminal Floating Point Unit Disk Memory (1.2 million word) Dual DEC Disk Drive A/D Converter Subsystem DEC System II Software Basic, Fortran	Medium Speed Analog Data Gathering
PDP-11/40* (Or Equivalent)	PDP-11/40 Central Processor 32K Core Memory DEC Writer II terminal Extended Arithmetic Option Floating Point Unit Disk Memory (1.2 million word) Dual DEC Cassette Transport Lab Peripheral System 12-Bit A/D Converter and Multiplexer 11-Bit D/A Converter DEC System II Software Basic, Fortran	General Application
* The specification of specific vendors is not intended at this time. Reference is made to specific products which are considered typical for costing and interfacing purposes.		

HIGH SPEED DATA ACQUISITION SYSTEM

Figure 4.3.5-8. High Speed Data Acquisition System

at a rate of 800,000 points per second. This information may be loaded into a high-speed disk store. In some cases it may be preferable to use digital-storage oscilloscopes to obtain direct readings of raw data which may later be entered into the data-analysis system.

g. Data Reduction and Display System (DRDS)

The DRDS is a medium-scale computer system used to analyze and display plasma experimental data and for storage/display of operating data in a time frame suitable for determining the results of one machine-pulse operation prior to the next machine-pulse operation. The DRDS computer system is described in the summary Table 4.3.5-2 and Figure 4.3.5-9.

The peripheral equipment supplied with the DRDS computer system is completely software supported by the monitor and provides flexibility for expansion of the system. Each peripheral device is interfaced to function under interrupt control with the priority of each interrupt specified under program control.

The DRDS includes six (6) computer display terminals (Tektronix Model 4012 or equivalent) with each terminal mated to a hard-copy unit (Tektronix Model 4610 or equivalent). Four of these display terminals are used for plasma experimental data, while the remaining two terminals are used for display of important operating data, e.g., neutral beam data, etc.

h. Radiation and Tritium Monitors

The characteristics for the radiation monitoring system are summarized in Table 4.3.5-3.

A summary equipment list is shown in Table 4.3.5-4 for the complete radiation and tritium monitor system to satisfy the design requirements of Section 4.1.14.

i. Central Control Station

The central control station layout is shown in Figure 4.3.5-1, the equipment placed in the central control room listed in Table 4.3.5-5.

j. Computer and Data Storage Area

The computer and data storage area is shown in Figure 4.3.5-2. All of the computers used in the Data Handling System are assigned to the computer and data storage area. In addition, the interfaces to the control and data systems are in this area.

TABLE 4.3.5-2 DRDS COMPUTER SYSTEMS SUMMARY

Computer System	Features	Functions
*PDP-10 (Or Equivalent)	KL10 Central Processor 256K Words Memory Tape and Disk Systems Line Printer CALCOMP XY Plotter Graphics System Memory Port Multiplexer Data Line Scanner Memory Link Interfaces for PDP-10 and PDP-11's DEC System-10 Software System and License	Experimental Data Reduction and Display Operating Data Storage and Display
*The specification of specific vendors is not intended at this time. Reference is made to specific products which are considered typical for costing and interfacing purposes.

Figure 4.3.5-9. Data Reduction and Display System

TABLE 4.3.5-3 SUMMARY OF RADIATION MONITORING SYSTEM CHARACTERISTICS

Location (Quantity)	*Type	Range mr/hr	Alarm Range
Test Cell (3)	(β , γ)	0.1 to 1000	1-1000 mr/hr
Test Cell (1)	(β , γ)	10^1 to 10^5	None
*Exterior Wall of Test Cell (2)	(β , γ)	.01 to 100	0.1 to 100 mr/hr
Warm Cell (2)	(β , γ)	.1 to 1000	1-1000 mr/hr
Gallery Area (6)	(β , γ)	.01 to 100	0.1-100 mr/hr
Test Cell Chiller Room (2)	(β , γ)	.01 to 100	0.1-100 mr/hr
Diagnostic Equipment Area (2)	(β , γ)	.01 to 100	0.1-100 mr/hr
Mockup Assembly Area (2)	(β , γ)	.01 to 100	0.1-100 mr/hr
N.B. Test Cell (2)	(β , γ)	.01 to 100	0.1-100 mr/hr
*Neutron Detectors as Required			

Status of Design Alternatives

An alternate design concept now being considered for the Central Instrumentation and Control System (CICS) is an integrated data acquisition/analysis central control and monitoring system as shown in Figure 4.3.5-10. This hierarchal digital computer configuration uses two medium size computers and six minicomputers to integrate the plasma data acquisition, data analysis and storage, and the control and monitoring systems. The subsystem real time tasks will be handled by the minicomputers, with the diagnostic and operating computers providing the data pool, data storage, data reduction, and display tasks, plus providing the man-machine interface for both.

4.4 EXPERIMENTAL AREA SYSTEMS DESIGN

The Experimental Area Systems are those necessary to supply the peripheral services for the operation and maintenance of the TFTR device, and which are generally located within or in close proximity to the Test Cell. These systems are:

- Remote Handling and Maintenance System
- Tritium Storage and Delivery System
- Tritium Cleanup System
- Non-Tritium Charge Gas Delivery System
- Machine Area Water Cooling System
- Torus Vacuum Pumping System

TABLE 4.3.5-4 SUMMARY EQUIPMENT LIST FOR RADIATION AND TRITIUM MONITORS

Item	Component	Quantity	*** Supplier Model No. (or equivalent)
1	Tritium Detector	20	Johnston Laboratories Model 955-B
2	Tritium Calibrator	1	Johnston Laboratories Model CL-1
3	Tritium Remote Alarm	20	Johnston Laboratories Model RA-1
4	Tritium Recorder	7	Honeywell Class 15
5	Gamma Detector	22	Victoreen Model 857
6	Gamma Readout Module	22	Victoreen Model 856
7	Gamma Remote Alarm/Meter	22	Victoreen Model 858
8	Continuous Gross Air Particulate Monitor	2	Victoreen Model 840-1
9	Continuous Gross Air Gaseous Monitor	1	Victoreen Model*
10	Gaseous Effluent Recorder	2	Honeywell Class 15
11	Equipment to Remove THO ³ and TH(T ₂) from sample Stream		
12	Neutron Instrumentation (and other) Equipment as Required		
13	Equipment Racks	As required to package the above equipment ready for use.	
14	Cables**	As required.	

* Used in conjunction with Item 8

** Average length from detectors to readouts are about 300 feet.

*** The specification of specific vendors is not intended at this time. Reference is made to specific products which are considered typical for costing and interfacing purposes.

TABLE 4.3.5-5 LIST OF EQUIPMENT TO BE PLACED IN THE CENTRAL CONTROL ROOM

1. Experimental Device Control Operator Desk and Operations Console
2. Neutral Beam Control Console
3. Electric Power Panel
4. Fuel Gas Handling Panel
5. Vacuum Systems Panel
6. Water Cooling and N₂ and other Non-Fuel Gas Handling Panel
7. Tritium Cleanup Panel
8. Facility Monitoring Panels
 - Heating
 - Ventilation
 - Fire protection and alarms
 - Radiation monitoring and recorders
 - Site environmental recorders
 - Other recorders
9. Communications Equipment for each Operator Station
10. Shift Supervisor Operational Data Accessing Console
11. Test Director Operational Data Accessing Console
12. Timing Control Computer
13. Process Control Computer
14. Neutral Beam Control Computer

Figure 4.3.5-10A Computer Room

Figure 4.3.5-10B. Control Room

Figure 4.3.5-10C. Control Area

Basic design concepts for most of these systems have undergone relatively little changes since the original work reported in June 1974 and without any major shift in concept. However, the Remote Handling and Maintenance Systems has been continuously evolved and refined in depth and scope of the system capabilities. The Tritium Storage and Delivery Systems is now located in a separate fuel vault below the test cell. The Tritium Cleanup System consists of five separate cleanup packages including one in the torus vacuum pump discharge lines. The chiller capacity and cooling water flow rate for Machine Area Water Cooling System have been changed from time to time as the conceptual design evolved but they are 11.3 MW (3300 tons) and .227 m³/s (3600 gpm) respectively. The Torus Vacuum Pumping System consists of four (4) 61 cm mercury diffusion pumps instead of the eight (8) 41 cm mercury diffusion pumps previously used.

4.4.1 REMOTE HANDLING AND MAINTENANCE SYSTEMS

Design Considerations

In designing the Remote Handling and Maintenance System (RHMS), prime consideration was given to the functions that the system must perform in order to assemble and provide remote maintenance to the TFTR.

One function of the system is to perform preventive maintenance.

Certain parts, such as electric connections, will operate trouble-free for indefinite periods of time if they are cleaned periodically and kept free from oxide films. Filter elements in various fluid lines must be replaced routinely to avoid possible overheating; pump-motor overload and other problems associated with restricted fluid flow. Bolts used to join electrically insulated interfaces, that are required to transmit structural loads must be retorqued periodically to assure equal load distribution and eliminate loose conditions that can cause unwanted deflections of the TFTR device.

The RHMS must perform diagnostic operations as required to identify and locate malfunctions that may occur in components of the TFTR or other equipment located in the Test Cell. Electrical cables must be checked for cleanliness and proper connections. Water lines and floor areas must be inspected for wet areas that may indicate leaks. Exposed surfaces of TFTR components must be examined for unusual discolorations that may indicate overheating. Electrical checks will be made periodically to verify acceptable performance of electrical circuits and components. Vacuum lines, connections and vessels must be examined and checked for leaks if the required vacuum level cannot be obtained.

If corrective maintenance cannot be accomplished, malfunctioning parts are to be removed and replaced by substitute parts or components that have been previously checked out for acceptable performance characteristics.

If defective parts or components removed from the TFTR assembly are suitable for repair, they will be repaired in either the Hot Cell or Transfer Bays. Defective Neutral Beam (NB) Injector units will be repaired in the NB Test Cell. Some parts and components are not considered suitable for repair either because of their low cost or inherent complexity. These items will be removed and disposed, using means appropriate for the activation and/or contamination levels involved.

The RHMS equipment is used to investigate failure causes. Investigative equipment will be used in either the Transfer Bays or Hot Cell to conduct post-mortem evaluations aimed at determining failure causes.

In addition to the above functions the RHMS must be capable of transporting TFTR components between the various cells and storage areas of the TFTR Building. Floor level rail systems and cranes are used for this function.

Reference Design Descriptions

The Remote Handling and Maintenance System (RHMS) provides the hardware needed to perform numerous assembly, servicing, handling and maintenance operations. These operations are to be accomplished using remote, semi-remote or contact methods, the choice depending on the most advantageous method at the time the operation is to be done. Figure 4.4.1-1 shows the installation of remote handling hardware in the Experimental Area.

A transport system using rails mounted on the floor of the Experimental Area is used to transport heavy components between cells of the complex. Two self-propelled cars serve as vehicles. One of the cars has a turntable mounted on the chassis to permit neutral beam injector units to be turned remotely through 90° while on the car. Remote controls are provided for both cars.

Manipulators are needed whenever the radiation environment in a maintenance or handling region is beyond limits of personnel safety. Operations are conducted remotely by using overhead or sidewall manipulators. Overhead manipulators are used in the Test and Hot Cells.

Each of these overhead manipulators is mounted on a telescoping tube that moves vertically from a bridge mounted carriage that is controlled to travel over the entire floor area of the cell. Vertical travel of the telescoping tube permits the manipulator arm to be moved from the floor to the elevation of the overhead bridge.

Thru-wall manipulators are used in the Test Cell and the Hot Cell. These units are the master-slave types and permit relatively delicate operations to be performed by an operator standing on the outside of the cell while viewing the operation with closed circuit television (CCTV), and through a thick shielding window.

Figure 4.4.1-1. Experimental Area - Tokamak Fusion Test Reactor

Viewing equipment is used in conjunction with remote manipulator equipment. Shielded windows are used in the Test Cell and Hot Cell. These windows are located roughly at the level of TFTR device or somewhat above in order to enhance viewing capability. The CCTV system also provides coverage of the cell interiors and offers another system for viewing objects in the cells.

Binoculars, periscopes, borescopes, telescopes and mirrors are expected to assist in providing sufficient viewing capability.

Overhead bridge cranes are used in several locations within the Experimental Area to lift and move heavy loads. Control for the cranes is accomplished remotely. Operations are viewed through shielding windows or on CCTV.

A shielded cab is used in the test cell to permit visual inspections, take high resolution photographs, give close control to remote handling operations and to perform minor maintenance operations. The cab is supported on the overhead manipulator trolley by a large diameter thrust bearing that permits the cab to rotate in a horizontal plane through 180° in either direction as well as move in x and y directions. Viewing from the cab is provided by a downward facing shield window. The cab contains controls for the overhead manipulator as well as cameras and other inspection equipment. Figure 4.4.1-2 depicts the cab as part of the Overhead Manipulator and Inspection System.

Semi-remote equipment includes items such as protective clothing, portable shields, glove boxes, and breathing apparatus. This type of equipment often permits a time consuming operation, if done remotely, to be accomplished quickly.

Special versions of conventional hand tools are needed for a variety of servicing operations. These tools normally have adaptive features that make them suitable for use with other general purpose remote handling equipment such as manipulators. Items such as impact wrenches, cut-off saws, welding tools, portable turntables, jacks and disconnecting tools are included in this category.

Handling frames are auxiliary pieces of equipment that permit large components of the TFTR to be handled remotely. These items are compatible in physical size and weight with each component that must be handled. Examples of equipment items in this category include the TF Coil Lifting Rig, the Vacuum Vessel Transfer Rig and Vacuum Vessel (Parting Plane) Welding and Cutting equipment.

Leak detection and repair equipment is included as part of the RHMS. The vacuum system may develop leaks during the lifetime of the TFTR. Equipment capable of detecting, locating and repairing such leaks by remote means is used. The equipment consists of a gas supply system (helium or some other traceable gas), plumbing to feed the traceable gas to localized regions of the vacuum system, and monitoring instruments (such as a mass spectrometer or gas chromatograph) capable of sensing

Figure 4.4.1-2. Overhead Manipulator and Inspection System

the gas that has passed through the leak. This equipment is to be capable of operation in the Test Cell, and NB Test Cell and the Hot Cell using remote methods and elsewhere using contact methods.

Status of Design Alternatives

Alternative design approaches are under consideration in several areas where the Remote Handling and Maintenance System is involved. In general, these alternate approaches relate to cost reduction versus performance gain tradeoffs or else provide fall-back designs that can be assumed in the event the reference approach cannot be used. Several alternate approaches are discussed.

a. Direct Hoisting versus Rail Transport

Transportation of activated components from the Test Cell to the Hot Cell or the NB Test Cell can be accomplished using a traveling overhead crane or cars on a floor level rail system. The "crane" method of transport would involve hoisting a component such as the NB Injector from its position on the Test Cell floor to a height that is sufficient to permit the unit to be moved horizontally until it is over the top of the NB Test Cell and then lowered through a hatchway in the NB Test Cell ceiling, onto the floor. The head-room requirements for the Experimental Area using such an operation are roughly equal to the height of the NB Test Cell plus the headroom required for a 45 tonne (50 ton) bridge crane and components to be moved. Since the overall height of a NB Injector is about 7.6 m (25 feet), the head-room required for the Experimental Area is greater than 19.8 m (65 feet) and therefore excessive. Therefore, the rail transport system was selected over the direct hoisting approach.

b. Experimental Area Alternates

The physical arrangement of walls, windows, floors, ceilings and other features of the Experimental Area has a direct effect on the servicing capabilities of the RHMS. The arrangement selected as the reference conceptual design satisfies the requirements of RHMS with limitations in the overhead clearance required for remote servicing operations on the NB Injection units. Figure 4.4.1-3 schematically depicts the reference arrangement along with an alternate arrangement. The alternate arrangement provides a full height cell for testing and servicing NB Injector units and has the advantage of accommodating NB Injection units that are significantly taller than those that can be serviced in the reference arrangement. The electrical power conversion equipment needed for testing an NB Injector should be located near the NB Test Cell. This factor is considered a limitation of the alternate arrangements.

c. Overhead Manipulator and Crane Elevations

Remote servicing operations in the Test Cell and Hot Cell require an overhead manipulator and an overhead bridge crane in each cell. The conceptual design arrangement

Figure 4.4.1-3. Experimental Area Alternates (Elevation Views)

chosen for the Test Cell locates both the crane and overhead manipulator on the same level to reduce the overall height of the Experimental Area.

Normal practice in the design of hot cells, when head-room is available, is to locate the crane at a higher elevation than that of the manipulator in order that each can operate independently and cross over the other if necessary. The reference design was selected as a cost reduction feature associated with a 2.1 m (7 foot) decrease in overall building height.

d. Viewing from Shielding Windows versus TV System

The Test Cell is equipped with a closed circuit television system capable of monitoring the entire volume inside the cell. Two monitoring stations are located external to the cell. The purpose of the system is to provide, in conjunction with the shielding windows in the cell walls and shielded cab, sufficient viewing capability for conducting all remote handling and maintenance operations in the Test Cell. This describes the approach selected for the conceptual design reference. A second approach under consideration uses TV coverage exclusively, except for the shielded cab windows, for controlling and monitoring remote operations. A third approach being considered uses TV exclusively and eliminates the shielded cab assembly.

Many factors are to be considered in evaluating the merits of these three approaches, the principle ones being:

- Initial cost of viewing system
- Cost of operating and maintaining system
- Flexibility for performing a variety of maintenance operations
- Capability of performing known maintenance operations
- Speed of conducting operations
- Reliability of maintenance operations
- Redundancy in visual accessibility
- Safety of operations

4.4.2 TRITIUM STORAGE AND DELIVERY SYSTEM

The function of this system is to store the on-site inventory of tritium to be charged into the torus, purify and maintain the purity of the stored tritium and generate and deliver, on demand, via remote control by the central control, prescribed quantities of tritium to the torus for each D-T pulse or T buildup. In addition it also supplies the following: (1) argon inerted containment housings for the process equipment and the associated cleanup (detritiating) system which operates on the containment atmosphere; (2) temperature and pressure sensing devices which generate and transmit

signals to the Central Instrumentation and Control System (CICS) which form the basis for remote-automatic operation of the system with regard to such functions as valve opening and closing, temperature-pressure-power-flowrate-regulation, and evacuation of components and lines; (3) gas sampling ports for the radioactivity and tritium monitoring systems for determining the tritium level in the system inert gas atmosphere; (4) inert gas and closed-cycle cooling water packages; (5) Department of Transportation approved container for shipping a tritium generator and its housing to an ERDA facility for initial charging and recharging with tritium; (6) sealed piping ports and instrumentation, power, and valve containment bulkhead feedthroughs required for the charging operations at the ERDA facility.

Design Considerations

As a means for minimizing the quantity of gaseous tritium on-site, the basic design requirement is to store the tritium in a non-gaseous immobile configuration, as the tritide of an appropriate reactive metal. By confining this tritide in a small volume generator and thermal partial dissociation of the contained tritide, small amounts of gaseous tritium can be generated on demand, in accordance with experimental program requirements. Further, the design requirements include the following limitations: (1) the quantity of reactive metal present in the tritium storage unit (generator) for fixing the virgin tritium in non-gaseous immobile solid form is to be such as to restrict the on-site in-process tritium to 25,000 curies (Ci); (2) the total on-site tritium inventory is to be restricted to 50,000 Ci; (3) the amount of gaseous tritium on-site within the tritium system during normal operations is to be restricted to that corresponding to a "pulse-worth" of isotope, nominally in the range 50 to 200 Ci (50 to 20 mg) for the anticipated scenarios of experiments, not counting the gas phase tritium contained in the tritium generator; (4) the tritium supply is to be restricted to a single-use routine, with no on-site recovery of tritium in a purified form followed by recycle as feed to the torus; and (5) hazardous tritium handling operations such as recharging of generators and reactivation of spent plasma trapping getters are to be accomplished off-site at an ERDA facility.

The design basis is to include a four-year service life and a capability to furnish tritium to the torus at a repetition period of five minutes. Supply of measured quantities of gaseous tritium to the torus is to be accomplished automatically and rapidly, within an 0.25 second delivery time for 99 percent of the initial charge in the measuring volume. The quantity of tritium so charged is to be subject to controlled variation, in keeping with program requirements, minimally over a decade range, with an uncertainty of less than 3.5 percent.

In order to meet the tritium purity requirements the design includes provisions for purity control of the tritium delivered to the torus. Passage of particulate matter into the torus is prevented by multiple barriers. The generator and system configuration includes a means for removal of He-3 generated by natural radioactive decay of the tritium. Additionally the generator design purifies the initial thermal-diffusion purified tritium as supplied by ERDA through its contractor and maintain a high purity of less than 100 ppm impurity content during storage up to delivery to the torus.

The design target for normal tritium releases to the facility out of the storage and delivery equipment is less than $100 \mu\text{Ci}$ per working day. This objective, in part, is to be assisted by designing the tritium storage system so as to maintain the stored tritide gas phase pressure at subatmospheric, with a maximum not exceeding $3.2 \times 10^4 \text{ Pa}$ (250 torr) during active experimentation or $3 \times 10^{-2} \text{ Pa}$ (2×10^{-4} torr) during shutdown conditions.

Additional safety and operational flexibility is to be achieved by designing in features that include the following:

- The storage and delivery equipment is to be housed within containment to minimize tritium releases, and sized and operated to minimize dilution of the released tritium within the containment.
- Automatic disconnect of the power supply to the component heaters is to take place when their temperatures or associated gas pressures attain a maximum allowable level.
- Redundant valving and parallel paths for the tritium flow are to be provided between the main components housed within the system enclosure.
- Two complete identical tritium gas generator-purifier units and their individual, sealed metallic containments are to be provided, with one of these units connected on-line and the other stored in a sealed fuel vault in standby state when not enroute to or at the ERDA facility for recharging.
- Emergency power is to be provided for the vital portions of the system in the event of a complete loss-of-power from the local utility.
- Fireproof materials are to be used for constructing the containment enclosures for housing the tritium storage and delivery system.
- The containment enclosures are to be provided with a leak-tight air-lock type pass-through adequate in size to accommodate ready insertion and withdrawal of replacement items and servicing tools.
- Materials used for the system components, piping, and valves must have a low permeability to tritium and will be restricted to those having a good performance record in tritium processing operations.

In connection with the tritium solidification question, a number of candidate reactive metals were examined as the working material for the system. The basis for selection of depleted uranium metal as the working material over other candidates (Ca, Cs, Gd, Hf, La, Li, Mg, Na, Nb, Pd, Sc, Ta, Th, Y, Zr) included many items

such as: (1) ready tritide formation at modest temperatures as 200-400°C; (2) tritides are essentially non-dissociated at ~20°C; (3) convenient dissociation pressure at 300°C in the range 4×10^4 - 1.3×10^4 Pa (30-100 torr); (4) essentially zero vapor pressure for the virgin metal up to 500°C; (5) acts as a vigorous and retentive getter for impurities as oxygen, carbon, and nitrogen; (6) is non-reactive with other materials of construction for the generator as stainless steels, tungsten, nichrome, alumina. In particular, the self-purification capability and favorable dissociation pressure are compelling attributes for the selection of uranium. Additionally, there is a very good experience record at many ERDA facilities in the use of uranium for purifying and trapping-storing tritium.

Reference Design Description

In order to provide safety and protection against tritium releases to the facility or plant environs, the tritium generator and dispensing equipment is housed within mated small volume inerted containments which are provided with a detritiating system that operates in closed-cycle mode on the containment atmosphere. Additional safety and protection against tritium releases from the generator is afforded by: (1) trapping any tritium permeated through the generator inner shell in the minimal water quantity of the small volume [114 liters (30 gal)] closed-cycle cooling water system; (2) providing a high integrity thick wall stainless steel (1.3 cm AISI 304-SS) sealed containment barrier around the tritium generating and purifying components. The latter requirement, along with maintenance of an inert (argon) containment atmosphere and housing all of the enclosed equipment within a doubly-sealed thick wall concrete fuel vault, provides safety protection of the stored tritium inventory against potential accidental releases stemming from equipment malfunctions, such as fires or explosions within the facility.

When integrated, the primary equipment of the tritium storage and delivery system includes the tritium storing-generating, purifying, dispensing and delivery devices. Additional equipment items include the containment housings for the process equipment, the associated cleanup system, an inert gas supply system, and a package-type cooling water unit. The cleanup system is one of the five subsystems described in Section 4.4.3. The complete array of equipment packages comprising the tritium storage and delivery system includes the following: (1) two tritium generator-purifiers contained in tritium generator enclosures; (2) one tritium measuring and supply system contained in a glove-box constructed of AISI 304-SS and laminated safety plate glass; (3) one Department of Transportation approved shipping container; (4) a closed-cycle cooling water system; (5) argon inert gas supply system; (6) array of sensors, instruments, heaters, ducts, piping, manifolds, valves, filters, connectors, blanking flanges, tritium monitoring ports, etc., as necessary for the proper functioning of the equipment.

The package-type thermostatted closed-cycle cooling water unit comprises the following: completely closed-sealed water storage tank, hermetically sealed, leakproof

pumps, filters, mixed bed cationic-anionic disposable cartridge type demineralizer, temperature sensors, thermostat controls, chiller and heater, electrical conductivity type water purity meter-indicator, package housing, valves, and piping.

Figure 4.4.2-1 schematically indicates the configuration arrangement and the basic equipment requirements. Figure 4.4.2-2 shows how the tritium storage and delivery system interfaces the vacuum vessel and the TFTR vacuum system.

The tritium supply is immobilized as a solid in the form of uranium tritide within the double-walled (1.5 cm thick SS-316) tritium generator, with the mass of uranium (~73 g) corresponding to a maximum of 25,000 Ci of combined tritium based on $UT_{2.8}$.

The tritide is supported centrally in the generator cavity within a tungsten crucible (generator capsule) and is capable of being electrically resistive-heated by an alumina bead-insulated nichrome heating coil which is positioned between the tungsten crucible interior wall and the tritide charge. Elevation of the temperature causes partial dissociation of the tritide thereby generating tritium gas. The dissociation pressure varies exponentially and very conveniently with the temperature, e.g., 10^{-2} Pa (7.5×10^{-5} torr) at 10°C and 10^4 Pa (93 torr) at 300°C . Through appropriate regulation of the tritide temperature and tritium gas pressure within the system, close control can be achieved for dispensing prescribed quantities of tritium for particular pulsing scenarios, within the range of values required.

Purification of the initial tritium charged into the generator and retention or improvement of this high purity is afforded by three direct means. Impurities initially present that contain oxygen, carbon, and nitrogen are eliminated automatically by "self-purification" through gettering action by the uranium, forming compounds which have extremely low dissociation pressures at the generator maximum temperature [less than 10^{-18} Pa ($\sim 10^{-20}$ torr) at 400°C]. Noble gases originally present in the initial tritium gas supply (such as He-3, natural He, Ar) or created subsequently by tritium radioactive decay (He-3) are removed down to a very low concentration (on the order of 3×10^{-6} mole percent) by means of the palladium alloy diffuser and by bypass purging to the vacuum system.

The prescribed quantity of gaseous tritium in the precursor gas for a specific plasma experiment is generated by heating the generator capsule to the appropriate temperature in the range $280\text{--}350^{\circ}\text{C}$, corresponding to tritium pressures of $7 \times 10^3 - 4 \times 10^4$ Pa (50-300 torr) within the generator. The required amount of gaseous tritium is "leaked" into the torus from a $300\text{--}350\text{ cm}^3$ (calibrated) measuring volume maintained at 20°C , which previously has been evacuated to 5×10^{-6} Pa (4×10^{-8} torr) and back-filled to the appropriate pressure by interconnecting the generator-diffuser array and measuring volume.

Figure 4.4.2-1. Schematic of Tritium Storage and Delivery System

Figure 4.4.2-2. Schematic of Interfacing of Tritium Storage and Delivery System with Vacuum Vessel and Vacuum System

Attainment of the principal safety and precision-operational flexibility requisites for the design is achieved by a number of features, including the following:

- Gas delivery precision is accomplished by maintaining the temperature of the measuring volume at $20^{\circ}\text{C} \pm 0.2^{\circ}$ and adjusting the gas pressure within the measuring volume by appropriate regulation of the tritium generator temperature/pressure.
- Provision is made for automatic power disconnection of the power supply to the component heaters when the maximum allowable levels are attained -400°C for the generator heater and 7×10^4 Pa (500 torr) for tritium pressure.
- As shown in Figures 4.4.4-1 and 4.4.4-2, redundant valving and parallel paths for tritium flow are provided between the main components housed within the system enclosures.
- Passage of particulate matter into the torus is prevented by an array of barriers including the palladium alloy diffuser and two porous (SS-316) metal filters.
- Provisions have been made for emergency power supply, in the event of a complete loss-of-power from the local utility, for the closed cycle cooling water package and the gas circulator and gas preheater of the tritium containment cleanup system.
- AISI 316-SS, which has an excellent service record in tritium processing equipment, is used throughout for the main component construction, and for lines and valves. Except for the generator head closure, all joints are full penetration TIG or electron beam welds, and all components are to be degassed by vacuum bakeout to $400\text{--}500^{\circ}\text{C}$ prior to assembly.
- As a means for minimizing dilution of potential chronic and accidental tritium releases the containment housing volume has been set low at 2.27 m^3 ($\sim 80 \text{ ft}^3$). These containments comprise an SS-304 $61 \text{ cm} \times 61 \text{ cm} \times 91 \text{ cm}$ (2 ft x 2 ft x 3 ft) tritium generator enclosure and a $183 \text{ cm} \times 122 \text{ cm} \times 91 \text{ cm}$ (6 ft x 4 ft x 3 ft) tritium measuring and supply system glove-box.
- The containment enclosures are provided with a leak-tight air lock type pass-through with size $61 \text{ cm} \times 61 \text{ cm} \times 91 \text{ cm}$ (2 ft x 2 ft x 3 ft), adequate for ready insertion and withdrawal of desiccant cartridges, other replacement items for the cleanup system, and servicing tools.

- Fireproof materials are specified for constructing the glove-box enclosure, for housing the tritium gas measuring equipment and the associated inert atmosphere detritiating system, with minimum use of organic materials (gloves, seals).

Reference design parameters are presented in Table 4.4.2-1.

Status of Design Alternatives

The conceptual design is based on the use of extremely high quality "zero leakage" remotely operated (solenoid), rapid operating valves that are both free of organic materials and are capable of vacuum bakeout at temperatures in the range 400-500°C. Meeting these requirements will involve a development program. In the event that this program is not successful, the fallback position will be to redesign the tritium storage and delivery system to use the best quality all metal double-bellows valves available, equipped with pneumatic (argon) operators. This will also necessitate some pressure control and piping complications in the design of the inert gas system for the containments.

In an earlier design of the tritium generators, means were provided for rapid internal cooling of the tritium generator internals (crucible and uranium tritide) to approximately -50°C.

This feature, which was incorporated in order to minimize tritium permeation in-between pulses and during outages, utilized a "cold finger" fed with a stream of cold argon or helium to accomplish the cooling. This approach was desirable in view of the low duty rate imposed by the 48-minute pulse repetition time that applied then, along with omission of containments about the generating and dispensing apparatus. Rapid cooling between pulses is not feasible for a five-minute repetition rate. However, there may be sufficient merit to reincorporate the cold finger as a means for reducing tritium permeation through the generator. This would be especially true if D-T experiments are conducted both early and over the entire experimental program instead of being deferred until the last four years of the testing program.

Analysis of the time required to transport the tritium charge from the measuring volume and supply conduit into the torus has indicated that the 0.25 second required time can be satisfied. In the event that shorter delivery times become necessary due to changes in plasma or operating scenarios, such shorter delivery time requirements can be met by inserting an additional component (ballast chamber with volume of ~10 liters) at the end of the supply conduit immediately before the leak valve feeding into the torus. By interconnecting the measuring volume, supply conduit and ballast chamber after measuring out the required tritium charge about 20-40 seconds prior to initiating the charge into the torus, the flow impedance of the system can be reduced, thereby permitting more rapid injections. Action on this matter is to be deferred, pending firm definition of operation scenarios.

TABLE 4.4.2-1. TRITIUM STORAGE AND DELIVERY SYSTEM DESIGN SUMMARY

Design Life, Yr.	4
Full Power D-T Pulses per Year	1000
Maximum Full Power D-T Pulses per Day	96
Expected Full Power D-T Pulses per Day	40
Maximum On-Line Tritium Inventory, Ci	25,000
Maximum On-Site Tritium Inventory, Ci	50,000
Time Interval Between Full-Power D-T Pulses, min	5
Tritium Charge Delivery Time for 99% of Charge, sec	0.25
Nominal Tritium Charge per Pulse, Ci	70-100
Tritium Charge Capability per Pulse, Ci	50-200
Stored Tritium Form	UT ₂
Tritium Generator Uranium Content, g	73
Purity of ERDA Supplied Tritium Charge, w/o	99.9 +
Target Purity of Tritium Charged to Torus, ppm Impurity	≤ 100
Maximum Uncertainty in Tritium Quantity Delivered to Torus, %	3.5
Volume Range for Calibrated Measuring Volume, cm ³	300-350
Target Maximum Tritium Release from Generator, μ Ci/operating day	100
Evacuations Base Pressure, Pa (torr)	5.3×10^{-6} (4×10^{-8})
Maximum Generator Pressure During Shutdown, Pa (torr)	2.7×10^{-2} (2×10^{-4})
Nominal Generator Pressure During Operation, Pa (torr)	3.3×10^4 (250)
Expected Generator Temperature During Operation, °C	280-350
Power Cutoff States for Generator	
Temperature, °C	400
Pressure, Pa (torr)	6.7×10^4 (500)
Generator Heater Power Rating, kW	1.2
Measuring Volume Jacket Temperature, °C	20 ± 0.2
Provision for Emergency Power to	Cooling Water Package Cleanup System Gas Circulator and Preheater
Palladium Alloy Diffuser	
Nominal Operating Temperature, °C	400
Startup Time, min	≤ 30
Capacity at 1.3×10^4 Pa (100 torr) Inlet Pressure, liter/hr (cfh)	≥ 28(1)
Materials of Construction	
General Internals	
Crucible	Tungsten
Heater Elements	Nichrome
Heater Elements Insulation Beads	Alumina
Thermocouple Sheath	SS-316
Generator Body	SS-316
Generator Closure Seals	Gold
Generator Auxiliaries Pass-Throughs	Alumina
Diffuser Body	SS-316
Valves and Piping	SS-316
Generator Enclosure	SS-304
Glove-Box Frame	SS-304
Porous Metal Particulate Filters (2)	SS-316
Containment Dimensions, cm (ft)	
Generator Enclosure	61 x 61 x 91 (2 x 2 x 3)
Glove-Box	183 x 122 x 91 (6 x 4 x 3)
Glove-Box Pass-Through	61 x 61 x 91 (2 x 2 x 3)

In the current design of the package water cooler the single thermostating controls may not be adequate in view of the different temperatures and heat dissipation loads for the various components (tritium generator, measuring volume, cleanup system after cooler) serviced by this subsystem. The alternate approach includes providing separate tanks, pumps and thermostatted controls for each of these components in the package water cooler.

In order to provide additional flexibility for the quantity of gas to be injected into the torus, consideration is being given to installing an additional measuring volume, appropriately sized ($\sim 1000 \text{ cm}^3$ vs. $300\text{--}350 \text{ cm}^3$ for the current design), in parallel with the smaller volume. This will offer the opportunity to cover a wide tritium injection range without requiring large spreads of generator operating temperature and gas pressure.

4.4.3 TRITIUM CLEANUP SYSTEM

The function of the tritium cleanup system is to detritiate the atmospheres of facility areas where tritium is handled or released down to activity levels which are below the ERDAM-0524 established maximum permissible concentration (mpc) for controlled areas ($5 \times 10^{-6} \mu\text{Ci/cm}^3$), via closed-cycle type of repetitive processing of these atmospheres.

In those areas where the anticipated chronic type of tritium release (due to permeation through equipment walls) results in activity levels below mpc, or where the quantity of in-process tritium is relatively small, the cleanup systems are held in a standby status, ready to cope with emergency situations resulting from accidental releases. The test cell cleanup unit (which under emergency conditions could also be called upon to service other areas) and the hot cell cleanup unit are in this category, and are intended primarily to serve only in the event of emergency situations. The tritium cleanup units that operate on the inert gas within the containments for the tritium storage and delivery system and the air of the fuel vault are intended to be operated continuously. The intermediate position between these two extremes of operation is that of the Torus Cleanup System. This has the additional function of operating on the non-getterable gas residues from spent plasmas, which are collected in the gas holding tanks of the torus and Neutral Beam Injector vacuum systems.

Specific functional duties of the tritium cleanup systems include:

- Providing safety to the facility personnel and public against exposure to above mpc concentrations of tritium.
- Recovering reasonably rapid use of the facility in the event of an accidental tritium release.

- Converting released gaseous tritium into tritiated water, removing this tritiated watervapor from the various carrier atmospheres by trapping it in solidified, firmly bound form, on a desiccant, and disposing the spent tritium bearing desiccant by off-site burial.
- Decontaminating the torus prior to opening it for maintenance or modification, by removing the contained residual gaseous tritium and the absorbed, more loosely bound, monolayer of tritium from the torus interior surfaces.
- Detritiating the non-getterable gases collected in the vacuum pumping system holding tanks to mpc through utilization of the torus cleanup system in its other dual-purpose role, thereby permitting discharge of these detritiated gases to the environment.

Design Considerations

The design basis includes a four-year service life for the process equipment and a capability to effect cleanups from major tritium releases in the range 2000-50,000 Ci within relatively short operating times, in the range 24-72 hours. All of the approaches considered involve catalytic oxidation of the gaseous tritium to tritiated water. The methods examined for removing the tritiated water from the carrier atmosphere(trapping), include cryogenic condensation on panel surfaces or absorbents (molecular sieves), scrubbing in water, room temperature physisorption on activated alumina and silica gel, gettering by reactive metal preparations, and fixing on chemi-sorption type of desiccants. The latter approach was selected on the basis of operational safety and simplicity, tight retention of the tritiated water in the material to be disposed, and in consideration of potential accidents during transportation to the disposal site.

The target tritium level in the detritiated atmosphere is $5 \times 10^{-6} \mu\text{Ci}/\text{cm}^3$ for controlled areas. To provide margin in capability of the systems, the design basis (for computational purposes) for the cleanup was set at $10^{-8} \mu\text{Ci}/\text{cm}^3$, despite the fact that this level is below the sensitivity of currently available commercial tritium monitoring equipment.

An important design consideration is the form of packaging for the desiccant for ease of safe disposal of the spent material. For small systems, packaging in a self-contained cartridge form is a desirable configuration. For large systems the housing for the desiccant packed bed serves as the inner containment for off-site disposal of the spent desiccant.

In the case of very low concentrations of tritium in the gases fed to the oxidation catalyst, it is both advantageous and necessary to use the common radiochemical "carrier technique" of injecting a small flow of hydrogen upstream of the catalyst. This technique is essential in the cleanup systems for the test cell and the hot cell. In the case

of the cleanup unit for the tritium storage and delivery system argon atmosphere, it is necessary to make up for the oxygen deficiency. This can be accomplished by including a solidified oxygen supply; e. g., Hopcalite in the gas treatment train. Hopcalite is a proprietary solid regenerable preparation (50 w/o MnO_2 , 30 w/o CuO , 15 w/o CoO , 5 w/o Ag_2O) which is a vigorous oxidizer at modest temperatures.

As indicated previously in Section 4.1.8, in the case of the torus cleanup unit, it is necessary to provide means to catalytically crack (chemically fragment tritiated non-getterables and hydrocarbons into forms which are readily reactive during passage through the catalytic oxidizers. These complex compounds may be formed during passage through the catalytic oxidizers. These complex compounds may be formed during the plasma discharges. However, the precise chemical makeup and quantity of these that may result during TFTR plasma discharges is not known.

Because of potential shaft seal problems, the gas compressor-circulators for the four larger cleanup systems have to be located within separate sealed containments in order to eliminate possibilities for leakage. This is not necessary in the case of the cleanup unit for the tritium storage and delivery system in that the entire unit is already contained within the system glove box.

Reference Design Description

Each of the detritiation systems comprise equipment that includes minimally a gas compressor-circulator, a gas preheater, a precious metal catalytic oxidizer, a gas after-cooler, a parallel set of dual in-series array of desiccant beds and desiccant chambers, process and isolation valves, temperature sensors, pressure sensors, flow indicators for gas and cooling water, gas sampling ports for tritium sampling for monitoring, heater controls, supports, and an assembly mounting. In the case of the two smaller systems (tritium storage and delivery, fuel vault), the desiccant configuration is in the form of canned cartridges with false tops and bottoms which are punctured automatically when the cartridges are locked into the chambers. For the three larger units the desiccant is in the form of packed beds.

Because of the differences in duty cycle, makeup of the atmosphere to be treated, and configuration within the TFTR facility, the five cleanup units vary among each other with regard to size, functional capability, and physical orientation requirements. All of the units accomplish detritiation by catalytic oxidation of the tritium and tight trapping of the resulting tritiated water through desiccation on a solid absorbent. Generically, the process schemes for the detritiation installations embody the elements shown schematically in Figures 4.4.3-1 as appropriate.

Specific design features and parameters for the five cleanup systems are as listed in Table 4.4.3-1. When designed in accordance with the requirements set forth in Table 4.4.3-1, the five cleanup systems will yield the decontaminations shown in Figures 4.4.3-2 and 4.4.3-3. The drop-off of the tritium concentration in the atmosphere

- NOTES: 1. Valves not shown.
 2. Dotted items optional, depending on specific cleanup system.
 3. All systems operate in "closed cycle mode," except for inleakage and compensating purge of detritiated stream.

Figure 4.4.3-1. Simplified Schematic Tritium Cleanup Systems

TABLE 4.4.3-1 CLEANUP SYSTEMS PRINCIPAL FEATURES AND DESIGN PARAMETERS

ITEM	Tritium Storage & Delivery	Fuel Vault	Torus	Hot Cell	TFTR Cell
Treated System Gas Volume, m ³	2.27	36.8	88.9	2140	19,440
Design Basis Tritium Release, Ci	25,000	50,000	2000	2000	2000
Gas Medium	Argon	Air	Air	Air	Air
Estimated Air In-leakage Rate, m ³ /min	0	0	0	0.37	3.40
Nominal Gas Medium Inlet Conditions					
Temperatures, °C	25	25	22	22	22
Absolute Humidity, %	< 1	40	40	40	40
Operation Mode	Continuous	Continuous	Intermittent	Emergency	Emergency
Maximum Initial Tritium Concentration, $\mu\text{Ci}/\text{cm}^3$	1.1×10^4	1.5×10^3	2.3×10^1	9.4×10^{-1}	1.0×10^{-1}
Design Basis Final Tritium Concentration, $\mu\text{Ci}/\text{cm}^3$	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}
Operations Goal Final Tritium Concentration, $\mu\text{Ci}/\text{cm}^3$	5×10^{-6}	5×10^{-6}	5×10^{-6}	5×10^{-6}	5×10^{-6}
Desired Cleanup Operation Period, hr	24	72	24	40	48
Design Basis Flow Rate, m ³ /min	0.042	0.20	1.13	15.6	113
Hermetic Type Gas Compressor-Circulator	No	Yes	Yes	Yes	Yes
Capability for Cracking-Oxidizing Hydrocarbons	No	No	Yes	No	No
Hopcalite Oxidizer	Yes	No	No	No	No
Capability for Hydrogen Carrier Addition	No	No	No	Yes	Yes
Regenerable Desiccant Absorbers	No	No	No	No	No
Desiccant Form	Cartridge	Cartridge	Bed	Bed	Bed
Catalyst Nominal Operating Temperature, °C	200	200-300	400-500	200-300	200-300
Design Basis T ₂ O Pickup by Desiccant, g	9.3	18.5	0.76	0.76	0.76
System Volume Turnovers for Cleanup Campaign	27	25	18	21	17
Heater/After-Cooler Capability, kW	0.2	1.2	10	80	700

Figure 4.4.3-2. Tritium Cleanup Campaigns - Temporal Variation of Activity Level

Figure 4.4.3-3. Tritium Cleanup Campaign - Operation Time Versus Cleanup Achieved

within the containment being treated with time elapsed since actuating the system is shown in Figure 4.4.3-2. Figure 4.4.3-3 is a companion plot which shows the course of drop-off of residual activity with time of operation of the systems, in terms of percent of the initial total release not yet removed from the containment atmosphere.

Status of Design Alternatives

Insofar as basic process concepts and general system configurations are concerned there are no other design alternatives under consideration. There are under consideration, however, the potential need for additional cleanup systems and means for providing tritium cleanup capability for the transfer bays.

In the event that glove-box facilities are to be provided for the Hot Cell and Neutral Beam Test Cell for maintenance operations on tritium surface contaminated and embedded equipment it will be necessary to outfit these boxes with appropriate box atmosphere cleanup units, comparable to those for the tritium storage and delivery system.

In order to utilize the Test Cell cleanup system to detritiate the atmospheres of the transfer bays it will be necessary to incorporate the requisite ducting and valving in the ventilation equipment of these areas to provide for closed-cycle treatment of these atmospheres through the test cell cleanup system, along with appropriate controls for automatic valve sequencing.

4.4.4 NON-TRITIUM CHARGE GAS DELIVERY SYSTEM

The main functions of this system are to store the on-site inventories of high purity stable isotope gases (H-1, H-2, He-3, natural He) to be used in the plasma physics and other experiments on TFTR, to purify these gases by selective permeation processes, and to deliver on demand prescribed quantities of these gases to the torus for conduct of the various experiments. In addition to supplying the equipment required to accomplish these duties, this system also provides the pressure and temperature sensing devices which generate and transmit signals to Central Instrumentation and Control System (CICS) and, in return, receives signals and commands from the CICS which accomplish remote-automatic operation of the system with regard to such actions as valve opening, temperature and pressure regulation, heater power regulation, cooling water flow adjustment, evacuation of components and lines, and timing of the gas injections.

Design Considerations

The non-tritium gases to be used in the plasma experiments will be obtained as analysis-certified "Research Grade" high purity gases. These are available in compressed form, nominally at 11-15 Mpa (1600-2200 psi), delivered in thick-walled alloy steel cylindrical bottles ranging in small volume contents of 100-500 liters at standard pressure and

temperature. Despite the high purity of the commercially available gases (H-1, H-2, and normal He), e.g., 99.9999 percent purity for H-1, experience with these gases with cleaned high-vacuum exposed extended surfaces in plasma experimental equipment has indicated the need for additional purification.

There is to be a separate system for each of the aforementioned non-tritium gases, each of which is designed for an eight-year service life. In summary form, the design of each system is such as to provide for the following:

- Storing the gases in the high pressure cylinders furnished by the vendors and manifolding these cylinders to the gas purification components.
- Manually setting and adjusting the first stage pressure reduction of the gas supplies from initial pressures in the range 11-15 MPa (1600-2200 psi) to a reduced pressure in the range 0.1-0.7 MPa (15-100 psi)
- Purifying the gases additionally by appropriate selective membrane permeation techniques.
- Controlled preheating of the gases and heating the purifiers to their optimum operating temperature, followed by cooling the purified gases to $\sim 25^{\circ}\text{C}$.
- Preventing passage of particulate matter into the purified gas to be delivered to the torus.
- Automatically filling each of the four measuring volumes with its particular gas with an uncertainty of less than 0.5 percent, nominally in the range corresponding to a mean ion density in the plasma of $1.3 \times 10^{13} - 1.6 \times 10^{14}$ particles/cm³.
- Supplying the aforementioned quantities of non-tritium gases to the torus automatically and rapidly with an 0.25 second delivery time for 99.5 percent of the initial charge in the measuring volume.
- Furnishing the non-tritium gases at a repetition period of five minutes.
- Controlling the temperature of the measuring volumes to $20^{\circ}\text{C} \pm 0.2^{\circ}$.
- Regulating the quantities of gas to be delivered through automatic setting and controlling the fill-pressure for the measuring volumes, in the range $1.3 \times 10^3 - 1.3 \times 10^4$ Pa (10-100 torr), in response to command signals from the CICS.

- Evacuating the measuring volumes and supply conduits automatically to the vacuum system base pressure at $\sim 5.3 \times 10^{-6}$ Pa (4×10^{-8} torr) prior to making gas fills and injections into the torus.
- Close coupling of the measuring volumes to their torus injection ports and the torus vacuum pumping system.

Reference Design Description

The main equipment of the system comprises four integrated arrays of apparatus, with each array dedicated to but one of the aforementioned gases. Each of these arrays consists of a gas supply tank, first stage pressure reducing-regulating valve, second stage pressure control valving, isolation and gas leak-in valves, valve actuators, filters, gas diffuser-purifier, gas cooler, thermostatted measuring volume, manifold, temperature and pressure sensors, bleed trim valve to adjust from over-fill, precision pressure controller, piping, supports, connectors to the thermostatted water supply and return line, controllers for the diffuser heater and gas cooler, and tie lines to the torus and its vacuum pumping system. Generically, the hookup of each of the four arrays embodies the elements shown schematically on Figure 4.4.4-1.

A package-type thermostatted closed-cycle cooling water subsystem provides close temperature control of the measuring volumes. The package comprises the following elements: storage tank, circulating pump, filters, demineralizer, temperature sensors, thermostat controls, chiller-heater, water purity meter, valves, piping, housings and frame.

Vacuum services required for the system are supplied by the torus vacuum pumping system. These operations include evacuation of the lines and measuring volumes to $\sim 5.3 \times 10^{-6}$ Pa (4×10^{-8} torr) prior to back-filling the measuring volumes with the prescribed quantities of the four non-tritium gases.

Compliance with the design requirements is achieved through incorporation of the following features, as indicated schematically in Figure 4.4.4-1 and summarized in Table 4.4.4-1.

- Reduction of the stored gas pressure from 11-15 MPa to 0.1-0.7 MPa is accomplished by the manually adjusted pressure reducing-regulating valves.
- Additional purification of the gases is effected by selective membrane permeation techniques using heated palladium-silver alloy diffusers for the hydrogen purifications and vitreous silica (quartz) permeators for the helium purifications.

Figure 4.4.4-1. Schematic of Typical Hookup for Each Array for Non-Tritium Gas Storage and Delivery

- Passage of particulate matter into the purified gases delivered to the torus is prevented by interposing two stages of particulate barriers (porous stainless steel filters) between the purifiers and the gas dispensing components.
- Automatic filling of each of the four measuring volumes with its particular gas with an uncertainty of less than 0.5 percent, nominally in the required STP volume range of 30-150 cm³, is accomplished by programmed manipulations on the array of remotely operated valves, pressure controllers, and the calibrated measuring volume which is jacket thermostatted to 20°C ± 0.2°C.
- Delivering the prescribed quantities of gas to the torus automatically and rapidly within an 0.25 second delivery time for 99.5 percent of the initial charge in the measuring volume is accomplished by a "discharge-to-vacuum" technique, through a rapid acting bleed-valve located at the torus which is closely-coupled to the non-tritium gas system by an appropriately sized delivery conduit.

TABLE 4.4.4-1 NON-TRITIUM CHARGE GAS AND DELIVERY SYSTEM DESIGN SUMMARY

Design Life, yr	8
Number of Subsystems	4
Subsystem Gases	H, D, He-3, Natural He
Subsystem Storage Cylinder Volumes, STP liters	100-500
Storage Cylinder Initial Pressure, MPa (psi)	11-15 (1600-2200)
Quality of Vendor Supplied Gases	Research Grade
Reduced Pressure of Gases, MPa (psi)	0.1-0.7 (15-100)
Gas Charge Delivery Time for 99.5% of Charge, sec	0.25
Maximum Uncertainty in Gas Quantity Delivered to Torus, %	0.5
Gas Charge Capability per Pulse, STP cm ³	30-150
Evacuations Base Pressure, Pa (torr)	5.3×10^{-6} (4×10^{-8})
Volume of Calibrated Measuring Volume, cm ³	390
Measuring Volume Jacket Temperature, °C	20 ± 0.2
Purifier Diffusers for Hydrogen and Deuterium	
Membrane Material	Pd Alloy
Operating Temperature, °C	350-400
Over-Temperature Heater Cutout, °C	420
Startup Time, min	20-30
Heater Rating, W	500
Capacity at 6.9×10^{-1} Pa (10 psi), liter/hr (cfh)	≥28 (1)
Housing Material	SS-316
Purifier Permeator for He-3 and Natural He	
Membrane Material	Vitreous Silica
Operating Temperature, °C	300-600
Over-Temperature Heater Cutout, °C	650
Startup Time, min	30-50
Heater Rating, W	900-1000
Capacity at 3.4×10^5 Pa (50 psi), liter/hr	≥28 (1)
Housing Material	SS-316
Particulate Filters	
Number per Subsystem	2
Type	Porous Metal
Material	SS-316
Capacity at 3.4×10^4 Pa (5 psi), STP liter/hr (cfh)	4000 (140)
Trapping Capability for ≤ 0.3 Micron Particles, %	>99.99
Materials of Construction	
Valves	SS-316
Piping	SS-316
Measuring Volume	SS-316

As indicated above, the non-tritium charge gas delivery system functions by remote-controlled leaking of required amounts of the individual purified gases from their measuring volumes into the torus. Prior to filling, the measuring volumes into the torus. Prior to filling, the measuring volumes are evacuated to 5.3×10^{-6} Pa, using the torus vacuum pumping system for the evacuation. Close regulation of the fill pressure in the measuring volume is achieved by a control subsystem. This includes a precision pressure sensor with digital readout and feedback to control the second-stage reduced pressure setting to the pre-selected value corresponding to the prescribed quantity for an experiment, and an automatically operated bleed valve used for correcting for inadvertant overfillings.

Status of Design Alternatives

Currently there are no strong contender design alternatives for the four non-tritium charge gas systems. Early in the study several different design approaches were considered for the hydrogen and deuterium, and the potential alternatives include these.

One approach was based on generating ultra-high purity gas by palladium alloy diffusion of "commercial grade" material. However, in view of the relatively small quantities of these materials required for the plasma precursor gas, the use of commercial grade gas was ruled out.

In view of the ready availability of "Research Grade" hydrogen and the decision to use it as the starting material, the need for incorporating additional purification capability via palladium alloy diffusers was scrutinized. Previous PPPL experience with the C-Stellarator indicated that plasma impurity problems encountered (accumulation of oxygen, carbon monoxide, moisture on the walls and release back to the plasmas even when utilizing "Research Grade" hydrogen) were mitigated through incorporation of palladium alloy diffusers. Experience gained from PLT operations will provide guidance regarding the question of whether or not to reopen the palladium alloy diffuser issue. Until the results of PLT experience dictate otherwise, the design will include the diffusers.

The form of the starting material was additionally examined, in this case focussing on the question of restricting storage of cylinder hydrogen gas on the premises for safety reasons. The alternative approach involves replacing the high pressure cylinders by a hydrogen (deuterium) gas generator that produces the quantities of gas required on demand by electrolytic dissociation of high purity light (or heavy) water, followed by permeation purification via palladium alloy membranes. Commercial units for producing ultra pure hydrogen and deuterium (~10 ppb impurity level, no detectable hydrocarbons) by this route are available. This approach remains viable in the event that subsequent safety considerations rule out the presence of small cylinders of high pressure hydrogen and deuterium within the TFTR facility.

4.4.5 MACHINE AREA WATER COOLING SYSTEM

The Machine Area Water Cooling System provides a low temperature heat sink for the Tokamak field coils (toroidal, equilibrium and ohmic heating) and the power conversion equipment (PCE) in the field coil power supplies.

Design Considerations

Since both heat sources, the field coils and the power conversion equipment, must have high resistance coolant to minimize electric current circulation, demineralized water was chosen for the low temperature (10°C) application. This temperature level is provided by water chillers using a refrigeration cycle. Several options were available for the flow arrangements supplying these two heat sources. The simplest would have been a series circuit in which the water would supply, first the field coils in the TFTR test cell and from their exit the PCE in a separate building with the flow returning to the chiller-pump facility enclosure. This scheme would permit any water contaminated by any means in the TFTR cell to escape to the exterior of TFTR cell boundary if a pipe rupture would occur in the pipeline to or from the PCA building, or in the equipment itself. Furthermore, the PCE units would be exposed to the warmest water and require increased size. Another arrangement where the field coils and the PCE coolant passages were in parallel would reduce the coolant temperature, but would still permit the escape of any cell contaminated water in the event of a break. These considerations led to the requirement that the water supplied to the coils must be contained in a flow loop independent of the coolant loop for the PCE. If each heat source had its own loop serviced by its own chillers, the disparity in load would prevent any standardization in the chiller unit size. Furthermore, partial operation would be difficult. All these considerations led to the system shown on Figure 4.4.5-1. One set of chillers (6 for the reference design) extracts the total heat load of both the PCE and field coil from the coolant. This loop services both the field coils and a heat exchanger acting as a heat sink for the PCE loop. In this way, the two loops are isolated and any contamination from the TFTR cell which might be introduced in the coil coolant will remain in the cell confines. Since both loops have their own demineralizer, any activated solute in the water would be removed in its own demineralizer preventing any contamination which might occur if a common demineralizer were used for both loops. Hence, such a design meets all the basic heat removal system requirements which are:

- (a) preparing the initial system charge of high-resistivity, demineralized, deoxygenated, and filtered water, starting the treated "domestic" water supplied by the local water utility.
- (b) maintaining the high purity and high resistivity of the closed-loop cooling water throughout all subsequent operations.

- (c) filtering the system initial water charge to trap any dirt or metallic debris against entry into the various coolant channels, and also continuously filter the circulating closed-loop water.
- (d) removing the thermal energy from the two separate cooling loops:
 - (1) field coils cooling loop, and
 - (2) the PCE cooling loop,
- (e) performing automatically, requiring minimum manual operator controls, even under wide ranges of cooling loads, grossly fluctuating load changes, and no-load conditions.
- (f) minimizing dissolved oxygen entry into the system, and
- (g) dumping the energy load, via the condensers of the chillers, to the facility raw water cooling system.

Since the water cooling system provides a path to ground from the conductors of the various coil systems, it is essential that the water be extremely pure, oxygen free, and of high resistivity. The resistivity should be at least 10^6 ohm-cm. Due to electrolytic transfer of copper to the circulating water, it is necessary to provide the water cooling system with a cleanup subsystem capability for purifying the initial water charge and subsequent makeup water as well as maintaining the high water quality. It is also necessary to assure that the cooling water which may pick up contaminated corrosion products from the field coils is confined to a closed system and does not mix with water which goes to the other areas of the facility.

Although it would be desirable to provide redundant components with sufficient capacity to permit operation of the system without performance penalty when components of the system are down for maintenance, it is not consistent with the cost objectives. Instead the number and size of components selected to perform the various functions will be selected on the basis of providing adequate capacity to allow for continued operation of the system by extending the duty cycle from the normal five-minute period to seven or eight minutes.

In order to minimize cost, the system will be designed using standard .34 MPa (150 psi) fittings (piping, elbows, reducers, etc.). In addition, the components will be designed and positioned to permit convenient access for maintenance and/or replacement of components. The component materials will be selected so that mating components will be compatible with the high purity water when coupled with copper, austenitic stainless steel, aluminum alloys and plastic-coated carbon steel.

Figure 4.4.5-1. Schematic Arrangement of Machine Area Water Cooling System

Reference Design Description

The Machine Area Water Cooling System shown on Figure 4.4.5-1 shows the field coil cooling loop linked by means of a heat exchanger with the PCE cooling loop. The components such as chillers, pumps, storage tanks, heat exchangers, valves, and piping are schematically shown. Several features are noteworthy, namely the partial operating capacity of the entire cooling system is compromised only to the fraction of the on-line chillers. For example, if the system provides 15% excess capacity with all six chillers operating, the loss of one chiller would still permit operation at 96%, and the loss of two chillers would still permit operation with 76% capacity. Redundancy for partial operation is also provided by duplicate pumps and filters. The capability of each loop can be described separately.

Field Coil Cooling Loop

This system removes the thermal energy generated in the magnetic field coils by electric resistance heating during pulses within 300 seconds following each pulse. Approximately 17% of the energy generated in the copper coils is transferred to the cooling water within the first nine seconds. The total copper coil energy is composed of:

- (a) 2645 MJ (2.51×10^6 BTU) in the toroidal field coils,
- (b) 42 MJ (3.98×10^4 BTU) in the ohmic heating coils,
- (c) 52 MJ (4.93×10^4 BTU) in the equilibrium field coils.

The Field Coil Cooling Loop is designed to be able to maintain a .34 MPa (50 psi) pressure differential for water flow across the coil cooling channel inlets and outlets, and to chill the water, with an installed refrigeration capability adequate to dissipate all of the energy deposited in the coils and return the cooling water temperature at the coil inlet back to its starting temperature 10°C (50°F) within five minutes after initiating a pulse. This requires minimally 9.12 MW (2597 standard tons) of refrigeration for the 300 second duty cycle under steady operation. Additionally, the system provides the heat sink (as shown on Figure 4.4.5-1) for the PCE in the amount of .73 MW so that the total chiller capacity required is approximately 9.85 MW (2805 standard tons of refrigeration). However, approximately 15% margin is provided so that the minimum designed capacity is 11.3 MW (3300 standard tons). During "no-load" conditions freeze-up of the chillers is prevented through heat injection into the water en route to the chillers, minimally at a power level of 2800 kW (2655 BTU/sec.) The water system flow rate must be adequate to satisfy the coil design requirement of a 121°C (250°F) maximum copper temperature. This requires a flow capability

of minimally $.226 \text{ m}^3/\text{s}$ (3600 gal/min). The maximum operating pressure within the system is less than $.69 \text{ MPa}$ (100 psi). The Field Coil Cooling Loop includes as a minimum the following components:

- Two identical main circulating pumps in a parallel hook-up. Each pump has a flow capability of $.226 \text{ m}^3/\text{s}$ (3600 gal/min.) when developing a dynamic head of $.69 \text{ MN/m}^2$ (100 psi).
- Two identical filters in a parallel hook-up. The filter medium is synthetic fiber mesh (e.g., Dynel) with openings that trap particles greater than 20 microns.
- At least three identical chillers, hooked up so that all possible combinations of operation can be effected with one, two, and all units in service at both full and partial refrigeration loads.
- A ballast (storage) tank with a capacity of 41.6 m^3 (11,000 gallons).
- A by-pass heater to prevent freeze-up of the chillers during "no-load" conditions by adequate heat injection into the water en route to the chillers.
- A means to over-pressure all gas cavity zones in the system (particularly the ballast tank, with a range of 3.4×10^3 to $1 \times 10^4 \text{ N/m}^2$ (.5 to 1.5 psi) inert gas, such as nitrogen (or argon).
- Accommodation for operation of the chillers at 10 to 20% of full load without surging by incorporating a condenser-to-evaporator hot gas automatic bypass.
- A water temperature control system that is of the anticipatory type, based on regulating the chiller outlet temperature in combination with the inlet water temperature to the chiller, so as to automatically keep the chiller on-line during periods of sudden shifts from high to substantially no thermal load by means of a process computer.
- A system monitoring and control from a main control panel which contains all operations controls, indicators, switches, annunciators, and a graphic panel that displays the system interlock configuration.

- A demineralization system as described below.

The cooling water for the Field Coil Cooling Loop is demineralized and deoxygenated by means of the subsystem shown schematically in Figure 4.4.5-2 which operates as follows:

- Turnover of the Field Coil Cooling Loop water volume through the purifying subsystem in a normal eight-hour working day through a continuous bleed and feed; i.e., by-pass flow of $1.9 \times 10^{-3} \text{ m}^3/\text{s}$ (30 gal/min).
- Each of the ion exchangers provides $.42 \text{ m}^3$ (15 ft³) of exchange resin, with type and capacity as follows:

Cation Exchangers. "Strongly acid" 8.5% cross-linked sulfonated polystyrene (e.g., Amberlite IR-120, Dowex-50, Nalcite HCR) having an exchange capacity from the H⁺ form of 2 meq/ml for an 8.5×10^5 meq total capacity.

Anion Exchangers. "Strongly basic" polystyrene matrix, trimethyl benzyl ammonium type, porous beads with 15-25% internal porosity (e.g., Dowex-21K, Duolite A-101, Nalcite SBR-P), having an exchange capacity from the OH-form of 1 meq/ml for a 4.3×10^5 meq total capacity.

Deoxygenation Exchangers. Same as anion exchangers, but regenerated to SO₃-2 form, having an exchange capability equivalent to deoxygenating 188 m³ (49,600 gallons) of air-saturated water at 20°C (9.3 ppm O₂) i.e., 1.75 kg (3.85 lb.) of oxygen.

Power Conversion Equipment Cooling Loop

This system removes the energy dissipated in the PCE by the power pulses within the five-minute interval following each pulse. The heat deposited in the PCE and its transfer to the water are estimated as:

- Starting approximately 15 seconds after the initiation of each pulse and being completed prior to the start of the next pulse.
- Approximately 185 MJ (1.75×10^5 BTU) per pulse being dissipated by the toroidal field coil power supply cooling water.
- Approximately 15 MJ (1.42×10^4 BTU) per pulse being dissipated by the ohmic heating coil power supply cooling water.

Figure 4.4.5-2. Field Coil Cooling Loop Demineralization System Piping Schematic

- Approximately 20 MJ (1.904×10^4 BTU) per pulse being dissipated by the equilibrium field coil power supply cooling water.

The PCE Cooling Loop includes as a minimum the following components:

- Two main circulating pumps with a flow capability of $3.15 \times 10^{-2} \text{ m}^3/\text{s}$ (500 gal/min.).
- Two identical filters in a parallel hook-up. The filter medium will be synthetic fiber mesh (e.g., Dynel) with openings that trap particles greater than 20 microns.
- A heat exchanger with a capacity of .733 MW (209 tons of refrigeration).
- A ballast (storage tank with capacity of 2.83 m^3 (750 gallons)).
- A demineralization system as described below.
- Turnover of the PCE Cooling Loop water volume through the purifying subsystem in a normal eight-hour working day through a continuous bleed and feed; i.e., by-pass flow of $6.3 \times 10^{-4} \text{ m}^3/\text{s}$ (10 gal/min).
- Providing each of the ion exchangers with $.14 \text{ m}^3$ (5 ft^3) of exchange resin, with type and capacity as follows:

Cation Exchangers. "Strongly acid" 8-8.5% cross-linked sulfonated polystyrene (e.g., Amberlite 1R-120, Dowex-50, Nalcite HCR) having an exchange capacity from the H^+ form of 2 meq/ml for a 2.83×10^5 meq total capacity.

Anion Exchangers. "Strongly basic" polystyrene matrix, trimethyl benzyl ammonium type, porous beads with 15-25% internal porosity (e.g., Dowex-21K, Duolite A-101, Nalcite SBR-P), having an exchange capacity from the OH-form of 1 meq/ml for a 1.43×10^5 meq total capacity.

Deoxygenation Exchangers. Same as anion exchangers, but regenerated SO_3^{-2} form, having an exchange capability equivalent to deoxygenating 188 m^3 (49,600 gallons) of air-saturated water at 20°C (68°F) (9.3 ppm O_2), i.e., 1.75 kg (3.85 lb.) of oxygen.

Component specifications for both loops are summarized in Table 4.4.5-1 and operating parameters are summarized in Table 4.4.5-2.

**TABLE 4.4.5-1 SUMMARY OF PRINCIPAL SPECIFICATIONS OF MACHINE AREA
WATER COOLING SYSTEM COMPONENTS**

Item	Field Coil Cooling Loop	Power Conversion Equipment Cooling Loop
Main Circulating Pumps		
Number of Units	2	2
Flow per pump	$.113 \text{ m}^3/\text{sec}$ (1800 gal/min)	$3.15 \times 10^{-2} \text{ m}^3/\text{s}$ (500 gal/min)
Suction Head	$65.8 \times 10^3 \text{ Pa}$ (9.54 psi)	$65.8 \times 10^3 \text{ Pa}$ (9.54 psi)
Dynamic Head	$6.9 \times 10^5 \text{ Pa}$ (100 psi)	$6.9 \times 10^5 \text{ Pa}$ (100 psi)
Motor	75 kW (100 HP)	37.5 kW (50 HP)
Material		
Casing	Austenitic SS	Austenitic SS
Impeller	Bronze	Bronze
Shaft	SS	SS
Ballast-Storage Tank		
Volume	41.6 m^3 (11000 gal)	2.83 m^3 (750 gal)
Material	AL 5052, SS or Fiberglass	AL 5052, SS or Fiberglass
Chillers		
Number of Units	(3 min)	
Total Capability	11.3 MW (3330 Tons)	
Condenser Coolant	Raw Water	
Unit Compressor HP	1,000	
Mode of Operation	Continuous	
Material, Water-Side	Aluminum	
Automatic Hot Gas Bypass	Yes	
Condenser Pressure	Valve	
Control Mode		
T Required for No-Load, °C	2.2	

TABLE 4.4.5-1 SUMMARY OF PRINCIPAL SPECIFICATIONS OF MACHINE AREA
WATER COOLING SYSTEM COMPONENTS (Continued)

Item	Field Coil Cooling Loop	Power Conversion Equipment Cooling Loop
Heat Exchanger	--	.733 MW (209 tons)
Filters		
Number of Units		
Unit Flow Capability	.113 m ³ /s ² (1800 gal/min)	3.15 x 10 ⁻² m ³ /s (500 gal/min)
Filter Medium	Dynel Cloth — 20 μ rating	Dynel Cloth — 20 — rating
Housing Material	Aluminum or SS	Aluminum or SS
Demineralizers		
Number of Units/Beds	2/6	1/3
Capacities		
Cationic	8.5 x 10 ⁵ meq	2.83 x 10 ⁵ meq
Anionic	4.3 x 10 ⁵ meq	1.43 x 10 ⁵ meq
Oxygen	1.75 kg (3.85 lb)	1.75 kg (3.85 lb)
Flow Rate	1.9 x 10 ³ m ³ /s (30 gal/min)	6.3 x 10 ⁻⁴ m ³ /s (10 gal/min)
Type	Rechargeable	Rechargeable
Bypass Heater		
Power Rating	2,800 kW (2655 BTU/s)	
Flow Capability	.013/m ³ /s (200 gal/min)	
Pump	11.1 kW (15 HP)	
Material, Water-Side	Aluminum or SS	
System Main Loop Piping		
Material	AL 5052 or SS	AL 5052 or SS
Size, Standard	45.72 cm (18")	15.24 cm (6")
Schedule No.	40	40

TABLE 4.4.5-2 MACHINE AREA WATER COOLING SYSTEM OPERATING PARAMETERS

	Field Coil Cooling Loop	PCE Cooling System
Water flow	.226 m ³ /s (3,600 gal/min)	.031 m ³ /s (500 gal/min)
Pressure drop across component	3.4 x 10 ⁵ Pa (50 psi)	TBD
Initial inlet water temperature to component	10°C (50°F)	12°C (54°F)
Nominal cooling cycle duration per pulse	300 sec.	300 sec.
Maximum inlet water temperature to chiller	26.5°C (79.7°F)	Not applicable
Maximum outlet water temperature from component	45°C (113°F)	17.5°C (64°F)
Maximum water outlet-inlet temperature differential across component	30°C (86°F)	5.5°C (10°F)
Maximum flow to bypass flow heater	1.2 x 10 ⁻² m ³ /s (200 gal/min)	Not applicable
Maximum system pressure	.69 MN/m ² (100 psi)	.69 MN/m ² (100 psi)

Instrumentation and Control

The water cooling system is initially operated from a control panel located in the chiller room. This panel board contains the individual controls and instrumentation to give assurance that the system is functioning correctly.

The Central Control Room contains the master control to start and/or stop the cooling water system remotely. A "ready" status and a "stop" or "fault" status indication is also provided in the Central Control Room. An indication of "fault" will require that a technician investigate the cause and clear the problem prior to any operation being allowed to continue.

An override is provided at the panel board in the chiller room so that the sequenced operations can be manually directed. (It is anticipated that the system will be manually operated during the initial check-out and debugging operations.) After the system has been proven to be operational, it will be remotely operated from the Central Control Room.

The control functions involved include the following:

- Preventing excessive cooldown and potential freeze-up of the water side of the chiller evaporators during "no load" conditions that could be encountered through delays in initiating an experiment subsequent to completing the coil cooldown following a previous experiment.
- Maintaining the proper pressure differential .34 MPa (50 psi) across the field coil cooling water inlet and outlet manifolds so as to provide the required cooling water flow.
- Ensuring that the system maximum pressure does not exceed 1.03 MN/m^2 (149 psia), in order to provide compliance with utilization of "USAS 150 lb. Standards" in the piping, fittings, valve and flange design.
- Maintaining a modest nitrogen over-pressure of approximately 6900 N/m^2 (1 psi) in the ballast tank and other gas-water interface zones in the system, as a means to prevent entry of oxygen into the cooling streams. The intent is to maintain a deoxygenated state for the water and thereby render the water less aggressive as regards corrosion of the copper coils and component materials.
- Maintaining cooling water high purity and high electrical resistivity by means of continuous demineralization and deoxygenation operations on a by-pass stream.

The chiller evaporator potential freeze-up problem stems from the limitation on chiller motor startup frequency. Protection against damage to motor windings limits the number of off-on starts of the motors per hour from an across-the-line start. For this reason the chillers are run continuously so long as additional pulses are scheduled for that particular day. The difficulty, however, is that the usual design of large chillers precludes operating under zero load conditions. To circumvent this difficulty it is necessary to impress a dummy load of 10 to 20% on the chillers. This is accomplished by means of the electrically heated small by-pass stream from the ballast tank to the main stream ahead of the pump suction. The controller for this side stream operates from both the chiller water inlet and outlet temperatures in a "Feed-forward" sensing mode, so as to automatically keep the chillers on line during periods of rapid change from high to essentially zero thermal load, meanwhile actuating the heated by-pass flow. This by-pass flow at maximum amounts to $.013 \text{ m}^3/\text{s}$ (200 gal/min) requires a power input of 2800 kW (2655 BTU/s heating the by-pass stream to 62°C (145°F), and maintains the temperature at the chiller inlet at 12°C (54°F). The chiller, in turn, cools the stream back to 10°C (50°F).

Control of the differential pressure across the supply and return headers to the field coils is accomplished by automatic actuation of a by-pass control valve which operates in concert with the system pressure limiting control valve by-passing flow from the main loop in an amount adequate to meet the hydraulic criteria.

Material

To the extent practicable, the same material of construction has been selected for fabrication of the cooling water system components. On the basis of such factors as availability, cost, fabricability, mechanical properties and compatibility with high purity water when coupled with copper and austenitic stainless steel, several aluminum alloys with proven compatibility records have been identified as candidates for construction of the piping, tanks, and filters. Additionally, those portions of the chillers exposed directly to the water (evaporator water side) are to be surface aluminized. The alloy candidates are:

Al 5052 (2.5 w/o Mg, .25 w/o Cr)

Al 6061-T6 (.24 w/o Cu, .25 w/o Cr, .6 w/o Si, 1 w/o Mg)

4.4.6 TORUS VACUUM PUMPING SYSTEM

Design Consideration

The control of impurities is of fundamental importance in the attainment of energetic plasmas. To this end the vacuum system, which means the vacuum vessel itself with its numerous appendages, together with the high vacuum pumps, should be capable of reaching a very low base pressure. Ultrahigh vacuum technology provides the basis

for such a design. In the case of TFTR there is a second important consideration in the pumping system design, namely that tritium will be used as a working gas and it cannot be discharged into the test cell or the atmosphere.

The conceptual design for the torus vacuum pumping system is based on a conservative estimate for the outgassing rate of 10^{-7} Pa m³/s m², for surfaces exposed to the torus vacuum. Mercury diffusion pumps evacuate the torus and discharge into a foreline which can be isolated from the atmosphere. Getter pumps sorb the hydrogen isotopes into solids, and mechanical pumps, discharging at atmospheric pressure, remove the ungettered gases. With the exception of an in situ bakeout of the vacuum vessel, ultrahigh vacuum technology will be utilized throughout. As a further measure all components which have surfaces exposed to the vacuum will be prebaked in vacuum to release impurities thus ensuring very low outgassing rates.

Experience has shown that satisfactory tokamak plasmas can be obtained in systems designed to have base pressures below 10^{-5} Pa. Since the residual gas is mainly hydrogen, impurities are well below this level. The principal contribution to the background gas is outgassing from the walls of the device. To limit outgassing, stainless steel is used throughout and the components are prebaked at 775 K before assembly. In addition, discharge cleaning of the device is carried out before normal plasma operation. Leaks in the system would also contribute to the background pressure; however, the exclusive use of ultrahigh vacuum flanges with metal gaskets essentially eliminates this contribution. In addition to the background gas, tritium and deuterium are injected into the vacuum vessel prior to each discharge. After the discharge these gases must be removed quickly by the pumping system.

The design of the vacuum pumping system takes into account the requirements that tritium and tritiated gases are to be pumped. This requires that the system be closed relative to these gases. This is achieved by providing a system that has the capability of holding these components in solid form. The solid form will minimize the hazards of storage, inventory, and transportation of the radioactive products. In addition, the exhaust for the pumping system will be permanently connected to the tritium cleanup system which purifies the gases before releasing them to the atmosphere.

Reference Design Description

The reference design is shown schematically in Figure 4.4.6-1 and summarized in Table 4.4.6-1. Given the total area of surfaces exposed to the vacuum, a base pressure of $< 8 \times 10^{-6}$ Pa can be obtained by providing a pumping speed of ~ 33 m³/s (for H₂). Mercury diffusion pumps were selected (over other types of pumps) because of extensive satisfactory experience in other plasma devices, their ability to pump all gases, their lack of tritium hold-up or isotopic exchange, and their reliability at moderate cost.

Figure 4.4.6-1. Torus Vacuum Pumping System Showing a Pair of High Vacuum and Backing Systems Plus Tritium Cleanup and Monitoring Systems.

TABLE 4.4.6-1 TORUS VACUUM PUMPING SYSTEM DESIGN SUMMARY

- 4 — 61 cm Diameter Diffusion Pumps
- An UHV Valve and Cold Trap at ≤ 148 K Above Each Diffusion Pump
- Diffusion Pump Stack Designed for Remote Maintenance
- A Total Pumping Speed for Hydrogen of $33 \text{ m}^3/\text{s}$ for the Four Diffusion Pumps
- Diffusion Pump Stack of Non-Magnetic and Nuclear Radiation Resistant Materials
- Pumpout e-Folding Time of < 10 sec (D-T)
- Base Pressure $< 8 \times 10^{-6}$ Pa
- Pre-Bakeout at 775 K for 2×10^5 sec (60 hrs) of Components Above Diffusion Pumps
- Discharge Cleaning
- Closed Backing System
- 4 — Mechanical Backing Pumps with Zeolite Traps for Removal of Non-Hydrogen Gases
- Mechanical Pumps Discharge Into 0.1 m^3 Ballast Tanks
- 8 — Zr/Al Canister Units for Removal of Hydrogen Gases Including "Cracker" to Breakdown Hydrocarbons
- Pumping Speed of $0.13 \text{ m}^3/\text{s}$ for Four Mechanical Pumps
- Rough Down Time of 1.8×10^4 s (5 hours)
- 2 — Residual Gas Analyzers with Resolving Power ≥ 50 , With Adjustable Scanning Speeds, and with Partial Pressure Sensitivity $\leq 10^{-8}$ Pa.
- 2 — Tritium Monitors for Checking Tritium Content in Ballast Tanks
- Interface With Tritium Cleanup System to Treat Mechanical Pump Exhaust
- Foreline and Backing System Located in Basement Below Test Cell
- All Hard (Metal) Seals
- Outgassing Rate $10^{-7} \text{ Pa m}^3/\text{s m}^2$
- Ultrahigh Vacuum (UHV) Standards
- Protective Interlock Systems
- Local and Central Instrumentation Control Systems for Manual and Automatic Operations
- Service Life of 8 years

The system of mercury diffusion pumps is made up of four 61-cm (24 in.) diameter pumps of which two are on ducts to the Neutral Beam Injectors, and the other two are mounted directly on the torus. Backing these pumps are four independent foreline and roughing systems, located in the basement. Each consists of a mechanical pump, oil traps, Zr/Al getter units, and the necessary valves and gauges. These systems are arranged in pairs with each pair sharing a residual gas analyzer for monitoring of tritium and for leak detection. In addition, they share a holding tank to contain the exhaust of the mechanical pump when tritium is being utilized.

The Deuterium-Tritium (D-T) operation of TFTR imposes several requirements not normally encountered in UHV system design. For example, the presence of tritium requires that the vacuum system be closed with respect to this gas, i.e., pumps cannot discharge tritium to the atmosphere. Further, because of hydrogen isotope exchange, hydrocarbons, elastomers, and the like are avoided for reasons of safety. Consequently, the diffusion pumps discharge into a foreline volume where hydrogen isotopes are removed by pumps which convert the gaseous hydrogen isotopes to solids, e.g., by solution in bulk metal or by hybridizing. The foreline volume is sized to permit accumulation of other gases for extended times before the critical backing pressure of the diffusion pumps is reached. These gases are removed by a mechanical vacuum pump. Other precautions against release of tritium to the atmosphere include "crackers" in the foreline to fragment gases such as tritiated methane, and a tritium clean-up system which can be valved to the exhaust of the mechanical pumps to remove any residual tritium. Monitoring of the partial pressures of tritium and tritium containing gases in the foreline is accomplished using a differentially-pumped, residual gas analyzer.

Design Alternatives

From the standpoint of obtaining a lower outgassing rate of all the surface areas, and hence a low-base pressure, an in situ bake is desirable. However, the advantages to be gained for the increased cost and complexity are not easily anticipated in quantitative terms. A more thorough investigation of the advisability of providing in situ bakeout will be conducted after the conceptual design phase. However, as it has been demonstrated that discharge cleaning is equivalent to bakeout as far as plasma behavior is concerned, discharge cleaning has been selected as an integral part of the pre-conditioning of the device in the current design.

4.5 NEUTRAL BEAM INJECTION SYSTEM

The neutral beam injection system provides up to 20 MW of 120 keV D^0 and 12-15 MW of partial-energy D^0 for the purposes of heating the plasma and driving the fusion reactions. The system injects the D^0 atoms at a target radius of either 270 ± 20 cm for Modes A or B (precompression plasma) or 220 ± 20 cm for Modes C or D (post-compression and quasi-static plasma). See Section 2.1.2 for definitions of A, B, C, D. The system also sweeps the beam free of unneutralized ions, provides for the measurement of the neutral beam power, prevents any excess of cold gas from entering the torus ($< 0.133 \text{ Pa m}^3/\text{shot}$), and provides a rapid and brief shut-off of any individual ion accelerator in case of a spark in an accelerator gap.

The neutral beam development groups at both Oak Ridge National Laboratory (ORNL) and jointly at Lawrence Berkeley and Lawrence Livermore Laboratories (LBL/LLL) have submitted conceptual designs for complete neutral injection systems, based upon current technology. The designs will be updated to include the results of development programs to be carried out by both groups. The development programs will have two parts: source and accelerator development on high voltage test stands; prototype beam line development, to test the integrated system. The developed version of one of these beam lines will be selected in January, 1979 as the final TFTR beam line design.

The development programs are being guided by a TFTR Neutral Injection Steering Committee composed of members from ERDA, PPPL, ORNL, LBL/LLL, and the TFTR industrial contractor. One consequence of this guidance will be frequent communication between the two development groups on the relative effectiveness of the various approaches to some of the design problems.

A likely result will be the inclusion in both final designs of features now found in only one proposal (or neither), or features that result from the development work at PPPL or at Brookhaven National Laboratory.

Both ORNL and LBL/LLL included in their proposals designs for power conversion and control systems; the design of these systems will be done by PPPL, however, and will not be the result of the selection of one out of two designs, as will be the case with the beam lines. The power and control systems described in Section 4.5.2 and SD 8B are the present conceptual designs of these PPPL systems, and not the proposals of either ORNL or LBL/LLL.

4.5.1 NEUTRAL BEAM LINE

Functions of the Beam Line

The functions of the four neutral beam lines are:

- Accelerate D^+ ions to 120 keV;
- Provide a total neutral beam power of 20 MW at 120 keV;

- Neutralize an optimum fraction of the accelerated ions;
- Remove the remaining ions from the beam before they enter the toroidal field;
- Provide a measurement of both the neutral and the total power;
- Provide a means of transmitting the beam into the torus without serious attenuation;
- Provide a means of aiming the beam at 2 (or more) different tangent radii (270 and 220 cm);
- Provide means of isolating the beam line from the torus, both on an absolute and on a fast (i. e. , between shots) basis;
- Provide pumping facilities to pump the beam line down from atmospheric pressure;
- Provide sufficient pumping to limit the gas flow into the torus to $< 0.033 \text{ Pa-m}^3/\text{shot/beam line}$;
- Provide for the regeneration of the high speed pumping system to release stored deuterium (contaminated with tritium).

Design Requirements

The design requirements for the neutral beam system that would fulfill the above functions were agreed upon in a workshop on 24 July 1975, which was attended by representatives of PPPL, Westinghouse, ORNL, LBL/LL, and ERDA, and are as follows:

- 1) Beam Specifications:
 - A) A single injection voltage and power for all plasma modes.
 - B) Accelerator voltage — 120 keV.
 - C) Accelerator power — 80 MW
 - D) Pulse duration — 500 msec.
- 2) Beam Lines:
 - A) Four beam lines, with provision for 2 more.
 - B) Three sources per beam line.
 - C) Injection at tangent radii:
 - Modes A or B - $270 \pm 20 \text{ cm}$
 - Modes C or D - $220 \pm 20 \text{ cm}$

3) Alignment:

The beam line will be aligned as in Figure 4.5-1 with one coil gap between the beam lines on each half of the torus, and 3 coil gaps between the beam lines on opposite sides.

4) Dimensions:

A) < 9 m from torus to far end of power connector.

B) <12 m from torus to nearest wall; 3 m clear passage between beam line and wall.

C) 6.5 m high.

5) Attachment Duct:

A) 70 cm high (inside clearance).

B) >40 cm wide (projection width).

C) $< 3 \times 10^{-3}$ Pa m neutral gas (measured from torus to sweep magnet).

D) 61 cm (24 in.) torus vacuum pumps.

E) Energy flux deposited on duct walls 1.2 kW/cm^2 .

F) Bellows in duct - a future consideration.

6) Gas Input to Torus - 0.1 Pa m^3 total/shot.

7) Fast Shutter:

A) Open and close times - 50 msec.

B) Located in beam line.

8) Effects of Beam-Line Magnets and Magnetic Shielding on Plasma (at Plasma Center)

A) $< 5 \times 10^{-3} \text{ T}$ from magnetization by plasma current.

B) $< 5 \times 10^{-4} \text{ T}$ from its own field.

9) Vacuum Considerations:

A) Hard seals throughout, except possibly on source.

B) Source and filaments easily removed by remote equipment.

C) Parts individually bakable (except possibly accelerator insulators), but not beam line.

D) Recycling of D_2 to be considered later.

Figure 4.5-1. Plan of Four (ORNL) Beam Lines

10) Conditioning and Cleanup Pulses:

- A) Power should be available to condition one beam line at a time at a high pulse rate.
- B) No provision for regular cleanup pulses on all sources (i.e., no pulsing of all sources every 30 seconds).

11) Cooling Requirements:

- A) $0.5 \text{ m}^3/\text{shot/line}$ of water to the outside cooling towers.
- B) High-flow cooling schemes to have a heat exchanger.

12) Maintenance Cell:

- A) A maintenance cell should be provided in the TFTR Facility.
- B) Enough power should be provided to break in 3 sources at once at fastest possible rate.
- C) 16 m long.
- D) 1.2 m shielding

Detailed Designs

The overall plan of the 4 beam lines on the tokamak is shown in Figure 4.5-1. An additional beam line could be located further around on each side of the tokamak. The beam lines shown are the beam lines proposed by ORNL; the LBL/LLL beam lines are somewhat longer and wider (but not so tall).

In general, each beam line has three accelerators coupled to three neutralizers, one iron-core sweep magnet with separate passages for the three beams, a calorimeter capable of handling the full power of the combined three beams, an ion beam dump similar in design to the calorimeter, a fast shutter, and an absolute valve connected to the attachment duct going into the torus. The beam line enclosures are divided into three chambers, all lined with cryopumping panels to provide pumping speeds of $\approx 3 \times 10^3 \text{ m}^3/\text{sec}$.

The beam line designs proposed by both ORNL and LBL/LLL have all of these general features, but differ in design approach and philosophy. The different approaches were to some extent forced upon the designs by the nature of the different accelerator systems used by the two groups. The different philosophies expressed in the written reports reflect two aspects of the design problem: Whereas the LBL/LLL proposal places more emphasis upon the physics choices to be made in the conceptual design, the ORNL proposal places more emphasis upon the engineering entailed in a particular concept. The two proposals are summarized (in the form of excerpts) in the next few pages.

ORNL Neutral Beam Line

Each ORNL beam line is composed of an arrangement of the beam components within a vacuum enclosure, which is designed to maximize performance by keeping the distance from the ion source to the plasma as short as possible. This is achieved by placing the components in a logical sequence as close to each other as functionally possible and by providing the required cryosorption pumping area by increasing the height of the chamber rather than length. The arrangement is shown in schematic form in Figure 4.5-2. The three sources are placed at the rear of the chamber mounted on a flange vertically one above the other. This allows the three beams to be in the same vertical plane, thus tangent to the same radius of the torus.

There is a neutralizing gas cell for each source. The three gas cells are also mounted on a removable flange which fastens to a bulkhead within the vacuum chamber. This bulkhead is made of stainless-steel-clad mild steel and forms a magnetically shielded box around the gas cells and the exit of the sources. Such a shield is necessary to permit the ions to travel through the gas cells in a straight line unaffected by the magnetic fields generated by the tokamak. The distance from the source to the entrance of the gas cells has to be sufficient to allow most of the source gas to be pumped away to prevent premature charge exchange of the ion beam within the accelerator structure and to allow the remainder of the vacuum enclosure outside the magnetic-shielded box to be pumped down through the gas cells by a 61-cm (24-inch) mercury diffusion pump.

Figure 4.5-2. Elevation of ORNL Beam Line

Immediately after the exit of the gas cells, the beams pass through the magnetic field of the iron-core bending magnet. This magnet is made in halves and is placed in cavities in the outside of the vacuum enclosure. The magnet return path is inside the vacuum chamber, but the magnet itself is supported by rods in the exterior cavities on each side of the chamber.

ORNL Operating Data

During normal operation the operating data for the neutral beam system are as given in Table 4.5-1.

ORNL Ion Source Assembly

The DuoPIGatron ion source was developed at ORNL for use in neutral beam injection plasma heating. The 125-kV version shown in Figure 4.5-3 will be developed on the 150-kV beam test stand now under construction. This source is an extension of the one developed for use on ORMAK, which has been proven highly reliable.

The source is divided into three subassemblies to make the system more accessible and easier to assemble. The top assembly contains the cathode, which consists of a tantalum-wire heater with emission carbonates coated on a support gauze. The tantalum filament will be removable as a unit with a remote manipulator. The deuterium gas feed to the source is also connected into the filament-support flange.

The second subassembly contains the intermediate electrode, magnet coil, and anode. The intermediate electrode is soft iron. The electrode shields the cathode region of the source from a magnetic field and shapes the magnetic field produced by the coil in the discharge region, compressing the arc discharge for greater ionization.

The third subassembly contains the four-electrode acceleration system. The target cathode operates slightly positive with respect to the intermediate electrode, and is optically aligned with the accelerating electrodes. The electrodes are copper, water-cooled around the periphery and also across the electrode. They are drilled in a programmed offset hole pattern of about 50% transparency and steer all individual beamlets to a common focal point. The insulators are fluted Al_2O_3 with metal terminations. The assembly is welded or brazed together to avoid elastomer seals. The entire source assembly may be remotely disconnected and removed from the beam line.

ORNL Ion Removal System

The ions are swept out of the beam and into the beam dumps in the bottom of the enclosure by a single-gap 0.27 tesla iron-core magnet. The three beams pass through different locations in the single gap, but are all deflected 1.22 rad (70°).

TABLE 4.5-1 OPERATING DATA

	120 keV	150 keV
Ion Current Per Source (Amperes)	53.0	44.0
D ⁺ Per Source	42.4	35.2
D ₂ ⁺ Per Source	10.6	8.8
D ⁰ at Rated Energy (As Amperes)	17.2	10.1
D ⁰ at Half Energy	13.9	10.3
Total D ⁰ to Torus (Calorimeter)	31.1	20.44
D ⁺ to Dump at Rated Energy	25.2	25.1
D ⁺ to Dump at Half Energy	7.3	7.3
Gas Loads Per Source (Pa-m ³)		
Un-ionized Gas	1.41	1.17
Gas to Neutralizer Cell	2.66	2.66
Gas to Beam Dump	.87	.87
Gas to Torus (Calorimeter)	.83	.55
Total Gas Per Source	5.77	5.25
Pressure in the Pumping Regions (Pa)		
Source Region	3.3×10^{-2}	3.1×10^{-2}
Beam Dump Region	3.3×10^{-3}	3.3×10^{-3}
Calorimeter Region		
Beam on Calorimeter	4.1×10^{-3}	2.8×10^{-3}
Beam into Torus	6.7×10^{-4}	6.7×10^{-4}
Water Flow Through Calorimeter	0.12m ³ /sec(2000 gpm)	0.15m ³ /sec(2550 gpm)
Water Pressure Drop Through Calorimeter	3.5×10^5 Pa ≈ (50 psi)	3.5×10^5 Pa ≈ (50 psi)
Peak Power Density of Neutral Beam in Plasma (at 9 m from source)	3.839 kW/cm ²	3.03 kW/cm ²
Peak Power Density of Total Beam at Calorimeter	10.4 kW/cm ²	10.8 kW/cm ²
Effective Power Density at Calorimeter (Taking geometry of calorimeter into account)	0.90 kW/cm ²	0.94 kW/cm ²

Figure 4.5-3. ORNL Ion Source and Accelerator Assembly

ORNL Cryosorption Pumping

The requirements for pumping speeds approaching $3 \times 10^3 \text{ m}^3$ per second indicated a choice of either cryosorption or cryogenic pumping. Cryosorption was the choice for the reference design, because of two principal factors: The first is the apparent ease of regeneration by warming the panel slightly to $\approx 50 \text{ K}$. There is no problem from storage or awaiting boil-off of liquid helium when regenerating. The second is the reduced refrigeration requirement. The refrigerator size requirement and operating cost at 4.2 K is three to four times that required at 15 K using equivalent pumping arrangements.

The geometry for the pumping panels is shown in Figure 4.5-4. The array consists of two chevrons of Clausing factor 0.25 cooled to 77 K and 15 K , the cryosorption panel at 15 K and a 77 K thermal shield between the panel and the outside wall. A conservative pumping speed of $50 \text{ m}^3/\text{s}$ per m^2 of incident area was calculated for this array.

The liquid-nitrogen-cooled chevron serves as a radiant heat shield and a means of cooling down the gas particles incident towards the cryosorption panels. The accommodation coefficient for hydrogen on a 77 K surface is 0.5, which indicates a molecule for hydrogen will statistically transfer half its energy in a single collision with the surface. A molecule will bounce several times while migrating through this chevron, so thermal energy will tend toward the temperature of the chevron array.

Figure 4.5-4. Typical Section Through ORNL Cryosorption Panels

A second chevron assembly between the cryosorption panel and the liquid-nitrogen chevron is cooled to 15 K. The accommodation coefficient for hydrogen on a 15 K surface is ≈ 1 so all molecules leaving the 15 K chevron toward the cryosorption panel are at panel temperature. The sticking coefficient of the particles on the molecular sieve panels will then be unity for all practical purposes.

ORNL Calorimeter

The calorimeter consists of three beam targets, one for each source. These targets are made into one assembly and the cooling water flows through all three in a parallel flow circuit. The entire assembly is supported by two actuators, which lift the targets out of the path of the beam during injection.

Each target is made up of 42 copper swirl tubes arranged in two double-row banks shown in Figure 4.5-5. Each bank forms a panel which intercepts the beam at a 8.7×10^{-2} rad (5°) incident angle. The two panels intersect near the center of the beam so that when viewed from the side the target looks like a "V" laying on its side with the beam going into the open part of the "V". The geometry of the target is such that no part of the tubes is ever perpendicular to the beam. This allows the beam to be spread over as large an area as possible, which reduces the power density to 0.94 kW/cm^2 .

Figure 4.5-5. Typical Target Section of ORNL Calorimeter Assembly

The panels of swirl tubes are manifolded into a 7.6-cm (3-inch) diameter pipe at one end that bridges two supply pipes 12.7-cm (5-inches) in diameter, and into another 7.6-cm diameter pipe that bridges two 12.7-cm diameter discharge pipes at the other end. The flow in each target is from the apex of the "V" outward. The two supply and discharge pipes are made of 304L stainless steel. The supply pipes are welded into a single 20.3-cm (8-inch) stainless pipe which penetrates the vacuum chamber. The discharge pipes weld to a similar 20.3-cm pipe which returns the water to the auxiliary cooling water system.

When the beam is pulsed on to the calorimeter, the heat is absorbed by the water flowing through the tubes. The temperature rise caused by this heat input is detected and converted to beam power data.

ORNL Fast Valve

The fast valve is a focal-plane-shutter type of device, which is intended to limit the conductance between the vacuum in the beam-line vacuum chamber and the vacuum in the torus.

The fast valve consists of a 0.25-mm-thick stainless-steel belt with two rectangular openings, 82 cm by 30 cm. This belt is wrapped around two drums which are free to

rotate. One drum is driven in order to align the two openings in the belt during injection, or to rotate the openings out of phase to close the conductance path. There are two idler drums or rollers which serve to hold the belt away from the calorimeter. The entire fast-valve assembly, with the exception of the vacuum chamber "feedthrough" and drive motor, is mounted on a frame which can be attached and removed from the cryopanel support structure using manipulators and remote tooling*.

LBL/LLL Neutral Beam Line

The LBL/LLL neutral beam injection line includes all hardware items needed to produce and transport neutral beams to each of the four TFTR injection ports e.g., ion generator, accelerator, neutralizer, sweep magnet and ion-beam dump, neutral-beam dump, isolation valves, and pumping systems. The entrance aperture, which is limited by the EF and TF coils, will be 40-cm wide and 70-cm high.

The ion-generator will be modeled after the LBL/LLL "50-Ampere" sources: a low-voltage, high-current discharge producing a quiescent, uniform plasma, with an ion-current density of 0.25 A/cm^2 , over a 10-cm by 40-cm extraction area. The ion accelerator will be of the multiple-slot type, which has exceptionally good optics in the direction parallel to the slot; this dimension of the beam will be focused at the entrance aperture. In the direction perpendicular to the slots, the beam will be focused at the plasma target. Extremely good ion optics are mandatory, not only to meet the beam requirements, but to eliminate the difficulties caused by stray beams impacting on entrance structures. At present a four-grid extractor, with two accelerator stages and one electron-suppressor stage, is envisioned.

The energetic ions are neutralized by electron capture in a gas neutralizer. Un-neutralized ions are swept out of the beam with a magnetic field.

The gas flow to each beam line is estimated to be $10 \text{ to } 13 \text{ Pa}\cdot\text{m}^3$ per second, and a cryogenic pumping system has been designed to handle this gas. The maximum allowable pressure in the beam line is dictated by the requirement that the gas flow to the torus from all beam lines must be less than $0.133 \text{ Pa}\cdot\text{m}^3$ per shot; furthermore the pressure in the drift region between the sweep magnet and the entrance aperture must be kept low, so as to minimize reionization of the neutral beam.

A 1% beam loss by reionization for every $2.7 \times 10^{-3} \text{ Pa}\cdot\text{m}$ of gas traversed by the beam is anticipated. These requirements have led to a differentially pumped beam line with a pumping speed of $1.7 \times 10^3 \text{ m}^3/\text{sec}$ in the high-pressure section near the source, and $1.5 \times 10^3 \text{ m}^3/\text{sec}$ in the drift region.

* The ORNL proposal includes many interesting features and devices not included in this excerpt. Of general interest is a traveling impact wrench attached to each large flange, which provides remote sealing of the flange.

Although one would desire the sources to be as close to the entrance aperture as possible, the length required for the neutralizer, beam sweep magnet, and the vacuum pumping structure have resulted in a beam line that is about 8.5 m long. At that distance the entrance aperture subtends angles of $\pm 2.36 \times 10^{-2}$ rad (1.35°) by $\pm 4.12 \times 10^{-2}$ rad (2.36°) from the center of the extractor array.

Layout of LBL/LLL Sources Within Beam Line

Figure 4.5-6 shows the sources and beam trajectories superimposed on an outline of the beam line vacuum tanks. The source arrangement was laid out by considering the geometry of the torus, the magnets, and aiming radii.

A view down an injection centerline so laid out shows a window limited horizontally by the TF coils and vertically by the equilibrium field (EF) coils approximately 0.77 m apart. The window is approximately rectangular, with the long dimension vertical, and the curvature of the TF coils giving one side concavity inward and the other convexity inward. The rectangular geometry of the LBL source is particularly suited to the window. The three sources are mounted with their long dimension vertical. They are spaced 0.5 m apart on centers in the horizontal direction.

The outlines of the beam paths shown are determined by the source characteristics. Each source has emitting face dimensions of 0.10 x .50 m, with the grid rails parallel to the 0.10 m dimension. The intrinsic beam spread, i.e., the spread from a point emitter, is $\pm 5 \times 10^{-3}$ rad (0.3°) in the direction parallel to the rails (plan view in the layout), and $\pm 1.75 \times 10^{-2}$ rad (1°) in the direction perpendicular to the rails (elevation view).

The sources are focussed by two techniques. The rails are given a slight curvature to focus in the plan view. By offsetting the positions of individual rails in the last two grids, the sources are focussed in the elevation view. Since the two focussing mechanisms are independent, different focal lengths may be chosen in the two directions. Earlier iterations considering the overall design problems of providing space for equipment and pumping panels established a required focal length in the plan view of ≈ 8.5 m. The location of the focal plane in the elevation view is not critical.

In the plan view, the individual beams are diverging from one another, and at the point of tangency for $R = 2.7$ m, the outer beam centerlines are ± 0.16 m from the middle one, well within the ± 0.2 m tolerance.

Overall LBL/LLL Beam Line Design

Figure 4.5-6 shows a complete beam line. The tanks are of cylindrical construction with ≈ 15 mm thick walls. The support structure is mounted on wheels running on circular tracks for changing the aiming radius. The structure is centered on a pivot

Figure 4.5-6. Complete Two-View of LBL/LL Beam Line

XBL 759-7981

pin below the flex joint in the attachment duct. The part of the structure outboard of the isolation valve is mounted on another set of tracks and wheels for longitudinal motion. This permits parting of the entire beam line at the valve for servicing.

Overall beam line length from the torus is ≈ 9.5 m, of which the last $\approx .5$ m is due to the SF_6 -filled source-housing which is removable. Overall tank diameter is ≈ 4 m, and the total volume is $\approx 80 \text{ m}^3$. The overall height is less than the 6.5 m limit, leaving room for cryogenic fluid bayonets and other utility connections.

LBL/LLL Layout of Chambers

Each beam line consists of three chambers, arranged for differential vacuum pumping. They are the first and second in-line chambers, and the ion dump chamber. The first and second chambers are separated by a bulkhead on which is mounted the deflection magnet. The gaps in that magnet also form the molecular flow conductance for thermal gas between the chambers. The ion dump chamber and the second in-line chamber are separated by a baffle which has three slots for passage of the D^+ ions bent out of the beam by the magnet. The slot lengths are 1.2 m, set by bent ion trajectories. Focussing properties of the magnet permit slot widths of 0.1 m. The exact shape and location of the slots will be determined in final design by consideration of effects of fringe TFTR fields on the trajectories. The slots form the conductance for gas flow between the dump and second in-line chambers.

The first in-line chamber is made ≈ 3 m long to accommodate the 2-m long neutralizer ducts and allow an additional meter of drift space. The additional meter permits dispersal of the source gas load by collisions just downstream of the duct mouths.

LBL/LLL Ion Source Design Description

The ion source proposed for this application is a straightforward development from the LBL "50 A" ion sources currently in use at LLL and PPPL. In this source the plasma is produced by a diffuse low-pressure, high-current discharge with a distributed, thermionically emitting cathode, which consists of a ring of hot-tungsten filaments. The source filaments are mounted on a separate sub-assembly, which can be replaced with remote handling gear without the necessity of dismounting the entire source.

LBL/LLL Ion Accelerator Design Description

The ion accelerator coupled to a plasma source consists of a 60% transparent, four-electrode, multiple-aperture assembly consisting of an array of 44 slots, each 0.54 by 10.0 cm. The shapes of the electrodes forming these slots, and the potentials on them, are generated by a computer optimization code, to produce an ion beam with minimum angular divergence, together with minimum interception of secondary ions, electrons, and neutrals by the accelerating structure. Figure 4.5-7 shows a cross

TFTR001,DUMP 0,TFTRM15 ,SOLUTION=0,ERROR=0, 19 AUG 75

V = 120000

101409

-2300

0

ION CURRENT DENSITY = $0.28 \text{ A/cm}^2 \text{ (D}^+)$

TRANSPARENCY = 0.6

BEAM DIVERGENCE ($\sqrt{2} \theta_{\text{RMS}}$) = 0.44 DEGREES

Figure 4.5-7. Computer Beam Trajectories Through Proposed LBL/LLL Accelerator Structure

section of such a structure. The calculated beam divergence is smaller by more than a factor of 2 than the required $\pm 1.75 \times 10^{-2} \text{ rad (1}^\circ)$. Interception of secondary particles by the structure, while not yet minimized, appears to be tolerable: temperature rises during an 0.5 sec beam pulse should be less than 775 K.

We envisage an accelerator structure as an array of shaped molybdenum or tungsten bars, free to expand at one end, and attached to a support structure with adjacent water cooling at the other end. The insulators are ceramic, and the entire support structure, with insulators, is brazed together.

LBL/LLL Ion Removal System

A deflection magnet is located between the first and second in-line chambers, as shown in Figure 4.5-6. In fact, the gaps between poles and cores of the magnet form the molecular flow conductance between the two chambers. The magnet's function is to remove charged particles from the beam in a controlled manner, rather than have them scattered by high fields further downstream.

The magnet's construction is shown in Figure 4.5-8. Each magnet has three gaps, 15 cm long. Between the gaps are two wire wound cores, and at the ends are two wound pole pieces. The flux returns through the rectangular iron frame. The pole and core faces are trapezoidal. Their height at the beam entrance is $\approx 51 \text{ cm}$, with exit height $\approx 71 \text{ cm}$.

Figure 4.5-8. Three-View of LBL/LLL Magnet

XBL 759-7979

The iron return path makes the stray field of the magnet negligible at the nearest plasma location (≈ 5.5 m away). Induced fields due to fringe fields from the torus are also negligible. Roughly, the iron in the magnet has a volume equivalent to a sphere of radius 27 cm. Thus, the induced dipole induction 550 cm away is:

$$B \approx \left(\frac{27}{550}\right)^3 B_c \approx 1.2 \times 10^{-4} B_o.$$

where B_o is the (vacuum) fringe induction at the location of the magnet. Thus, a fringe induction of the order of 0.01 tesla will result in a perturbation of only $\approx 1.2 \times 10^{-6}$ T at the nearest plasma location.

LBL/LLL Cryopumping

Each beam line is divided into three chambers, as shown in Figure 4.5-6. The inner walls and bulkheads of these chambers are lined with cryogenic condensation panels, which provide a pumping speed of $1.7 \times 10^3 \text{ m}^3/\text{sec}$ in the neutralizer chambers. The cryogenic condensation panels consist of liquid helium cooled surfaces at 4.2 K, surrounded by liquid-nitrogen-cooled surfaces (on the outside) and chevrons (on the inside), at 77 K. The panels have a pumping speed of $50 \text{ m}^3/\text{s m}^2$.

LBL/LLL Calorimetry

The total energy in each beam will be measured by monitoring the rise in temperature of the water which cools the beam-stop calorimeters. An attempt to obtain some spatial resolution by this method may be deemed feasible and worthwhile in the final design. However, detailed beam measurements will be obtained by optical spectroscopic techniques.

The function of the calorimeters described here is primarily to provide a beam stop for beam operation when the torus plasma is not present. Such operation will be required for conditioning and tuning up the sources. The calorimeters will also provide a measure of the total beam energy. Since the calorimeters will sometimes be used when the bending magnets are not on, and since the beam power losses cannot be predicted with certainty, the design heat loads are based upon source design extracted current density. This gives the highest possible intensity at the calorimeter, for conservative design. A calorimeter is placed just past the bending magnet, where the three beams are still separate; as shown in Figure 4.5-6, each section then intercepts the beam from one source.

The calorimeter plates are inclined so that the beams of width ≈ 10 cm are spread over a one-meter length. The resulting maximum heat flux is then 1/10 of the maximum extracted flux. To further reduce the surface flux intensity, the calorimeter faces are serrated with a $\Pi/3$ rad (60°) saw-tooth pattern running along the one meter length. Molybdenum was chosen as the calorimeter material. Its surface temperature rise

would be approximately 1,200 K, if the face were not serrated. Each slab is 1.1 m long x 0.7 m high x 20 mm thick, and has a mass of 157 kg. The total heat input during the pulse is approximately 7.5 MJ, and the bulk temperature rise after equilibration is 173 K.

Each slab is cooled between shots by low-conductivity water flowing in channels at the back side. Allowing for an average 50 K temperature rise in the water, approximately 36 kg of LCW would be required per shot, for a total of approximately 0.11 m³ per shot per beam line.

The basic calorimeter design is shown in Figure 4.5-9. The three slabs are mounted on a common frame which pivots upward for shots into the torus. The heated face, which would otherwise be exposed to the cryopump panels is shielded by a reflective sheet of electro-polishing copper which is mounted to the same frame. In its upward position, the calorimeter serves to intercept the relatively small negative ion currents bent upward by the bending magnet. A titanium plate is provided for the same purpose in case the magnet is operated while the calorimeter is in the down position. The exit cooling water lines for the three slabs are kept separate until they reach a point where temperature may be measured, so that the energy of the beams may be determined separately.

Neutral Injection Cell

A neutral injection cell, separate from the TFTR test cell, is necessary so that beam lines can be tested and maintained and so that sources can be conditioned off-line. The neutral injection cell will be used first to test the beam lines as they arrive from the manufacturer. For this purpose the cell will have the full power for one beam line (20 MW), connections to the vacuum and cooling services and sufficient shielding (concrete, 1.2 m thick) to attenuate neutrons from D-D reactions within the beam line. The facility will also have the necessary beam line test equipment and access to the control and data acquisition computers.

The power and service facilities will also be used for off-line conditioning of new and rebuilt ion guns (sources and accelerators). This procedure will be going on nearly continuously, so that at least 3 guns can be kept in standby condition; all the guns mounted on any one beam line can be replaced at one time.

The maintenance on the beam line will be divided between regular maintenance jobs to be performed in the neutral injection cell and large repair jobs, which will be performed in the test cell. The hot cell will have remote handling equipment capable of repairing large, complex, and radioactive devices; there is no need to duplicate them in the neutral injection cell. On the other hand, the neutral injection cell will have the necessary remote tools for the routine replacement of ion guns, the checking for and repairing of leaks in cryosystems, etc. Intricate jobs, such as the replacement of individual filaments or the realignment of accelerator grids will be done outside the tritium area in "cold" shops. In the tritium phase of operation such items will not be reused.

Figure 4.5-9. Two-View of LBL/LLL Calorimeter

XBL 759-7985

4.5.2 NEUTRAL BEAM PULSED ENERGY CONVERSION SYSTEM

Major Design Considerations

The conceptual design of the power system for the TFTR neutral beams emphasizes flexibility and operational control. The neutral beam sources are presently in an early development stage. Operational parameters are, therefore, expected to vary significantly over the next two years. The specifications developed by the Neutral Beam Steering Committee are based on early experimental performance and will require considerably more operational data to become final. One requirement, however, is the need for operational flexibility and independent control of each source.

The neutral beam power system essential specifications are listed below in summary form.

- a. The power supply system must supply energy to four beam lines composed of twelve sources; i.e., three sources per beam line.
- b. The system must have a high-voltage pulse capability of 120 kV with a one-percent voltage regulation.
- c. The peak power input to the power-supply system will be 110 MW.
- d. The maximum pulse width will be 500 msec., with a rise and fall time of 20 microseconds.
- e. The operating cycle duty will be one pulse per five minutes during experiments or operations, and one pulse per 30 seconds while conditioning or seasoning the neutral beams. This conditioning will occur at the beginning of the day, or after repairs, to ensure that the neutral beams remain stable and to lessen the probability of an inadvertent breakdown.
- f. Independent voltage control must be provided for each source, since variations in construction, vacuum or gas conditions may necessitate that each source operate at two different voltages.
- g. The power system must be compatible with either the ORNL or the LBL/LLL source.

Several major design approaches were investigated to meet the above criteria. These were:

- a. Phase-controlled ignitrons or silicon controlled rectifiers (SCR's) to rectify the output of high-voltage transformers.

- b. Tap changers, Inductrols,* or saturable reactors on the primary of high-voltage power-supply transformers.
- c. Ignitron or SCR switches in the output of the high-voltage power supplies.
- d. High-voltage vacuum-tube switches and regulators in the output of the high-voltage power supplies.

In theory, all of the above approaches and combinations thereof can be made to work. The problem is to determine which design approach has the highest probability of success within the time and money constraints imposed on the project.

The selected power supply design approach is similar to existing designs for high-power radar transmitters and accelerators. The power supply design is being used for about twenty different neutral beam sources in operation at various CTR-funded laboratories. The selected design, corresponding to approach d and part of b listed above, uses a vacuum tube to apply, regulate, and limit the energy to a neutral beam source and uses tap changers of Inductrols to provide coarse voltage control on the primary of the high-voltage power supply.

The number of high-voltage dc power supplies (HVPS) required to power the twelve sources was evaluated at length during the Conceptual Design Study. An individual HVPS could be provided for each neutral beam; i. e. twelve HVPS. At the other extreme, a single HVPS could be used to power all twelve sources. A cost/size/weight analysis indicated that although the per-unit cost of the transformer/rectifier assemblies decreased for fewer large-size power supplies, the cost for peripheral voltage-controlling equipment increased. Also, the size and weight for a single HVPS (over 250 tons) was such that movement, assembly, installation, and maintenance problems (and costs) increased significantly. Since the Neutral Beam System is one of the most developmental aspects of the TFTR Project, an effort has been made to reduce the technical risk and development required for the HVPS that supplies power to it. An individual HVPS for each of the twelve neutral beam sources has been selected.

The choice of using series or parallel combinations of vacuum tubes or of developing a single high-voltage vacuum tube has been explored in depth and will continue to be a design option over the next two years. A single high-voltage vacuum tube that could both switch and regulate the voltage to the neutral-beam-source grids would be the most desirable operating option and will be selected if available within program constraints.

*Inductrol — General Electric trade name

Reference Design

Figure 4.5-10 is a simplified schematic of the conceptual design for the High-Voltage Power Subsystem (HVPS). Power derived from the motor generators is used to operate the following functional elements:

- a. Auxiliary power: Power required to supply the source filaments and to power the arc supply which ionizes the injected gas.
- b. Source grid power: Power in the form of regulated high-voltage pulses supplied to the source accelerating grid and other auxiliary grids.

The auxiliary power is fed to the source via a high-voltage isolation transformer, since the source filaments and arc chamber float at 120 kV during the pulse. This auxiliary power represents about five percent (5MW) of the total neutral-beam-source pulse power.

The source accelerating grid and other auxiliary grids require high-voltage power. A variable-voltage transformer is used in the 13.8 kV primary side of the high voltage transformer for varying or adjusting the voltage to the HVDC power supply. The variable voltage source will be either a tap-changing transformer or an Inductrol type

Figure 4.5-10. Simplified Schematic of the Conceptual Design

variable transformer. This variable voltage provides operating flexibility in breaking in a new source, or in initial start-up for an experimental run. The variable-voltage ac power is supplied to the HVDC transformer rectifier assembly (T2) where it is converted to approximately 135 kV at full load current. The capacitor bank C1 supplies transient power during initial turn-on of the source and during transient control changes.

The high-voltage vacuum tube (V1) is used to switch and regulate the voltage to the source grids. The use of a series tube provides a clean and positive disconnect from the energy source when a fault occurs in a neutral beam source. This protection is backed up by a shunting crowbar (S1), should the high-voltage vacuum tube fail to operate.

The regulated voltage bus at the no. 1 source accelerator grid is reduced in voltage by high-voltage regulating tubes to provide various lower voltages to the other auxiliary grids.

Operation of the system is under control of the Central Instrumentation Control System. For normal experimental operating modes, start-up of the neutral beam system, power levels, gas, and vacuum system will be under the direct control of the central computer. Main timing signals will originate at this location and start timing sequences which are built into the neutral beam equipment. A second level of status will be displayed in the neutral beam local control area. The function of the local control system is to provide more detailed monitoring of parameters necessary for equipment maintenance, fault location and analysis, and minor (fine trimming) adjustment. In addition, the local control system provides off-line testing and operation of a neutral beam line or a portion of the power system without interfering with experimental operations.

Status of Design Alternatives and Extensions

Some of the design alternatives mentioned on page 4-237 are still being evaluated. The saturable-reactor voltage control with its current-limiting feature is being tested on a small scale by the Lawrence Berkeley Laboratory and this configuration may be implemented in one of the LBL test stands. A shunt regulator which uses high-power tetrodes may also be tested by LBL in conjunction with this design.

ORNL is constructing a test stand for neutral beam development testing that will follow a concept similar to the reference design described on page 4-239 and above. The switch tubes that will be used are existing units that will be connected in series to hold off the voltage. Voltage control of the ac primary is by a variable transformer.

Operation of 150 kV Sources

One of the design options considered during the conceptual design study was the use of 150 kV beams and associated equipment. This design option was dismissed for two

significant reasons. First, the voltage rating of the equipment at constant power causes an increase in cost. Second, the impact of operating at 150 kV voltage level was expected to result in a reduction in the reliability factor. Although no quantitative data have been developed, a qualitative judgment is that the failure rate is a function of the voltage stress. Reliability improvement could be achieved by derating the power-supply components. Operation at 20-30 percent under-rating will improve the reliability of the system significantly, but this increase in reliability would be accompanied by a further increase in cost. The 120 kV approach was therefore adopted for the conceptual design.

RF Heating Systems and NB Power Supplies

All of the major experimental plasma machines at PPPL over the past 10 years have utilized some type of auxiliary RF-heating system. Lower hybrid (microwave) and ICRH (high frequency) systems are being investigated for use on PLT and PDX. Recent experimental results at PPPL have given impetus to the desirability of incorporating an RF-heating system. While there has been no authorization to proceed with an RF-heating system, it is appropriate to make provision for an optional RF-system should the requirement become established in the near future.

Present indications are that the size of an RF system would probably approach that of the presently proposed neutral beam system. The power level into the plasma to meet physics requirements would be in the range of 30-50 MW. The frequency range could be anywhere in the HF range of around 50 MHz (for ICRH) or in the microwave range of 1-2 GHz (for LH). The transmission of RF power is rather costly, due to the losses encountered, particularly in the microwave range. Equipment should, therefore, be housed as close as practical to the tokamak. The use of the neutral beam power supplies for an RF system was investigated to determine the cost differential for this added flexibility. The HVPS, as presently proposed for the TFTR neutral beams, will deliver 135 kV at 67 A. If this supply were designed for use on lower hybrid or ICRH systems, it would need to supply the following:

- a. For LH, 82.5 kV at 110 A.
- b. For ICRH, 35 kV at 265 A.

Since the total MVA rating is the same for all three applications, no change would be required on the ac side of the rectifier transformer. For the LH application, it would be necessary to connect rectifier bridges in parallel instead of series. This could be accommodated in the design with no change in rectifier ratings. An interphase transformer would be required within the HVPS tank for use when the bridges are paralleled.

For the ICRH application, the parallel arrangement could be used with the lower voltage (35 kV) provided by tap changes on the voltage-control transformer. The diode ratings would have to be increased to handle the higher current and the transformer secondary copper would have to be larger for the same reason.

To provide the additional flexibility in the neutral beam transformer/rectifier assemblies would increase the base cost by \$500K to \$1000K.

References

1. LBL/LLL CTR Staff, Lawrence Berkeley Laboratory Report LBL 3296 (August 1975).
2. UCC-ND Engineering Staff, Oak Ridge Holifield National Laboratory Special Report ORNL-CF-75-9-15 (September 1975).

4.6 DIAGNOSTIC SYSTEMS

4.6.1 EXPERIMENTAL REQUIREMENTS

Operational Modes

From the experimental physicist's point of view, the TFTR device will operate in three modes.

- a. A hydrogen or deuterium phase during which all device systems will have been tested, best operational conditions of the ohmic plasma and neutral beams will have been obtained and the diagnostic system will have been thoroughly tested.
- b. Tritium operation at a level as near as possible to the optimal conditions obtained before. During these high power pulses, very large amounts of data must be collected. The limited number of tritium discharges at full power established the requirements for data acquisition.
- c. Hydrogen or deuterium operation intermixed with the tritium operation. These discharges will be used to investigate new ideas, provide comparison data, check out beam changes and possibly allow some diagnostic investigation without the high prompt γ -ray background flux. The relative interchangeability of hydrogen and tritium will depend on the release of neutral particles from the surface of the vacuum vessel after a change of gas.

The planning of the diagnostic system is predicated upon having a complete system ab initio, so that the discharge behavior is well understood and the diagnostics are proven reliable before tritium operation.

Experimental Apparatus

The diagnostic system will have to yield as much experimental information as possible during a D-T discharge. (The diagnostics are listed in Table 4.1.12-1). The data

acquisition system must be consistent with this requirement and be able to handle more than five million words of information per discharge. One-tenth of these would be permanently stored.

Many of the diagnostics will evolve from the equivalent apparatus to be used on PLT; they will be more sophisticated in number of sensors and in data recording capability to make up for the limited number of D-T discharges available and the slower rate of discharging. For example, the measurement of the radial dependence of the electron temperature across the plasma by Thomson scattering should be repeated ten times during the discharge and the x-ray pulse height analysis system should make a ten-point radial scan concurrently in one discharge. The diagnostics also require better built-in internal calibration methods. The size and density of the plasma require that new techniques for measuring the ion-energy and current-density distributions be developed, and that far infrared laser techniques be adapted for use in a multichannel interferometer for measuring electron density. Also, some entirely new diagnostic techniques will have been developed before the startup of TFTR.

Thus, detailed design of individual components is subsidiary to the setting up of a flexible environment within which many sophisticated diagnostics can co-exist. A necessary requirement is that "hands-on" maintenance should be possible on all components except those which mate up with the vacuum vessel. The most convenient arrangement is to have every possible component in a shielded basement, with as few components as possible above the Test Cell. Above the floor 0.6 m thickness of high density shielding is needed.

To determine the likely requirements for space in the Test Cell, a layout has been prepared and is shown in Figure 4.6-1. It is not a unique arrangement and allows for the use of some techniques, particularly x-ray techniques, to be used in the Test Cell without shielding, in the early hydrogen phase of operation. Access ports on the vacuum vessel have been allocated for both ion cyclotron heating and lower-hybrid heating. Their symmetry requirements restrict the choice of diagnostic locations, though the use of rectangular holes on the outside midplane would largely remove this restriction. The large distance, ~ 11 m, of many of the components from the toroidal axis is set by the difficulty of shielding large volumes from the magnetic fields. It is the distance at which the stray vertical field is approximately 10^{-2} T. The space requirements in the shielded basement below the Test Cell are almost as demanding, requiring approximately 93 m^2 (1000 ft^2) under the device and 280 m^2 (3000 ft^2) elsewhere.

4.6.2 PRINCIPAL INTERACTIONS WITH OTHER SYSTEMS

The principal interactions of the diagnostic system with other systems in the TFTR design are with the buildings and with the tokamak device. The other interactions are straightforward and, except in the case of the computer, are relatively inexpensive.

Figure 4.6-1. Conceptual Layout of Diagnostics in Test Cell (Oct. 75)

Interaction with Building Design

The prime requirement of an environment within which the bulk of the diagnostics area can be maintained "hands-on" has been met by the basement facility with its shielding ceiling. This shielding reduces the activation to an acceptable level and cuts damage to normal electronic components considerably. No operation of the device, even with ohmic hydrogen discharges, will be done while personnel are present in the basement because of the potential flux of high energy x-rays from the plasma itself. Electrical safety will be ensured only by isolators outside the Test Cell wall. The computer and the physicists' working area (control room) are designed to accommodate thirty physicists during an experimental run.

Interaction with the Tokamak Device

The diagnostic requirements for numerous access points and for straight viewing lines to the plasma interact in many ways with the device design, especially with the vacuum vessel but also with the coils' arrangement and support and with the device support structure. The mating of diagnostics with the vacuum vessel is complicated by the need for remote manipulation and by the need to keep the high-density shielding close to the device as complete as possible. The making of seals under the torus is particularly difficult; in the Conceptual Design, welds or cryo-seals have been proposed. An alternative scheme is shown in Figure 4.6-2 for the x-ray pulse-height-analysis system. The small vacuum tubes that are necessary in order to pass between the coils and to allow good shielding are shown mounted on large adapter flanges. (The tube arrangement is needed to cut the severe background signal caused by prompt γ -photons from the vacuum vessel wall).

In this design, two large ports and two small ports have been made on the top and bottom of the vacuum vessel. Their dimensions have been chosen to allow access for any of the diagnostic or heating equipment already specified and could, therefore, be common to all vacuum vessel sections between TF coils. All vacuum seals could be conventional hard seals made up with bolts inserted remotely. Space has been left on the large flanges for nut-plates to be used rather than tapped holes in the vacuum vessel flanges with the attendant possibility of seized bolts. However, to allow access for the adapter plates to the vessel from the basement, the design of the device support structure and the basement ceiling must allow shielding to be removable in the form of plugs. The support structure should have basically circular symmetry, with radial girders under the TF coils; interference would occur only at a few locations because of the rails necessary for the splitting of the device.

There is now very good access above and below the device following careful alignment of the vessel support rail and the shear panel of the TF coil frame. Future interactions on the locations of the EF coils may further improve the access. This is shown clearly in Figure 4.6-3 which shows a speculative arrangement of the Thomson Scattering. The viewing system is an $f/2$ lens focussing onto a fiber-optic array which takes the light to

Figure 4.6-2. Alternate Scheme for the X-Ray Pulse Height Analysis System
 Showing Tubes Passing Through Coils and Structure

Figure 4.6-3. Speculative Arrangement of Laser, Windows, and Light Collecting System for Thomson Scattering

the basement detector system. (Both radiation-resistant glass and advanced fiber technology will be necessary to make this arrangement work.) The laser fires horizontally through a window 12.2 m from the minor axis of the vacuum vessel. This dump will extend into a hole cut into the steel structure of the TF coils and will have to be removed through the porthole on the outside midplane.

Figure 4.6-4 shows a possible arrangement of an adjustable limiter using the same basic set of holes at another location on the vacuum vessel. Two separate blades are shown at the top and bottom to allow for adjustment to conform to the various suggested plasma scenarios. The detailed design of the blades themselves, assumed to be curved 1.3 cm (0.5-in) tungsten backed by molybdenum blocks, must await experience of the operation of limiters in PLT. Mechanically, they could be withdrawn through the top and bottom flanges by careful design of a hinge mechanism at the blades' joints with their support tubes.

At this time, no detailed analyses have been made of the effects of the penetrations through the armor-cladding of the vacuum vessel that would be necessary for the diagnostic access. The rearrangement of local shielding around all the access ports and their associated diagnostic protuberances also has not been considered extensively.

4.7 CIVIL, ARCHITECTURAL AND STRUCTURAL SYSTEMS

4.7.1 SUMMARY

General Description

Facilities necessary to house the TFTR device with its auxiliary and support systems, utility and service systems and site improvements are to be constructed on an open tract of land lying to the east of and abutting the main complex of the Plasma Physics Laboratory buildings at the Forrestal Campus, C-Site. Additional facilities to house TFTR Staff will be constructed at C-Site in conjunction with the construction of the TFTR facilities. Construction is scheduled to begin before the end of calendar year 1976 in order that the facilities may be ready for occupancy and installation of the device and its auxiliary systems by 1978.

Design Concept

The design concept developed for the TFTR Project facilities responds to the requirements and constraints governing facility developments on the site as well as the basic functional, experimental, and operational requirements of the TFTR program. In addition to housing the device and its related auxiliary systems, the facilities provide radiation shielding for personnel safety and for protection of the surrounding environment.

Figure 4.6-4. Potential Alternate Arrangement of Movable Limiter Blades

The decision to locate the main working floor level of the Experimental Area, including the Test Cell, 3.0 m below existing ground level is intended to accomplish two fundamental design objectives:

- a. To accommodate the full height required for Experimental Area high-bay operations within the allowable height limit for construction at the site.
- b. To utilize earth for shielding insofar as possible in order to minimize the amount of concrete required in the construction.

The resulting spatial envelope allows considerable latitude for the development of the Test Cell structure, for the planning of overhead cranes and remote handling systems, for the arrangement of water lines, electrical and diagnostics equipment, and other system components that must be located around the TFTR device.

The buildings are arranged on the site so that two exterior walls of the Test Cell are kept free of any abutting construction in order to facilitate future expansion of the Experimental Area. Site areas to the south and west of the main Test Cell have been reserved for this expansion.

The Experimental Area is to be constructed of reinforced concrete with design and thickness of the various structural elements to be determined on the basis of analysis of both structural requirements and radiation shielding considerations.

Material from building excavations will be utilized for the construction of earth berms around the Test Cell in order to meet overall radiation shielding requirements. The utilization of the on-site excavated material for berms permits a substantial reduction in the amount of concrete required for radiation shielding above grade and saves the cost of transporting the excavated material off site. Because of the reduced thickness of concrete walls required with this design, the problems of dismantling the structure and disposal of the activated elements after the conclusion of the experimental program are simplified.

The combination of earth and concrete shielding also provides considerable flexibility for making changes and adjustments to the shielding design in the future. Areas to be occupied by personnel during periods when the device is being pulsed have been concentrated in the Control Building which is isolated from the Experimental Area by shielding walls and labyrinths, with tritium sealing doors for spaces with cleanup systems.

The experimental areas are organized within a large high bay shield structure which is designed to adjust to changing needs and development of the experimental programs. The spaces within this structure include the Test Cell, Neutral Beam Test Cell, Hot Cell, Transfer Bay, and the Mock-up and Assembly Bay with requisite observation, control and support areas. Power buildings and site support facilities are planned in accordance with the respective functional and operating requirements and the proximity requirements for power supply and distribution systems.

Site

The site is relatively open in the areas where construction is planned with little variation in grade. The open area is bordered on three sides by wooded areas with mature stands of trees, and on the fourth side by an existing paved service road which will be improved to provide access to the site and to the adjacent Plasma Physics Laboratory service areas.

A series of five exploratory test borings taken at the site in August, 1975 indicates the top of a stratum of weathered shale approximately 3.0 m below the existing surface with firmer rock strata alternating with sandstone below. Analysis of the material recovered from the test borings suggests that excavation can probably be carried out to depths of 7.6 m to 9.1 m with conventional excavating equipment and that blasting will not be required except in isolated instances.

In addition to the subsurface conditions, other major site constraints include the following:

- a. 18.3 m limitation on the height of any structures above existing grade.
- b. A requirement that the TFTR device be located at the center of a 250 meter diameter circular "exclusion area," within which personnel access will be strictly controlled.

Facility Requirements

TFTR Project facilities include the following:

- a. The Experimental Area, consisting of the main Test Cell housing the device; a Neutral Beam Test Cell; a Hot Cell for performing maintenance operations on activated components; observation and control galleries; a Mock-up and Assembly Bay; Transfer Bays for performing maintenance operations on slightly activated components; a shielded bay for crane and manipulator parking; air lock areas for environmental control of entrances; and basement support areas with space to house equipment for Experimental Area and auxiliary systems.
- b. The Control and Mechanical Equipment Building with computer, space for the central control console, data storage, offices, laboratories, and mechanical equipment.
- c. Power buildings including the Motor Generator Building, Field Coil Power Conversion Building, and Neutral Beam Power Conversion Building.
- d. Office and laboratory space for staff.

- e. Auditorium.
- f. Expansion of the existing cafeteria.
- g. Addition to the technical shop.
- h. Construction of utility and service systems and support structures.
- i. Site improvements including parking, roads, lighting, and landscaping.

4.7.2 DESIGN DESCRIPTION

The Site Plan is shown on Figure 4.1.11-1. Figures 4.7-1 through 4.7-6 provide floor plans, sections and elevations for the Experimental Area, Control and Mechanical Equipment Building, and power buildings corresponding to descriptions in the subsections that follow.

Site Plan

The main TFTR buildings, including the Experimental Area, the Control and Mechanical Equipment Building, and the power buildings are located within a 250 meter diameter (820 ft.) exclusion area which has been defined on the basis of radiation analysis to meet radiation safety requirements.

The exclusion area has been located with its center approximately 168 meters (550 ft) east of the nearest existing building at C-Site in order to provide space for future expansion at C-Site without encroachment of the buildings in the exclusion area. The area is completely fenced in with personnel access limited to a single point which is controlled by a manned guard station, 24 hours a day.

The Test Cell is located at the center of the exclusion area with the Power Buildings, the Control and Mechanical Equipment Building, and other experimental areas surrounding it having been arranged on the basis of functional, operational, and cost considerations and requirements. The office and laboratory space, auditorium, and cafeteria expansion have been located adjacent to the existing C-Site office facilities. The technical shop space will be added adjacent to the existing C-Site shop area.

Electrical power will be supplied from the existing 138 kV service by means of a new substation and transformers to be installed in the existing C-Site switchyard. Power will be reduced to 13.8 kV for underground distribution to the motor generator sets and to 4.16 kV for underground distribution for auxiliary requirements.

The site plan provides space for expansion of existing cooling tower and pumping capacity and an addition to the existing C-Site Maintenance Building to accommodate TFTR boilers, chilling equipment and pumps. Some expansion of existing C-Site

FLOOR PLAN AT LEVEL -16'

Figure 4.7-1. TFTR Complex - Floor Plan at Level -16'

FLOOR PLAN AT LEVEL 0'

Figure 4.7-2. TFTR Complex - Floor Plan at Level 0'

FLOOR PLAN AT LEVEL 32'

Figure 4.7-3. TFTR Complex - Floor Plan at Level 32'

SECTION A-A

SECTION B-B

Figure 4.7-4. TFTR Complex - Sections A-A and B-B

SECTION C-C

SECTION D-D

SECTION E-E

Figure 4.7-5. TFTR Complex - Sections C-C, D-D and E-E

WEST ELEVATIONS

NORTH ELEVATIONS

NORTH ELEVATIONS

Figure 4.7-6. TFTR Complex - West and North Elevations

parking areas will be required, along with additional roads, walks, and landscaping for the TFTR Project. A new service road connecting with the existing C-Site service road is planned as part of the TFTR Project. This road will provide access to the TFTR site and exclusion area and will also provide service access to the TFTR-related buildings located outside the exclusion area.

Experimental Area

Test Cell

The Test Cell which is a large, heavily shielded concrete structure, 36.6 m x 33.5 m x 13.7 m high (120 ft x 110 ft x 45 ft high), houses the TFTR device with its magnetic field coils, neutral beam lines, diagnostics systems, and auxiliary equipment. In addition to physical accommodation of the device, the dimensions of the space have been determined on the basis of operational considerations, including space required for separation of the device for maintenance purposes, access requirements for remote maintenance, clearances for crane operations and remote handling equipment, and space required to facilitate the removal and/or replacement of major components such as a toroidal field coil, a section of the vacuum vessel, or a neutral beam line. The initial installation will include four neutral beam lines with space reserved for the addition of two more lines in the future (if needed).

The Test Cell is constructed of reinforced concrete with wall, roof, and floor thicknesses having been determined by both structural and radiation shielding requirements. The east and west walls of the cell are designed to support a crane rail with a 90.7 tonne capacity crane bridge and the floor is designed to support the static weight of the TFTR device (approximately 1800 tonnes) in an 18.3 m x 18.3 m center bay, and a live load of 2400 Kg per square meter at any other point over the Test Cell floor area. The present design calls for conventional steel reinforcing throughout the concrete structure with the exception of the 18.3 m x 18.3 m bay supporting the device which will be reinforced with non-magnetic stainless steel bars with no closed loops. Concrete thicknesses of the various structural elements are as follows: roof slab, 1.2m; walls, .9 m for the west and south walls (with an additional 4.6 m of earth berm to provide required shielding); 1.8 m for the east wall; and 1.2 m for the north wall consisting of 0.6 m permanent wall and 0.6 m of removable shielding (the wall of the adjacent Neutral Beam Test Cell provides an additional 0.6 m of removable concrete shielding, making the total north wall thickness 1.8 m).

The 90.7 tonne capacity crane bridge carries two 45.4 tonne capacity trolleys and a shielded cab with operators and manipulators for the remote maintenance system. Railroad type tracks with a 127-tonne load capacity run the full length of the west wall of the Test Cell, and extend through the Transfer Bay to the Mock-up Assembly Bay to provide for the transport of large, heavy components for assembly, maintenance, and replacement purposes within the Experimental Area.

In order to facilitate control of crane and remote handling operations, observation galleries with radiation shielding windows are provided at several locations in the Test Cell. A shielded crane parking bay is provided in the space above the Neutral Beam Test Cell. The crane bridge will be moved into this parking bay in order to reduce activation of components through exposure to neutrons from the device during periods of pulsing.

Access to the Test Cell for equipment is provided by means of shielded concrete doors of movable concrete hatch covers in the floor. The hatches are located so that the covers may be removed by means of the overhead crane. Shielded bulkhead doors are provided at several locations for personnel access.

For tritium operations, the Test Cell must be constructed as an air-tight sealed chamber with all openings and penetrations requiring special sealing to minimize the possibility of leakage or discharge of tritium-contaminated air or water to the outside environment.

The Test Cell ventilation system is designed to function in either of two different modes with continuous monitoring of air at all times when the system is functioning. During non-operational periods (when the device is not being pulsed) and prior to tritium operations, the Test Cell will be forced ventilated and supplied with conditioned dehumidified air at the rate of two air changes per hour with approximately 20% of the circulated air being discharged to the outside after passing through absolute filters. Fresh air is introduced as required to maintain system balance. Quantities are to be sufficient to dissipate the heat generated by the various operations and equipment in the Test Cell with filtering to clear the cell air insofar as possible of any residual radioactivity, fumes, or dust.

During experimental operations involving tritium, the ventilation system will function as a closed, sealed system with no discharge to the outside. In this mode, the cell atmosphere will be maintained at a pressure equivalent to 2.54 cm of H₂O below normal atmospheric pressure in order to minimize the possibility of leakage of tritium-contaminated or highly radioactive air to the outside.

In the event of an abnormal release of tritium to the test cell, the air will be recirculated and passed through the tritium handling and cleanup system with continuous monitoring of levels of contamination. The ventilation system is maintained in this operational mode until the tritium level is brought below the MPC (Maximum Permissible Concentration).

For tritium operations, the Test Cell must be constructed as an air-tight sealed chamber with all openings and penetrations requiring special sealing to minimize the possibility of leakage or discharge of tritium-contaminated air or water to the outside environment.

Neutral Beam Test Cell

The Neutral Beam (NB) Test Cell provides space for the testing of neutral beam lines and components and is planned to house a single operation beam line with three ion sources, beam dumps, vacuum pumping system, and a dummy load.

The NB Test Cell is 13.1 m wide, 20.7 m long, and has a ceiling height of 8.5 m. The floor of the NB Test Cell is at the same level as that of the adjacent Test Cell with both spaces being served by the rail system for the transport of heavy components.

Construction is of reinforced concrete throughout, with the floor slab and roof each 1.2 m thick, and the west and south walls (the latter separating the NB Test Cell from the Test Cell) 0.6 m thick. The 0.6 m thick common concrete wall between the NB Test Cell and the Test Cell will be augmented by 0.6 m of movable concrete shielding on both sides in the future to provide a total of 1.8 m of concrete shielding between the NB Test Cell and the Test Cell. The additional shielding may be required for safety reasons if the NB Test Cell is to be occupied during tokamak pulsing with deuterium-tritium.

The structural floor slab and the 127 tonne rail system are designed to support and permit movement of the 45.4 tonne weight of a fully assembled beam line. The NB Test Cell is structured to support a 22.7 tonne capacity overhead crane and an overhead manipulator for remote handling of activated components. The observation gallery, with radiation shielding windows on the north and east walls of the NB Test Cell, will permit visual observation for control of the crane and remote handling operations.

A removable hatch is provided in the NB Test Cell roof slab to allow the 45.4 tonne capacity crane in the Test Cell to be brought into the NB Test Cell, thus providing the capability for moving a 45.4 tonne beam line.

Access to the NB Test Cell for the rail system is accomplished by means of a 3.6 m wide x 8.5 m high x 0.6 m thick concrete sliding door between the NB Test Cell and the Transfer Bay. In addition, there are hatches in the floor to permit movement of smaller components into the NB Cell through the basement utility space, using a 22.7 tonne capacity crane in the NB Test Cell.

Personnel access is accomplished by means of shielded bulkheaded doors leading from the Transfer Bay.

The ventilation system is designed to function in two modes similar to the Test Cell system, so that the NB Test Cell must be constructed as a leak-tight chamber. Should there be operations involving tritium in the NB Test Cell, conditioned dehumidified air will be recirculated in a closed, sealed system with no venting to the outside. In this mode, the cell atmosphere is maintained at a pressure slightly below that of surrounding spaces in order to minimize out-leakage of tritium contaminated air. The entire volume of air in the NB Test Cell is recirculated through the tritium cleanup system as necessary to ensure that contamination levels do not exceed MPC.

During normal operational periods, the NB Test Cell is supplied with conditioned, dehumidified air as necessary to dissipate heat generated by the operations, and to maintain comfort conditions for personnel. In addition, the system will filter and clear the cell air in order to control dust and radioactivity levels. A portion of the cell air is exhausted to the outside after having been filtered through absolute filters, with the balance of the air being recirculated.

Hot Cell

The Hot Cell is a shielded area 13.1 m x 19.8 m x 9.1 m high for maintenance and repair operations on activated components from the TFTR device and the NB Test Cell. Construction is of reinforced concrete with 0.6 m thick floor, walls, and roof. Access for the rail system is provided by means of a 3.6 m wide x 8.5 m high sliding door constructed of 0.6 m thick reinforced concrete. The Hot Cell is equipped with a 22.7 tonne capacity overhead crane, an overhead manipulator, and master/slave type manipulators at several locations around the side walls. A two-level observation gallery with shielding windows on the south and east sides of the cell permits visual observation and control of the crane and remote handling systems when required for the Hot Cell operations.

The ventilation system is designed to provide conditioned dehumidified air to the Hot Cell, and to remove dust and radioactive particles from the cell atmosphere. During operational periods when maintenance work is being performed on activated components, the cell atmosphere will be maintained at a pressure slightly below normal atmospheric pressure in order to minimize out-leakage of radioactive or contaminated air. Cell air is recirculated in this mode, with fresh makeup air being introduced to offset the air discharged to the outside after having been passed through absolute filters to remove any radioactive dust.

In the event of abnormal tritium release within the Hot Cell, the system will function as a leak-tight sealed chamber with no outside air discharge and with pressure maintained at 2.54 cm H₂O below pressure in adjacent areas. In this mode, all air will be recirculated through the tritium cleanup system in order to remove and recover any airborne tritium.

Mock-up/Assembly Bay

The Mock-up/Assembly Bay provides space in which models and mock-ups of the TFTR device and its various components will be placed to study assembly, installation, and maintenance techniques. This bay will also provide space in which personnel may be trained to operate the device, auxiliary systems, and remote maintenance systems. The space may also be used for fabrication and maintenance of non-activated components during the course of the program. Since there are no shielding requirements for this space, the building may be constructed with a conventional industrial type steel structural system; concrete foundations with a reinforced concrete slab-on-grade floor; and insulated metal-faced wall and roof panels.

In order to facilitate movement of heavy components, the rail system is extended from the Test Cell through the Transfer Bays and into the Mock-up/Assembly Bay where a 45.4 tonne capacity bridge crane is provided. The space has dimensions of 29.9 m x 36.6 m with approximately 13.7 m clear to the underside of the roof structure.

Service access from the exterior is provided by means of an overhead motor-operated rolling steel door 6.1 m wide and 7.6 m high.

Components and assemblies will be transported from the Mock-up/Assembly Bay to other parts of the Experimental Area through an opening with an 0.6 m thick shielded concrete sliding door leading from the Mock-up/Assembly Bay to the Transfer Bay.

Ventilation is accomplished by means of a conventional forced ventilation system with no special requirements for temperature or humidity control or filtering.

Transfer Bays

The transfer bays provide for movement and transport of various equipment and components of the TFTR device within the Experimental Area for maintenance, repair, and assembly operations.

Construction is of reinforced concrete; floor, wall, and roof thicknesses have been determined by radiation shielding and structural support requirements. The two transfer bays are separated by a 1.2 m thick concrete wall, 6.1 m wide x 7.9 m high and a 1.2 m thick concrete sliding door which allows the rail system to pass from one transfer bay to the other.

Transfer Bay I is exposed to the Test Cell with high levels of radiation and, therefore, has shielding requirements similar to those of the Test Cell with a minimum of 1.2 m of concrete for the floor, 0.6 m on the wall adjacent to the Neutral Beam Test Cell, and 0.9 m of concrete, plus 4.6 m of earth berm for the outer wall on the west side. This Bay is ventilated by the Test Cell ventilation system and, therefore, has the same operational requirements and modes of operation.

Transfer Bay II has shielding requirements similar to those of the adjacent Hot Cell, with a minimum of 0.6 m of concrete for floors, walls, and roof. This Bay is ventilated by the Hot Cell ventilation system and will function in the same operating modes with a similar requirement for interconnection with the tritium cleanup system.

The primary transport system in the transfer bays consists of the rail system with 90.7 tonne capacity transport carriages. Provision has also been made for an alternative method of transporting small components and pieces of equipment between the Test Cell, the Neutral Beam Test Cell, the Hot Cell, and storage and work areas in the basement. Removable concrete hatches have been provided at various locations in the transfer bay floors, giving access to a rail transport system in the basement utility

area and providing a pathway for equipment movement which bypasses the main shielding doors and permits the integrity of the tritium sealed areas to be maintained.

Basement Areas

Space for a variety of functions has been provided at the basement level below the main experimental areas.

Basement space below the Test Cell and the NB Test Cell is assigned for diagnostics equipment, vacuum pumping equipment, electric bus runs for power supply to the device, coil cooling water piping for the device magnetic field coils, the tritium vault, tritium cleanup and air handling equipment, and liquid nitrogen equipment. The area beneath the Hot Cell and observation gallery is assigned for storage of spare parts for the device and its auxiliary systems and for storage of activated components which may require maintenance or repair work but for which the activity must decay before the work can be performed. The area below the transfer bays serves as an equipment handling and movement area, and also includes an electronic calibration workshop where repairs, maintenance, and adjustments can be made on small electrical components, controls, and monitoring equipment.

Experimental Area Ventilation System

Normal Operation

With the exception of the Mock-up/Assembly Bay, the entire TFTR Building is air conditioned. For normal ventilation during periods when the device is not in operation, the building ventilation system provides two air changes per hour in all spaces, recirculating approximately 80% of the total air volume and exhausting approximately 20% of the air to the outside after having been passed through absolute filters to remove radioactive dust. Exhaust air is constantly monitored to ensure that the concentrations of radioactive isotopes which cannot be filtered out are suitable for release. Fresh air is introduced to the system in required quantities, mixed with the recirculated air, and after cooling and dehumidification, is supplied to the various spaces through the building duct distribution system. Air flow is directed and controlled by means of dampers in the distribution system.

Equipment for the Experimental Area HVAC System is located in the mechanical equipment rooms in the adjacent Control and Mechanical Equipment Building. The system is operated from the Central Control Room in this later building.

Tritium Operation

Because of the danger of tritium contamination or contamination from radioactive dust in the system, provisions must be made in the system design to isolate and shield all mechanical HVAC equipment which might come into direct contact with contaminated

air. The required isolation is accomplished in the shielded mechanical equipment space located at the lowest level of the Control and Mechanical Equipment Building which will house air handling equipment for closed cycle operations of the Experimental Area ventilating systems.

The tritium envelope for the TFTR Building includes the main floor of the Experimental Area (except Mock-up and Assembly Bay) and all areas in the basement which could be exposed to tritium and contaminated as a result of tritium operations with the device.

Potential sources of tritium contamination include accidental release of tritium to the Test Cell atmosphere as a result of vacuum vessel failure, exposure to tritium-contaminated air circulated through the ventilation system and escape of tritium from tritium-contaminated components of the device or the auxiliary systems which may be transported within the Experimental Area for maintenance purposes.

The ventilation systems for all spaces within the tritium envelope are interconnected with the tritium cleanup equipment by means of dampers and bypass ducts, so that tritium-contaminated air may be circulated through the tritium cleanup and recovery equipment. Within the tritium envelope, all spaces are designed as sealed spaces capable of being operated at reduced pressures to prevent out-leakage of tritium-contaminated air. In addition, major spaces within the tritium envelope are to be separately zoned to reduce the total volume of air that will have to be processed by the tritium cleanup system within a given time period.

Control and Mechanical Equipment Building

The Control and Mechanical Equipment Building houses the central control functions for the entire TFTR complex and provides space for the computer, data acquisition and storage, office and service space for operating staff and experimentalists, and laboratory and support space for radiation monitoring and safety personnel.

The building also provides space for chilling equipment, pumps, fans, filters and air handling equipment for the TFTR complex cooling water and HVAC systems.

All functional spaces, including the computer areas, the main control console space, offices, laboratories, service and support spaces, are to be fully air conditioned, with normal ventilation to be provided in the mechanical spaces as required.

The Central Control Room accommodates thirty separate control consoles for use by experimental personnel. Eight offices are provided adjacent to the Central Control Room. These offices are unassigned and are intended to be used as general-purpose work space for TFTR staff and personnel.

The Control and Mechanical Equipment Building serves as the primary entrance to the Experimental Area for personnel and for shipping and receiving for the Experimental

Area. All personnel access to Experimental Area will be controlled at entrance points which are accessible only through the Control and Mechanical Equipment Building. Exits from the Experimental Area and basement areas will be provided with monitoring stations to monitor radiation contamination levels of personnel and radiation levels of equipment or materials that are removed from the areas.

The Control and Mechanical Equipment Building has overall dimensions of approximately 25.9 m x 38.1 m. The Central Control Room floor is at a level approximately 5.5 m above the main floor level of the Experimental Area and 2.4 m above existing grade. Mechanical equipment space, the receiving and service areas, and a radiological survey laboratory, are located at the same level as the main floor of the Experimental Area with the shielded mechanical equipment space and storage space at the lowest level adjacent to the Experimental Area basement.

The building has a reinforced concrete and partial steel frame, with flat slab floor construction to reduce vibration and sound transmission to the Central Control Room. The exterior is of pre-cast concrete and glass. The south wall being designed for radiation shielding and as a retaining wall for the adjacent earth berm.

Power Buildings

a. Motor Generator Building

The Motor Generator Building is an industrial type steel framed structure, 60.7 m x 33.5 m in plan with a 15.2 m high central bay and side bays 7.6 m high. The exterior walls are insulated panels and glass. The building provides space for three concrete-lined pits to receive motor generator sets with vertically mounted flywheels. Two of the motor generator sets with flywheels will be installed initially with the third planned for the future. The building also houses necessary switchgear equipment, an emergency diesel powered generator unit, a control room for local control and monitoring of the power equipment, and an open floor area for equipment assembly operations.

The control room is fully air-conditioned and located at a mezzanine floor level with a view of the motor generator sets and switchgear areas.

The 15.2 m high central bay is designed to support a 181.4 tonne capacity crane required for the installation of the motor generator sets and lifts of major components.

The building will be ventilated by means of roof-mounted exhaust fans with a capacity sufficient to provide two air changes per hour in all areas except the control room.

b. Field Coil Power Conversion Building

The Field Coil Power Conversion Building is a single-story industrial type steel framed structure 50.3 m x 30.5 m in plan with a clear interior height of 7.6 m. The building

is enclosed with insulated wall panels and glass and has a single overhead rolling steel door 6.1 m wide and 7.6 m high for equipment access at the north end of the building.

The building houses rectifiers and switchgear equipment for conversion of the electric power for the TFTR device magnetic field coils. The building has been planned to satisfy the requirement for close proximity of the rectifiers, which are located at the inside face of exterior walls, and the corresponding transformers which are mounted on concrete pads adjacent to the outside face of the exterior wall of the building. Power design considerations dictate a maximum distance of 7.6 m from the transformers to the rectifiers. Capacitor yards 24.4 m x 24.4 m with concrete pads for equipment-mounting are located adjacent to the north and south walls of the building. The building is ventilated by means of rooftop exhaust fans with capacity sufficient to provide two air changes per hour in all areas.

c. Neutral Beam Power Conversion Building

The Neutral Beam Power Conversion Building is a single-story steel framed industrial type structure 60.0 m x 36.9 m in plan and with a clear height of 8.5 m to the underside of the roof structure. The building is enclosed with insulated panels and glass. The building is located against the east wall of the Test Cell with the concrete slab-on-grade floor at the same level as the main floor of the Experimental Area.

Design and power supply requirements for the neutral beam system dictate that the modulator-regulator units in the Neutral Beam Power Conversion Building be located as close as possible to the beam lines in the test cells, and that the total length of cable connecting the modulator-regulator unit and the beam lines be no more than 15.2 m. This consideration is a primary determinate in the overall planning of the TFTR complex.

The balance of space in the building is occupied by power supply components and equipment, local system control areas and a small space for equipment maintenance and repair.

Access to the building for equipment is provided by an overhead rolling steel door 6.1 m wide x 7.6 m high, and personnel access is provided by means of doorways leading from the Experimental Area observation gallery with emergency exits directly to the exterior. Building equipment includes an 18.2 tonne crane on an 18.3 m bridge spanning the modulator-regulator area. The building is to be fully air-conditioned and the various equipment areas are to be isolated for high voltage operation and protected with fire walls as required for safe operation.

d. Utility Tunnel

The utility tunnel connecting the Control and Mechanical Equipment Building, the Field Coil Power Conversion Building and capacitor yards, and the basement beneath the Test

Cell, is constructed of reinforced concrete with adequate provision to ensure that the structure will be waterproof. The tunnel provides for distribution of the chilled water lines from the chilling equipment in the Control and Mechanical Equipment Building to the device, and for the electric bus run from the Field Coil Power Conversion Buildings to the device. The entire tunnel is to be air-conditioned and force-ventilated in order to prevent condensation or buildup of moisture. The inside dimensions of the tunnel are sufficient to allow maintenance personnel to inspect the lines and to perform any required maintenance. The tunnel is completely isolated during operations.

Engineering Lab Office Building

The Engineering Lab Office Building is planned to accommodate approximately 200 people and to provide space for a computer complex, engineering and physics laboratories, and related mechanical and service functions. The building has five floors including a basement, with a total gross floor area of 5600 m² and a net assignable area of 3440 m².

The structure is reinforced concrete with an exterior of architectural pre-cast concrete with glass. All spaces are to be fully air-conditioned with florescent lighting and acoustic ceilings throughout. Floors in offices are to be carpeted, with resilient tile in all other areas. Two elevators are required for vertical transportation of personnel, equipment, and supplies.

All systems are to be designed to achieve maximum efficiency in the utilization of energy consistent with occupancy requirements. The interiors will be planned to permit a high degree of flexibility in space layout so that the building will meet both present and future needs as effectively as possible. The building is located adjacent to existing Plasma Physics Laboratory office space to facilitate contact and communication between TFR staff and other laboratory personnel.

Auditorium

The auditorium building has capacity for 350 seats, a raised platform with speakers' podium, a projection room for movie and slide projection, lobby space, toilet rooms, and necessary mechanical and service areas. The auditorium is to be fully air-conditioned and acoustically treated to ensure proper acoustic performance. The building will have a steel framed structure with masonry enclosure walls and a sloped floor in the auditorium proper. Exterior design and materials are to be compatible with the location adjacent to the existing Plasma Physics Laboratory structures.

Cafeteria Expansion

The existing Plasma Physics Laboratory Cafeteria is to be expanded to double the present seating capacity. Construction and equipment are to be similar to those in the existing building and the space is to be planned so that it can be subdivided into small, private dining rooms suitable for use by small groups of people. The new space will be served from the existing serving lines and kitchen.

Technical Shops

A 420 m² high bay structure is to be added to the existing C-Site technical shops to provide space for the increased shop service needs of the TFTR Project. The building will have a structural steel frame, concrete slab-on-grade floor and exterior walls of insulated transite faced panels and glass. The structure will support a 4.5 tonne capacity crane on a bridge spanning 12.2 m. The space is to be heated and air-conditioned for personnel comfort. Since the building will be an extension of the existing technical shop, additional toilet facilities will not be required. The facility includes personnel access and exit doors and a loading dock with access to the crane bay by means of a 4.6 m wide by 6.1 m high overhead rolling steel door.

Maintenance Building

The Maintenance Building provides 930 m² of space to house new boilers, chillers, compressors, pumps, and control equipment for the HVAC Systems. Construction is to be steel frame with concrete slab-on-grade, masonry faced cavity walls, and pre-cast concrete plank roof with insulation and built-up roofing.

Utility Systems

The TFTR Project includes the construction and installation of all necessary utility and support systems as follows:

- a. Sanitary Sewer - The TFTR sanitary sewer system is to be connected to the existing C-Site system in the vicinity of the existing sewage treatment plant. A new pumping station installed on the sewer main will pump sewage a distance of approximately 975 m to a connection point with the existing Forrestal Center development sewer system and will then be discharged into the South Brunswick Township Municipal sewer system. Sewage is to be monitored at the pumping station to ensure that no radioactive wastes are discharged into the system. Utilization of the existing C-Site sewage treatment plant is to be determined.
- b. Storm Sewer System - Storm water is to be led by a combination of underground piping and swales to a new retention basin located adjacent to the cooling towers, and then discharged to Bee Brook.
- c. Power Supply and Distribution - Electric power will be supplied from the existing 138 kV service at the C-Site switchyard. Primary distribution of 13.8 kV power will be underground from the new TFTR substation and transformer to the Motor Generator Building. 4.16 kV electric distribution for building power, mechanical equipment, signal, control and lighting systems are to be distributed in underground duct banks.

- d. Potable Water - The potable water system will be distributed by a new loop serving the TFTR complex buildings within the exclusion area, and connected to the existing C-Site potable water system. The system will be supplied from a new 15.2 cm main connecting the C-Site system with the 40.6 cm Elizabethtown Water Company main located to the west of C-Site.

The connection between C-Site and the Elizabethtown main is not included within the scope of the TFTR Project.

- e. Process Water and Canal Water - Process water will be supplied from the existing water towers which are in turn supplied with water drawn from the Delaware-Raritan Canal. The canal water is presently filtered and treated at the existing canal pumping station before being pumped to the water tanks at C-Site. Process water for the TFTR Project will be supplied from the existing water towers and will include blow-down and cooling tower makeup water, cooling water for machine area cooling, water for cooling motor generator sets and electrical equipment, condenser water for all TFTR-related air-conditioning, and water for the fire main system. Distribution will be through underground lines to the various locations at pressures as required. It is estimated that total TFTR cooling water requirements will approximate 470 m³ per day.
- f. Cooling - The Machine Area Water Cooling System consists of a closed circuit demineralized water system with piping from the machine area to the mechanical equipment space in the Control and Mechanical Equipment Building running through the utility tunnel. Demineralizing equipment and pumps for the closed circuit system are located in the mechanical equipment space. Cooling tower water is pumped from the cooling towers through the new cooling tower pumping station to the mechanical equipment space where the connection between the two systems is effected by means of chillers located in the mechanical equipment space. The pumping station consists of a new pumphouse located near the cooling towers with a pumping capacity of 2.8 m³ per second. Chilling equipment and pumps for the air-conditioning condensate return system are located in the New Maintenance Building. Heat generated from the various sources will be dissipated to the atmosphere through the cooling towers. 44 MW of new cooling tower capacity is included in the TFTR Project.
- g. Heating - Heat for TFTR facilities will be provided from boilers located in the new Maintenance Building at C-Site. The project will include one 4.9 MW boiler, with provision for future expansion boiler capacity in the Maintenance Building.
- h. Compressed Air - Compressed air is to be distributed through an underground piping system from compressors located in the Maintenance Building

to the Control and Mechanical Equipment Building, the Experimental Area and all power buildings. Pressure reducing valves are provided in the buildings where required.

- i. Liquid Nitrogen System - Liquid nitrogen is required in the Experimental Area and the Neutral Beam Power Conversion Building. Equipment consisting of valves, and monitoring and control equipment is centrally located in the basement of the Experimental Area with piping runs to the various locations where the liquid nitrogen is required. The dewar is located outside the Experimental Area and will be recharged from trucks supplied from off-site sources.

Radioactive Waste Disposal System

The Radioactive Waste Disposal System has two major subsystems:

- a. Liquid waste disposal system consisting of storage tanks with sufficient capacity to contain the entire volume of water in the closed circuit Machine Area Water Cooling System plus the water in the Experimental Area fire protection sprinkler system and pumps, filters, and monitoring equipment. Liquid wastes are stored in tanks until the levels of radioactivity have been reduced to levels suitable for discharges. The liquid will then be disposed of to the storm sewer system or, if necessary, pumped to tank trucks for disposal off-site.
- b. Solid waste disposal system consisting of an isolated storage area for containers with activated debris, such as used absolute filters, equipment parts and used materials which must be trucked off-site for ultimate disposal. The system also includes the equipment for collecting the activated debris and packaging it in suitable containers for disposal.

Fire and Explosion Protection

The Fire and Explosion Protection system provides the capability to detect fire, to sense conditions which might constitute an explosion or fire hazard, to activate fire extinguishing and explosion containment systems and to sound alarms in appropriate monitoring offices.

The system will include rate-of-rise thermal detectors, products of combustion detectors, pressure and molar concentration monitoring equipment, automatic sprinklers and total flooding Halon systems as appropriate for the functions, occupancies and hazards of the various areas.

Because of the fire hazards posed by exposed high-voltage electrical equipment in the Experimental Area, Power Buildings, Utility Tunnel, and basement areas; and because of the necessity to maintain tritium seals for personnel and environmental safety, the detailed design of the Fire and Explosion Protection system must be very carefully coordinated with the design of the ventilation, cooling water, and tritium containment systems.

All TFTR complex facilities are to be provided with fire and explosion in accordance with ERDA and PPPL requirements.

Site Improvements

Site improvements include provision of parking space at C-Site for TFTR personnel, construction of the security fencing surrounding the exclusion area with the gates and guard booth at the exclusion area entrance, construction of walks, road, drainage structures such as the retention basin for storm drainage, exterior site lighting, grading, planting and landscaping for the entire project area at C-Site.

5.0 PROGRAM PLAN

5.1 ORGANIZATION AND KEY PERSONNEL

The Division of Controlled Thermonuclear Research (DCTR), with responsibility for development of the Fusion Power Program, has jointly, with Princeton, established project objectives, and through its Technical Projects Office, will monitor overall project progress, provide technical inputs to the ERDA Princeton Area Office and perform other programmatic functions. The Technical Projects Office (TPO), established within DCTR, has the responsibility for assuring that ERDA Headquarters activities are accomplished in support of TFTR requirements. The TFTR Conceptual Design, as described in this report, meets the objectives jointly established by PPPL and ERDA-DCTR.

The Princeton Area Office (PAO) has the delegated authority (from the ERDA Chicago Operations Office) to administer the prime contract with Princeton University (ERDA Contract E(11-1)-3073) under which PPPL will accomplish the TFTR Project. ERDA-PAO will also administer the prime contracts for Architect/Engineer services and for construction of the TFTR buildings and facilities. The Neutral Beam Lines for TFTR will be developed through ERDA-DCTR at ORNL and at LBS/LLL in accordance with requirements established by PPPL.

PPPL has direct responsibility as prime contractor for the design, development, fabrication, installation, check-out and operation of the tokamak experimental device and associated buildings, facilities and systems. The resources of PPPL, under the direction of Professor Melvin B. Gottlieb, will provide an integrated technological base for the TFTR Project. The theoretical and experimental studies of controlled thermonuclear fusion over the past 23 years have produced a team of physicists, engineers, and support personnel at PPPL with broad experience in the design and operation of fusion research machines of the tokamak type. Figure 5.1-1 shows the organization of PPPL which provides a broad technology base for effective support of multi-project research, while also maintaining specialized core skills in each functional area. Professor Gottlieb has established a TFTR Project within the PPPL organization to perform this important PPPL assignment.

Paul J. Reardon has been appointed TFTR Project Manager at PPPL. He will report on the administrative activities of TFTR with several ERDA offices as appropriate, and establish a project organization to effectively manage the project. Figure 5.1-2 shows the project organization Mr. Reardon is setting up to perform administratively, scientifically and technically, the work of the project, including the coordination of the activities of all the project participants.

Figure 5.1-1. Princeton Plasma Physics Laboratory Organization

Figure 5.1-2. TFTR Project Organization

A TFTR Headquarters has been established to provide centralized management control for the project. Both technical and administrative control will be maintained within the headquarters office. To accomplish the goals of the project, Professor Harold P. Furth will be responsible for physics and Mr. Kenneth E. Wakefield will be responsible for engineering. In addition to these major assignments, Mr. Reardon intends to appoint an Administrative Section Leader and an Operations Section Leader. Mr. Reardon will manage the Administrative Section in addition to his duties as TFTR Project Manager until a Section Head is appointed.

The Physics Section will continually evaluate and interact with developing TFTR designs to provide assurance that the resulting design represents the best operational choice within project constraints, i.e., technical performance, schedules and budgets. Operating results from PLT, PDX and other PPPL projects, as well as other controlled fusion projects throughout the United States and the World, will be factored into the design (when pertinent) as the project evolves.

The Engineering Section will have direct responsibility for the technical design of the device, energy storage and distribution, power conversion and all other systems required for the TFTR Project. These responsibilities will be performed both directly with PPPL personnel and through the organization of the Industrial Subcontractor selected by PPPL. Engineering liaison will be maintained with the National Laboratories charged with the development of the Neutral Beam Injector System and with the Architect/Engineering firm selected by ERDA to design the buildings and facilities.

An Operations Section will be established during the engineering phase. It will be responsible for check-out and operation of the various subsystems, and for selecting and training the crew of physicists, engineers and technicians required for operating the facility, including the related Remote Handling, Hot Cell, the Neutral Beam Test Cell and the several Safety and Control Systems. Detailed requirements for this section and initial Operational Procedures will be prepared during the preliminary design phase of the project.

The Industrial Subcontractor (IS) will perform a major portion of the work required to complete the TFTR and will provide a rapid expansion of the technical and management resources available to the project as the need develops. This will permit the project to progress at maximum speed without overextending the permanent manpower base of PPPL.

5.2 WORK BREAKDOWN STRUCTURE

The Work Breakdown Structure (WBS) provides a basis for dividing the project into related work packages for Project Planning and Control. It establishes a number system which is used for reporting progress. Major TFTR systems are broken into subsystems and are then broken into logical work packages which represent, in general, the work flow from preliminary design (Title 1) through installation, check-out and operation of the facility. Further breakdown of the WBS work packages can be accommodated where justified by the size, complexity, or other factors.

The TFTR Project has been broken into twelve major systems for project planning and control purposes as follows:

1. Project Management

- 1.A Project Administration
- 1.B Project Planning and Control
- 1.C Technical Coordination
- 1.D Quality Assurance
- 1.E Procurement
- 1.F Data Management
- 1.G Computer Terminals

2. Safety and System Engineering

- 2.A System Engineering
- 2.B System Analysis
- 2.C Plasma Physics and Neutronics Analysis
- 2.D Operations and Test Analysis Planning
- 2.E Configuration Control System
- 2.F Environmental and Safety Engineering

3. Tokamak Systems

- 3.A Tokamak System Engineering
- 3.B Vacuum Vessel Assembly
- 3.C Structure and Shielding
- 3.D TF Coil Assembly
- 3.E EF and OH Coil Assembly
- 3.F Auxiliary Coils

4. Electrical Power Systems

- 4.A Electrical Systems Engineering
- 4.B Energy Supply and Distribution
- 4.C OH Pulsed Energy Conversion
- 4.D TF Pulsed Energy Conversion
- 4.E EF Pulsed Energy Conversion
- 4.F Auxiliary Field Pulsed Energy Conversion

5. Experimental Area Systems

- 5.A Experimental Area Systems Engineering
- 5.B Tokamak Assembly
- 5.C Remote Handling and Maintenance Systems
- 5.D Tritium System
- 5.E Tritium Clean-Up Systems

- 5.F Non-Tritium Charge Gas Delivery
- 5.G Machine Area Water Cooling Systems
- 5.H Torus Vacuum Pumping System
- 5.J Machine Area Electrical Buswork
- 6. Architect/Engineering Services
 - 6.A A/E Management and Systems
 - 6.B Site Development
 - 6.C Buildings
 - 6.D Special Systems
 - 6.E Utilities
 - 6.F Equipment
- 7. Facility Construction
 - 7.A Construction Management
 - 7.B Site Development
 - 7.C Buildings
 - 7.D Special Systems
 - 7.E Utilities
 - 7.F Equipment
- 8. Neutral Beam Systems
 - 8.A Neutral Beam Injector
 - 8.B Neutral Beam Pulsed Energy Conversion
- 9. Diagnostic Systems
 - 9.A Diagnostic Systems Engineering
 - 9.B Basic Diagnostic Apparatus
 - 9.C Scientific Diagnostic Apparatus
- 10. Central Instrumentation, Control and Data Acquisition
 - 10.A System Engineering Design
 - 10.B Facility Measurement and Control
 - 10.C Experimental Measurement and Control
- 11. Associated Research and Development
 - 11.A Research and Development Coordination
 - 11.B R&D Support for the Tokamak System
 - 11.C R&D Support for the Electrical Power Systems
 - 11.D R&D Support for the Experimental Area Systems
 - 11.E R&D Support for the Neutral Beam Systems
 - 11.F R&D Support for the Central Control System

12. Facility Operation

12.A Start-Up and Pre-Operations

12.B Operations and Maintenance

5.3 PROJECT COST ESTIMATE

The cost estimate for the TFTR Conceptual Design was prepared by a "bottoms-up" item item-by-item estimate of the buildings and equipment costs, including assembly and installation. The depth of detail used in preparation of the costs provides a reasonably high degree of confidence that the project can be completed within the PACE budget. Detailed cost estimates for the buildings and equipment items were prepared for each subsystem and summed by systems according to the Work Breakdown Structure identified in Section 5.2. The responsibilities for preparing the various subsystem cost estimates were divided among Westinghouse, Princeton and others as indicated below:

	Organizations Providing Input for Cost Estimate
● Tokamak System, including assembly and installation	<u>W</u>
● Electrical Power and Control Systems including assembly and installation	<u>W</u>
● Building and Facilities Construction	<u>W</u> , ERDA, PPPL, Wood and Tower
● Neutral Beam Systems, including assembly and installation	<u>W</u> , PPPL, LLL/LBL ORNL
● Diagnostic Systems, including assembly and installation	<u>W</u> , PPPL
● Supporting Research and Development	<u>W</u> , PPPL

Westinghouse has prepared engineering specifications and has solicited budgetary cost estimates for 58 bid packages from qualified vendors for major equipment items in the Tokamak System, Electrical Power Systems, and the Experimental Area Systems. Engineering personnel evaluated the vendor quotations and selected the cost estimates that appeared most responsive to the technical specification. Engineering cost estimates were prepared by Westinghouse for those hardware items for which there were no bid responses or where there were no qualified vendors (only 15 percent of the hardware items). The assembly and installation costs were estimated

by Westinghouse on the basis of craft-labor rates as provided by Wood & Tower, Inc. for the Princeton, N. J. area. All cost estimates were prepared to a common set of guidelines. For example: all estimates were made at October, 1975 dollar value; escalation and contingency factors were excluded; estimates were separated into PACE (Plant and Capital Equipment) and operating (R&D) funds.

Princeton (PPPL) retained the services of Collins, Uhl, Hoisington & Anderson (Architects, Engineers and Planners) and Wood & Tower, Inc. (Construction Management Specialists) to assist the TFTR Project personnel in developing the final conceptual designs and cost estimates for the buildings and facilities. Inputs on structural requirements were received from the Loftus Engineering Company and on soil excavation studies from Carlin-Ferretti and Associates. Earlier conceptual studies and cost estimates prepared by Westinghouse and Loftus Engineering were used as a baseline reference design, and numerous iterations were made to improve the functional value and cost effectiveness of the buildings and facilities. Final conceptual design drawings for the buildings and facilities included in Section 4.7 were used by Wood and Tower to prepare detailed cost estimates using established construction costing procedures.

A Neutral Beam Steering Committee was formed in the Spring of 1975 to develop the criteria and procedures to be followed in defining the conceptual design of the Neutral Beam Injection lines for TFTR. Separate studies and cost estimates were prepared by Oak Ridge National Laboratories (ORNL) and a team composed of personnel from the Lawrence Berkeley and Lawrence Livermore Laboratories. Their cost estimates were based on a common set of guidelines from the Work Breakdown Structure established for the TFTR Project. The Neutral Beam Power Supply estimate was done by PPPL, using inputs from the participating laboratories and industry.

Diagnostic System cost estimates were prepared by physicists and engineers at Princeton, based largely on experience developed with diagnostics for PLT and other fusion research projects at Princeton. Mechanical design studies were made to assure that the diagnostics conceptual design configuration were compatible with the Tokamak Experimental Area including remote handling interface requirements. Some of the TFTR diagnostics equipment are a direct outgrowth of prototype diagnostics which are presently being built for PLT.

Total PACE costs for the TFTR Project are summarized in Table 5.3-1. Buildings and equipment costs are all backed up by detailed cost estimates described above. Table 5.3-2 provides a summary of buildings and equipment costs at the subsystem level. The EDIA (Engineering, Design, Inspection and Administration), contingency and escalation allowances were made by Princeton alone.

The contingency allowance is believed to be adequate based on the level of design definition at the end of the Conceptual Design study and with the understanding that the supporting research and development programs will be funded at PPPL and the national

laboratories (for neutral beam systems) in a timely manner, consistent with the needs of the PACE schedule. The contingency funds will not be distributed to the various subsystems, but will be maintained under the direct control of the TFTR Project Manager at PPPL.

The escalation allowance was obtained by applying an 8.5 percent yearly factor to the funds on an obligation schedule (\$15.0 million in FY76, \$5.5 million in FY76A, \$80 million in FY77, \$94.4 million in FY78, and \$33.1 million in FY79), which is consistent with the planned funding schedule and program requirements. Any delays in the program due to funding, authorization, selection of contractors or other programmatic conditions would require reevaluation of escalation allowance.

5.4 PROJECT SCHEDULE

The TFTR schedule has been prepared using critical path networks. The schedule indicates that project completion will occur in June, 1981, predicated on contract placement with the Industrial Subcontractor by Princeton, and the Architect/Engineer by ERDA in the first quarter of calendar year 1976. The TFTR Level 1 Schedule Logic Diagram, Figure 5.4-1, shows an elapsed time of 62 months and is consistent with previously reported schedules.

5.4.1 SCHEDULE ASSUMPTIONS

In general, conservative estimates, consistent with present conceptual design maturity, were chosen to assure a high probability of schedule success. It should be noted, however, that other uncertainties, such as vendor shop load at time of order placement, the future availability of raw materials, and the timely success of supporting research and development programs may cause subsequent schedule adjustments.

The TFTR Level I Master Schedule shows Title I tasks starting on April 1, 1976, and it was assumed that all project participants would be funded and fully staffed for both PACE and R&D tasks at that time. PPPL received authorization to proceed with the TFTR Construction Contract (PACE) on February 24, 1976, and both the A/E and the IS are under letter contract to initiate their respective tasks. PPPL is proceeding on critical path items such as the Motor-Generator Set and long-lead materials for the tokamak and expects to initiate Titles I, II and IV on all items as required to meet the project schedules.

TABLE 5.3-1 TFTR PACE ESTIMATE

Activity	(Dollars in Thousands)
	Buildings & Equipment
Tokamak Systems	\$ 17,123
Electrical Power Systems	37,725
Experimental Area Systems	12,300
Installation	8,831
Buildings and Facilities	28,500
Neutral Beam System	20,048
Diagnostic System	1,223
Central Instrumentation and Control	4,450
Total Buildings and Equipment (1)	129,750
EDIA Allowance at 23% (2) (3)	30,300
Contingency Allowance	30,750
Escalation Allowance	<u>37,200</u>
TOTAL PACE ESTIMATE	\$228,000
<p>(1) Equipment portion is based largely on 85% detailed bid estimates provided by vendors. Construction cost for buildings and utilities were prepared by Wood & Tower, Inc., using architects' drawings. Assembly and Installation is based on detailed engineering estimates of man hours required and prevailing craft labor rates.</p> <p>(2) EDIA, contingency and escalation allowance made by Princeton alone.</p> <p>(3) Includes \$400K of construction planning and design funds expended in FY75 and FY76.</p>	

TABLE 5.3-2 TFTR BUILDINGS AND EQUIPMENT COST SUMMARY

Task	(Dollars in Thousands)		
	Subsystem	System	Total
TFTR Buildings and Equipment			129750.0
TOKAMAK SYSTEM		17123.0	
Vacuum Vessel Assembly	2895.0		
Structure and Shielding	5505.0		
TF Coil Assembly	6436.0		
EF and OH Coil Assembly	2287.0		
ELECTRICAL POWER SYSTEMS		37725.2	
Energy Supply and Distribution	15234.8		
OH Pulsed Energy Conversion	6140.2		
TF Pulsed Energy Conversion	9800.0		
EF Pulsed Energy Conversion	6550.2		
EXPERIMENTAL AREA SYSTEMS		12300.1	
Tokamak Assembly	1844.1		
Remote Handling and Maintenance Systems	5658.9		
Tritium System	184.2		
Tritium Clean-Up Systems	1239.0		
Non-Tritium Charge Gas Delivery	90.0		
Machine Area Water Cooling System	2177.9		
Torus Vacuum Pumping System	1106.0		
INSTALLATION (WESTINGHOUSE)		8380.3	
Installation Management and Site Indirect Cost	352.0		
Equipment Installation Craft Services	8028.0		
FACILITY CONSTRUCTION		28500.0	
Site Development	1096.4		
Buildings	21836.1		
Utilities	5367.0		
Equipment	200.5		
NEUTRAL BEAM SYSTEMS		20058.4	
Neutral Beam Injector	9148.4		
N.B. Pulsed Energy Conversion	10900.0		

TABLE 5.3-2. TFTR BUILDINGS AND EQUIPMENT COST SUMMARY (Continued)

Task	(Dollars in Thousands)		
	Subsystem	System	Total
DIAGNOSTIC SYSTEMS Basic Diagnostic Apparatus	1223.0	1223.0	
CENTRAL INSTRUMENTATION AND CONTROL SYSTEMS Facility Measurement and Control	4450.0	4450.0	

Specifically, it was assumed:

- That adequate obligational authority will be obtained on a timely basis to permit financial commitments to be made as scheduled (i.e., \$15.0 million in FY76, \$5.5 million in FY76A, \$80 million in FY77, \$94.4 million in FY78 and \$33.1 million in FY79).
- That adequate expenditure authority will be available to permit progress milestone and final delivery payments to subcontractors as they come due to avoid delays in production.
- That mobilization of manpower assigned to this project by the A/E and Industrial Subcontractor will be rapid, permitting essentially immediate initiation of preliminary design activities upon contractor selection.
- The Research and Development tasks in support of preliminary and final design activities will be adequately identified, approved, and funded for timely initiation.

5.4.2 CRITICAL PATH ASSESSMENT

Critical paths on the Level I logic network are identified in Figure 5.4-1 by heavy horizontal lines tracing the activities associated with the Vacuum Vessel, Toroidal Field Coils, Neutral Beam Injection System, and the Motor Generator Set.

System schedules as well as detailed Level II Network Plans have been prepared for the Tokamak System, the Electrical Power System, the Experimental Area Systems, the Neutral Beam Injection Systems, and the Diagnostic System. These schedules will be expanded and updated as the project progresses.

Figure 5.4-1. TFTR Level I Master Schedule

5.4.3 PROJECT MILESTONES

Table 5.4.1 lists the TFTR Project Milestones as they appear on the TFTR Level I Schedule Logic Diagram.

5.5 SAFETY AND ENVIRONMENTAL IMPACT

Early evaluations reported in the TCT Joint Conceptual Design Study considered the potential impact of the TFTR on the safety of the general public and on the environment with the facilities at an unspecified location. Major hazards to the operating personnel were also identified to permit incorporation of suitable safeguards into the conceptual design. The impact of the facility on the immediate area was considered minimal due to the proposed design, the planned mode of operation and the safeguard to be implemented. Subsequently, the Princeton Forrestal Campus was selected as the site for the TFTR.

5.5.1 SAFETY

The safe design and operation of the TFTR will be regulated in accordance with the provisions of the ERDA (formerly AEC) Manual Chapter 0500 series and the generally applicable environmental radiation guidelines of the EPA (formerly vested in the Federal Radiation Council). In particular, the applicable routine and accidental occupational and public exposure standards and guidelines are specified in ERDA Manual 0524. These guidelines, published in the Code of Federal Regulations, are consistent with the Nuclear Regulatory Commission (NRC) radiation standards and guidelines which apply to privately owned radiation facilities.

Approval has been received from ERDA (CH-Safety and Headquarters Operational Safety) on the format and scope of the Safety Analysis Reports that will be required to support a request for the initiations of construction and operations of the facilities, although specific ERDA Manual Chapter of the 0500 series does not exist for Safety Analysis Reports for fusion devices (as one does for fission devices). Many of the 0500 series chapters are directly applicable. ERDA has directed that a Preliminary Safety Analysis Report (PSAR) will be prepared during Title I Engineering approved by ERDA before significant construction is undertaken. Likewise, a Final Safety Analysis Report (FSAR) will be required to support ERDA-CH final review and safety assessment of the facility before the start of operations.

Consideration was given to following a format equivalent to that established in ERDAM-0540 for SAR's for nuclear power plants. However, early evaluations indicated that the potential hazards to the general public associated with the operation of the TFTR are considerably less than those for a nuclear fission power plant, and the systems used are different in nature and generally less complex. In addition, ERDA's interest and responsibility for the full safety of the facilities and the operating personnel of their contractors requires consideration of many aspects of equipment and operations

TABLE 5.4-1 TFTR PROJECT MILESTONES

Milestone Number	Description	Date
1	Complete Conceptual Design	3/76
2	Start Title I Preliminary Design for major components, buildings, and facilities	3/76
3	Start A/E Title I	3/76
4	Order Long Lead Material (Toroidal Field Coils)	2/77*
5	Start A/E Title II	2/77
6	Start A/E Title III	4/77
7	Order Motor-Generator-Flywheel System	4/77*
8	Order Long Lead Material (Vacuum Vessel)	5/77*
9	Complete Facility Design Review	7/77
10	Start Site Preparation	7/77
11	Complete Tokamak System Final Design Review	8/77
12	Start Facility Building Construction	10/77
13	Conduct Remote Maintenance Design Review	2/78
14	Release Fabrication of Vacuum Vessel	5/78
15	Release Fabrication of Toroidal Field Coils	10/78
16	Complete MG Building Construction	11/78
17	Release Pulsed Energy Conversion Fabrication	1/79
18	Start Installation of MG Set	10/79
19	Complete Construction of Test Cell	1/80
20	Complete TFTR Complex Construction	1/80
21	Start Final Assembly of Tokamak	5/80
22	Complete Vacuum Vessel Testing	12/80
23	Start Integrated Electrical Power System Testing	1/81
24	Complete Final Assembly of Tokamak	2/81
25	Complete Integrated Tokamak System Check-out and Test	6/81
* PPPL is planning to initiate orders for long lead materials by June, 1976.		

which exceed those generally treated under the scope of ERDAM-0540. Consequently, the existing guide for the format and content of Safety Analysis Reports for nuclear power plants was not considered adequate or directly applicable.

A draft outline and preparation guide for the Safety Analysis Reports for TFTR has been prepared which incorporates those elements normally found in such reports for facilities handling radioactive materials, and an evaluation, with supporting data, of the safety of operating personnel. This outline has received ERDA's approval, with some slight modification, and it will be followed for the preparation of the PSAR and subsequently for the FSAR.

The proposed contents of the various chapters of the Safety Analysis Reports are as follows:

The first chapter will contain a summary description of the TFTR device and its supporting facilities; the role of TFTR in the development of fusion power capabilities; and a summary of the conclusions of the report.

Chapter 2 will describe the site characteristics using information that was reported in the Environmental Statement and other information as required. The topics covered will include geography, demography, nearby land use, meteorology, hydrology, geology and seismology.

The layout of the TFTR site and a description of the facilities and buildings will be included in Chapter 3.

Chapter 4-7 will describe and evaluate the systems and major components. These chapters cover the TFTR Equipment (Chapter 4), Plant and Auxiliary Systems (Chapter 5), Instrumentation and Control (Chapter 6) and Electrical Power (Chapter 7). The discussions will include as applicable:

- a) Functional requirements for the systems and components with emphasis on those features which have safety implications.
- b) Design bases followed in the design and analysis of these systems and components to meet the functional requirements.
- c) Description of the systems and components emphasizing those features which verify conformance with the requirements and design bases, and appropriate schematics and drawings of equipment.
- d) Safety evaluation of the design, indicating the potential failure modes that have safety implication and the safeguards included in the design to protect against such failures or to reduce the consequences of such failures to an acceptable level.
- e) Test and inspections performed prior to initial operation and the requirement for continued surveillance and maintenance.

The systems and procedures for the collections, processing, disposal and monitoring of the effluent of liquid, gaseous and solid waste material will be described in Chapter 8. Chemical and radioactive waste will be considered.

Chapter 9 will discuss the operational health and safety program and will summarize, and as required describe and evaluate, the safeguards provided for the protection of operating personnel. It is intended that all safeguards be considered in this chapter, but reference to earlier descriptions and safety evaluations contained in the report will be made where appropriate.

The management controls over the conduct of the operations will be described in Chapter 10, including the organizational structure and authorities of the operations group, training, emergency planning, procedures and industrial security.

Chapter 11 will summarize the tests to be performed during the construction phase leading to acceptance of the facility for planned experimental operations, the initial experimental program and those planned maintenance and support operations which may be potentially hazardous or of particular significance to safety.

Chapter 12 will present the quantitative safety analysis of the failures identified in earlier chapters. Where appropriate, the consequences of the failures on the public and operating personnel will be determined and reported.

Chapter 13 will include significant operating limits, conditions, and requirements imposed upon TFTR operation in the interest of health and safety of the public and operating personnel.

Chapter 14 will describe the Quality Assurance programs applicable to design, construction and operation of the facilities.

5.5.2 ENVIRONMENTAL IMPACT

During the latter part of 1974, detailed evaluations of the environmental impact of the TFTR on the Princeton area were undertaken by the project, and subsequently a Draft Environmental Statement (DES) was issued by ERDA in January, 1975. Review and comment by interested Federal, State and local government agencies was solicited and responses were received through May, 1975.

The additional data and considerations proposed in these responses were incorporated into the Final Environmental Statement (ERDA-1544) which was issued by ERDA in July, 1975.

As a general summary, the comments received on the DES covered all aspects of the report and were considered pertinent and penetrating, but not adversarial in

nature. Many reviewers requested final design detail which could not be accommodated, but sufficient information was made available to adequately respond to all comments. As a matter of comparison, the DES's for commercial facilities which are associated with the nuclear fission fuel cycle, and with which people are most familiar, are normally prepared to support the issuance of a Construction Permit by the Nuclear Regulatory Commission. These are prepared at a stage when the preliminary design of the facility has been completed. For the TFTR, however, the Statement was prepared to support ERDA's proposal for legislative authorization and appropriation for the project.

SUMMARY OF FINAL ENVIRONMENTAL IMPACT STATEMENT

The location for the TFTR is in central New Jersey, about 3 miles to the east of the main campus of Princeton University. The closest urban centers are New Brunswick, 14 miles to the northeast and Trenton, about 12 miles to the southwest.

The Princeton area has become a focal point of research and development facilities and light industrial development, but the majority of the land in the three county region in the vicinity of the site remains agricultural. However, these counties are expected to undergo a marked change from present rural land use to suburban-oriented land use over the next 5 to 10 years.

The site is not located in an earthquake-prone area and no geologic deposits of economic value are presently known to exist at the site. The climate of the site is largely continental, and winds are not often strong enough to be damaging except due to occasional tropical storms. The site is located 40 feet above the highest recorded flood for surface waters in the vicinity.

There were no rare plant species observed during an on-site survey, and the only endangered mammal, bird, or reptile species in the area occupy the lowland and swamp forests. There are no National Parks, Landmarks, State Forests, Wilderness Areas, or Wetlands in the immediate area, and the local landmarks within the vicinity will not be adversely affected by the TFTR program.

Construction of the TFTR complex will have little impact on the general public, since the closest residence is approximately 0.4 miles from the site. Approximately 4 acres will be disturbed by clearing and excavation, and the habitats of a few ground dwelling mammals may be displaced. Surface waters will be affected by increased runoff and deposited silt, but the impact will be mitigated by the construction of a berm to catch the drainage from the construction site. Nevertheless, the increased siltation may cause a disruption of Bee Brook's benthic macroinvertebrate population. Depending upon atmospheric conditions, truck and equipment exhaust fumes and dust from traffic may be detectable offsite.

During the TFTR experimental program, plasma containing deuterium and tritium generated within the toroidal vacuum vessel of the TFTR will react and release energetic neutrons and gamma rays, some of which will penetrate beyond the confines of the building. Additionally, traces of the tritium within the vacuum vessel will be released to the test cell by permeation through the vessel wall and will be released to the environment, together with relatively short-lived radionuclides created in the test cell atmosphere by neutron activation.

The releases of radionuclids to the environment during normal operations of the TFTR will provide small radiation exposures to biota and the general public in the vicinity. However, the maximum individual dose of 5.9 mrem/yr to an individual at the exclusion area boundary and 0.3 mrem/yr to an individual occupying the closest off-site residence are considered to be a small fraction of natural background and are well within applicable standards. Within a 50-mile radius of the site, the total population exposure is 4 man-rem. This is insignificant in comparison with the 1.4×10^6 man-rem to the general public in the same area from natural background radiation.

The TFTR will consume electrical power, most of which will be dissipated as heat to the atmosphere from induced-draft evaporative cooling towers. Makeup water for the cooling towers and other requirements for non-potable on-site water will be drawn from the Delaware and Raritan Canal, which is supplied by the Delaware River. The operation of the evaporative cooling towers will release to the atmosphere visible vapor plumes and dissolved solids entrained in the drift. However, no significant ground level fogging is anticipated during normal operating hours of the TFTR, and concentrations of deposited solids in the surrounding woods are expected to be below the levels for which sensitive plant species would be affected. Chemicals used for treatment of the circulating water and solids dissolved in the water supply and concentrated by evaporation will be discharged to the watershed in the blowdown, but these materials will be significantly diluted by other wastewater and will be well within water quality and effluent standards before entering surface waters.

Considerable growth is anticipated in the area surrounding the TFTR, consequently the 220 additional operational employees will have a negligible incremental socio-economic impact on local communities. Moreover there should be no significant congestion in local traffic. Water consumption will not affect the water supply, which is fed by the Delaware River. Nor, for that matter, will the consumption of electric power significantly affect the capacity of the local electrical supply.

The construction and operation of the TFTR facility is not expected to conflict with Federal, State, or local land use policies or plans.

The proposed design of the TFTR has been reviewed for potential abnormal operation conditions which could have off-site consequences. To the extent possible, the likelihood of accidents will be mitigated by safeguards systems and administrative controls.

The structures will be designed to withstand all credible natural phenomena. The most severe accident postulated, failure of the vacuum vessel, would involve the release of approximately 65 Curies of tritium, and would result in a maximum dose to an individual at the exclusion area boundary of 8 mrem. This dose would be about 14 percent of the annual dose from natural background.

Among the alternatives to the implementation of the TFTR program described in the Environmental Statement, the termination and the postponement of the program were considered. These were determined to be unsatisfactory because they would compromise the goal of achieving useful power from fusion in the United States on a timely basis. Additional controls for reducing radiological effluents from the facility would either entail significant reduction in the operating flexibility of the facility or involve large incremental expenses for additional equipment or services. These additional controls are not justified in view of the very low predicted doses resulting from the reference facility. Once-through cooling and cooling ponds are not feasible alternative cooling systems on the basis of water and land availability, respectively. Non-evaporative cooling does not offer environmental advantages commensurate with its higher economic costs. Chemical effluents in liquids will meet applicable Federal, State, and local water quality and effluent standards, and waste treatment will be at the state-of-the-art commensurate with minimal environmental impact when the TFTR program is initiated.

Alternative site locations, including the construction and operation of the TFTR at a National Laboratory or at an isolated site in the Princeton area, were examined. These options could involve important economic and operational costs, while yielding only marginal reductions in the environmental impacts.

The facility can be decommissioned at the end of the TFTR program if alternative uses of the buildings and equipment cannot be justified. Facility equipment would be dismantled and packaged for off-site disposal including decontamination if necessary. The entire area occupied by the facilities could be returned to essentially its original state. The short-term use of the site will not jeopardize the health and well-being of the public or the environment. The most significant short-term use of the environment will be the commitment of a certain amount of land, materials, manpower, and capital resources. The principal long-term potential benefit is the development of fusion power as a viable energy source, for which the TFTR is an important step in achieving this goal.

In assessing and balancing the economic and environmental costs against the benefits, and after considering the range of available alternatives and their environmental impacts, it was concluded that the design, construction, and operation of the TFTR as conceptually described in the Environmental Statement should be accomplished as proposed.

6.0 SUPPORTING PROGRAMS

6.1 INTRODUCTION

Operating costs will be incurred over the period FY76 through the first 9 months of FY81 as summarized below:

Research and Development Activity	\$ 35.2M
Experimental Research	20.8
Facility Operation	31.1
Spare Parts	<u>14.0</u>
Total	\$101.1

The operating costs are in February, 1976 dollars, with contingency, but without escalation. By June, 1981, it is planned that the TFTR group will be on the operation budget.

The Research and Development Activity Plans (RDACs) describe the specific areas where the PACE design must be preceded by technical information which is not yet available. The RDAC effort includes component development, production of prototypes, studies of alternative designs of components and studies of proposed diagnostic equipment. Neutral Beam development is not included, though the Neutral Beam Power Supply prototype is, since the latter is the responsibility of PPPL.

The Experimental Research effort includes preparation for physics testing and staff build-up, experiment planning and development of data analysis methods.

The Facility Operations effort includes preparation for operating the facility and hardware, as distinguished from conducting the physics test. Included are salaries of personnel hired in advance of operations, formal training of personnel, operational system acceptance tests, environmental monitoring and surveillance during construction and maintenance of equipment.

The Spare Parts Inventory Listing for the TFTR is based on operational requirements to minimize down-time.

\$9.7M of non-PACE equipment is included in the above budgets.

6.2 RESEARCH AND DEVELOPMENT ACTIVITIES

Research and Development requirements were developed during the Conceptual Design Study to support specific hardware design, analysis and fabrication decisions for TFTR. Development plans were prepared for each Research and Development Activity (RDAC) to show the technical requirements, scope of work, end products and documentation. Also included in the RDAC plans are cost and schedule data. All RDACs are identified by numbers to locate them in the Work Breakdown Structure, and each is cross referenced with the WBS number for the appropriate element of the TFTR requiring the RDAC results.

The RDAC cost is estimated at \$35.2M and is broken down among the TFTR systems as follows:

Diagnostics	\$13.8
R&D Coordination	3.9
Tokamak	4.4
Electrical	2.3
Experimental Area	1.9
Neutral Beams	6.7
Instrumentation & Control	0.4
	<hr/>
	\$33.4M
Contingency	1.8M
	<hr/>
TOTAL RDAC COST	\$35.2M

Neutral Beam RDAC costs include the Neutral Beam Power Conversion developmental activities and the Princeton liaison for the Neutral Beam Injector development tasks. It is expected that separate funding will be made available to the National Laboratories for the Neutral Beam Injector development.

The planned time phasing of the \$35.2M is as follows:

FY76	\$ 0.4M
76A	1.3
77	10.4
78	12.4
79	4.3
80	3.7
3/4 81	2.7
	<hr/>
TOTAL	\$35.2M

The RDAC schedule and expenditure rate is in general agreement with the TFTR schedule, with priority being given to tokamak related research in the first and second years; electrical and experimental area systems in the second and third years; and diagnostic development, simulation and remote handling interface development in the third and fourth years. Since the RDAC work must precede final design of equipment, it is most important that the RDAC funding and scheduling be maintained. Revisions in specific RDAC plans are expected to occur as the design proceeds, but the overall level of effort is not expected to change significantly from the breakdown and schedule given herein. A further breakdown of the RDAC budget is given in the February, 1976 revision of the "Supporting Programs" report.

The TFTR Project Manager is responsible for overall management of the Research and Development Activities. He will utilize the TFTR Physics and Engineering Sections to provide technical direction and supervision for the individual RDACs.

The Industrial Subcontractor may also provide direction and supervision in specific cases.

Each RDAC will be performed by the organization which is most qualified to do the particular tasks. In many cases, the RDACs will be performed by PPPL or the Industrial Subcontractor. However, other laboratories or commercial organizations will be utilized when their special talents or other programmatic conditions justify their usage.

6.3 EXPERIMENTAL RESEARCH

The Experimental Research Plan describes the effort required by PPPL physicists and support staff to prepare for physics testing, experiment planning and data analysis development for the TFTR research program. The plan, which is in Section 3 of the Supporting Programs, defines six main categories:

- a) plasma physics considerations
- b) electrical and magnetic field analysis and nuclear radiation analysis
- c) data acquisition studies
- d) experimental tests and facilities
- e) experiment planning
- f) coordination with PLT & PDX

The plasma physics considerations will extend and modify those initiated for the Conceptual Design. This task includes the study of problems such as neutral beam

injection into the plasma, losses due to toroidal field ripple, gas injection studies, consideration of the vacuum vessel armor, plasma scenarios, plasma breakdown and plasma termination, and discharge cleaning and bakeout requirements.

The field and radiation analyses will include studies of forces on magnetic field coils, examination of components to prevent induced currents and stray magnetic fields, circuit analysis for proposed plasma scenarios, optimization of shielding with respect to diagnostic requirements and activation studies.

Since it is planned that essentially all diagnostic information will be processed through a computer before presentation to the experimental physicists, there must be a very tight coupling between the physicists and the design group responsible for the TFTR computers and instrumentation interfaces. In the Experimental Research effort, it follows that there will be a considerable effort devoted to planning, designing and procuring a compatible hardware and software system for all diagnostic instruments. This effort will require roughly one third of the total Experimental Research budget.

The experimental tests and facilities category is required since many diagnostic concepts will have to be evaluated. Obvious items for Experimental Research are limiters and neutral beam calorimeters, gas injection techniques and various prototype diagnostic components. Some use will be made of existing PPPL tokamaks, such as PLT, by testing prototype diagnostic components in the PLT research program.

During the latter part of the Experimental Research program, the physics group will plan the various aspects of the operational research program. An appreciable amount of training of new physicists and technicians will occur during this time.

Summary of Manpower and Budget

	<u>FY77</u>	<u>FY78</u>	<u>FY79</u>	<u>FY80</u>	9 Mo. <u>FY81</u>
Physicists	2.5	10	13	20	22
Supporting Staff	1	21	29	45	39
Budget M\$	0.3	2.2	4	7.3	7.0

TOTAL \$20.8M

6.4 FACILITY OPERATION

Significant planning for Facility Operation will start in FY77.

The first phase of planning will include formulation of plans for electrical and radiation safety, handling of radioactive materials, start-up and shut-down procedures, interlocks,

site design from an operational point of view, and environmental monitoring. The first elements of the Instrumentation and Control System will become available during this time and will be used for commissioning the prototype Neutral Beam Line.

The second phase of planning will start roughly in FY78. Various major tasks will be undertaken as well as continuing studies of the subjects listed above. Procedures will be established to assemble, operate, maintain and disassemble the tokamak. Test specifications and operations manuals will be produced. Equipment layouts will be reviewed, and the prototype control system will be operated and expanded as more equipment is delivered during FY79.

The third phase, starting in FY79, will consist mainly of operator training, testing remote control procedures, documenting test results, checking safety and interlock systems and continuing the testing and modifications of equipment as it is installed. This facility is scheduled to be operational by June 1981.

A summary of operations personnel needed during FY77 to the first 9 month of FY81 period is given below:

	<u>FY77</u>	<u>FY78</u>	<u>FY79</u>	<u>FY80</u>	<u>FY81</u>	<u>(9 Mo.)</u>
Staff	2.5	13.0	27.0	46.0	34.0	
Support	<u>0.5</u>	<u>14.0</u>	<u>32.0</u>	<u>82.0</u>	<u>66.0</u>	
Total Manpower	3.0	27.0	59.0	138.0	100.0	
Budget (\$K)	330	2,200	4,750	12,300	11,550	
				TOTAL	\$31,130	

The \$31.1M budget estimate for Facility Operations includes roughly a 10% contingency. It is in February, 1976 dollars with no escalation included. Details of the planned Facility Operations are contained in the February, 1976 Supporting Programs document issued by PPPL.

6.5 SPARE PARTS

The spare parts requirement for the TFTR Project was prepared with the objective of minimizing downtime resulting from equipment failure or outage in any of the components or auxiliary systems. The spare parts list was prepared from recommendations by Westinghouse, the various vendors' estimates for equipment, and from operating experience at PPPL on existing equipment.

The cost of the initial inventory of spare parts for the project is estimated at \$14M without escalation. The cost breakdown by major systems is as follows:

Tokamak	\$ 3.9M
Electrical Power	5.5
Experimental Area	.4
Neutral Beams	3.2
Buildings and Facilities	<u>1.0</u>
Total Spares	\$14.0M

An annual replacement or upgrading of the inventory is estimated at \$1.1M.

A Motor-Generator-Flywheel with auxiliaries is the major element of cost in the electrical power system spares.

Neutral Beam Injector and High Voltage Pulsed Energy Conversion spares have been estimated at 15% of the equipment cost, since they both require a large amount of development in the next few years, and consequently their reliability is largely unknown.

The Spare Parts List for the TFTR Buildings and Facilities includes identified items in each of the following areas:

- a) General Building
- b) HVAC and Plumbing Systems
- c) Building and Electrical Systems
- d) Tritium Cleanup and Containment Systems
- e) Cranes and Remote Handling Systems
- f) Radioactive Waste Disposal System
- g) Fire Control and Protection

Summary of Operating Programs

The operating budget for TFTR is estimated at \$77.4M, plus \$9.7M non-PACE equipment, plus \$14M for spares. The breakdown of expenses is given for the first two categories by fiscal year below:

COSTS PER YEAR IN K\$

FY	R&D	Experimental Research	Facility Operations
1976	\$ 365	\$ --	\$ --
1976A	1,270	--	--
1977	9,500	250	250
1978	11,743	1,700	1,950
1979	3,100	2,800	4,500
1980	2,500	5,580	12,000
1981 (3/4)	<u>2,300</u>	<u>6,300</u>	<u>11,355</u>
TOTAL	\$30,778	\$16,630	\$30,050

EQUIPMENT COSTS IN THOUSANDS OF DOLLARS

FY	R&D	Experimental Research	Facility Operations
1976	30	--	--
1976A	30	--	--
1977	910	60	80
1978	650	500	250
1979	1,200	1,200	250
1980	1,200	1,700	300
1981 (3/4)	<u>400</u>	<u>700</u>	<u>200</u>
TOTALS (K\$)	4,420	4,160	1,080

Operations Budget	\$ 77.4M
Equipment Budget	9.7M
Spare Parts Budget	<u>14.0M</u>
TOTAL	\$101.1M