APPENDIX A

REVIEW TEAM MEMBERS, AND MEETINGS

A.1 ITER Review Panel Members

Prof. Robert W. Conn (Chairman)*
Dean and Walter J. Zable Professor, School of Engineering
University of California, San Diego

Dr. David Baldwin
Senior Vice President
General Atomics

Dr. Richard J. Briggs*
Senior Program Manager
SAIC

Prof. James D. Callen*
Kerst Professor of Nuclear Engineering & Engineering Physics and Physics
University of Wisconsin

Prof. Robert Goldston
Associate Director for Research of PPPL
    and Professor of Astro-Physical Sciences, Princeton University
Princeton Plasma Physics Laboratory

Prof. Richard D. Hazeltine *
Director of the Institute for Fusion Studies, and Professor of Physics
University of Texas, Austin

Dr. Richard E. Siemon
Fusion Energy Program Manager
Los Alamos National Laboratory

Dr. Tony S. Taylor *
Senior Technical Advisor and Group Leader, Stability
General Atomics

*Also FESAC member

A.2 ITER Review Sub-Panel Members

Sub-Panel I. Physics Performance, Projections, Experimental & Theoretical Basis, Global Scaling

Dr. Tony S. Taylor (Co chairman)*
Senior Technical Advisor and Group leader, stability
General Atomics

Dr. William Tang (Co chairman)
Principal Research Physicist, head, Theory Division
Princeton Plasma Physics Laboratory

Prof. Glen Bateman
Professor of Physics
Lehigh University

Dr. Keith Burrell
Senior Technical Advisor
General Atomics

Dr. Vincent Chan
Director, Core Physics
General Atomics

Prof. Lui Chen
Professor of Physics and Astronomy
University of California, Irvine

Dr. Steven Cowley
Professor of Physics and Astronomy
University of California, Los Angeles

Dr. Patrick Diamond
Professor of Physics
University of California, San Diego

Dr. William Dorland
Research Associate, Institute for Fusion Studies
University of Texas, Austin

Dr. James Drake
Professor of Physics, Laboratory for Plasma Research
University of Maryland

Dr. Raymond J. Fonck
Professor of Nuclear Engineering and Engineering Physics
University of Wisconsin

Dr. Martin J. Greenwald
Principal Research Scientist and Group Leader, Plasma Fusion Science Center
Massachusetts Institute of Technology

Dr. Gregory W. Hammett
Research Staff Physicist
Princeton Plasma Physics Laboratory

Prof. Richard D. Hazeltine *
Director of the Institute for Fusion Studies, and Professor of Physics
University of Texas, Austin

Dr. Wayne A. Houlberg
Research Staff
Oak Ridge National Laboratory

Dr. Stanley M. Kaye
Principal Research Physicist
Princeton Plasma Physics Laboratory

Dr. Michael Kotschenreuther
Research Scientist, Institute for Fusion Studies
University of Texas

Dr. Joseph A. Johnson, III *
Professor of Physics
Florida A&M University

Dr. John D. Lindl *
ICF Scientific Director
Lawrence Livermore National Laboratory

Dr. Kevin M. McGuire
Principal Research Physicist, Deputy Head, TFTR
Princeton Plasma Physics Laboratory

Dr. Janardhan Manickam
Principal Research Physicist
Princeton Plasma Physics Laboratory

Dr. Stewart C. Prager
Professor of Physics
University of Wisconsin

Dr. Mickey Wade
Research Staff
Oak Ridge National Laboratory

Dr. Ronald Waltz
Senior Technical Advisor and Group Leader: Theory
General Atomics

Dr. Steven M. Wolfe
Principal Research Scientist and Group Leader, Plasma Science Fusion Center
Massachusetts Institute of Technology

Dr. Michael Zarnstorff
Principal Research Physicist
Princeton Plasma Physics Laboratory

<u>Sub-Panel II. Divertor Concept, Integrated Fuel Cycle</u>

Dr. Earl S. Marmar (Chairman)*
Senior Research Scientist, Plasma Fusion Center
Massachusetts Institute of Technology

Prof. Ira B. Bernstein*
Professor of Mechanical Engineering and Physics
Yale University

Dr. Bastiaan J. Braams
Research Associate Professor, Courant Institute of Mathematics
New York University

Dr. Katherine B. Gebbie *
Director of Physics Laboratory
NIST

Dr. John R. Haines
R&D Program Manager, Fusion Energy Division
Oak Ridge National Laboratory

Dr. Dave Hill
Lawrence Livermore National Laboratory

Dr. Charles Karney
Principal Research Physicist
Princeton Plasma Physics Laboratory

Dr. Bruce Lipschultz
Principal Research Scientist and Group Leader, Plasma Science Fusion Center
Massachusetts Institute of Technology

Dr. Stanley Luckhardt
Research Scientist
University of California, San Diego

Dr. Peter Mioduszewski
Confinement Section Head, Fusion Energy Division
Oak Ridge National Laboratory

Dr. Gary Porter
Group Leader, DIII-D Modeling and Data Acquisition
Lawrence Livermore National Laboratory

Dr. Charles Skinner
Princeton Plasma Physics Laboratory

<u>Sub-Panel III. Disruptions, VDE's, Blanket/Shield Attachment</u>

Dr. Stewart J. Zweben (Chairman) *
Principal Research Physicist
Princeton Plasma Physics Laboratory

Prof. Hans Fleischmann
Professor of Applied and Engineering Physics
Cornell Univiversity

Dr. Eric Fredrickson
Principal Research Physicist
Princeton Plasma Physics Laboratory

Dr. Robert S. Granetz
Principal Research Scientist, Plasma Science Fusion Center
Massachusetts Institute of Technology

Dr. Arnold Kellman
Manager, Tokamak Operations
General Atomics

Dr. George Sheffield
Engineer
Princeton Plasma Physics Laboratory

Sub-Panel IV. Advanced Modes/Flexibility

Dr. Farrokh Najmabadi (Chairman)
Associate Professor
University of California, San Diego

Dr. Steven Allen
Program leader for DIII-D collaboration
General Atomics

Dr. Nathaniel Fisch
Principal Research Physicist, Associate Director for Academic Affairs
Princeton Plasma Physics Laboratory

Dr. Richard Freeman
Director, RF Physics and Technology
General Atomics

Prof. Michaele E. Mauel
Professor of Applied Physics
Columbia University

Dr. Dale Meade
Principal Research Physicist, Deputy Director
Princeton Plasma Physics Laboratory

Dr. Miklos Porkolab
Director of the Plasma Science Fusion Center
Massachusetts Institute of Technology

Sub-Panel V. Achieve Availability Goals, Achieve Safety Assurance Goals, PFC Tritium Inventory

Dr. James Luxon (Chairman)
Director, DIII-D Operations
General Atomics

Dr. Richard Callis
Manager, RF Systems
General Atomics

Dr. Tom Casper
Goup Leader, Computer Systems
Lawrence Livermore National Laboratory

Ms. Melissa Cray*
Program manager for Inertial Confinement Fusion
Los Alamos National Laboratory

Dr. John DeLooper
Associate Director for ES&H/QA
Princeton Plasma Physics Laboratory

Dr. Richard Hawryluk
Principal Research Physicist, Project Head TFTR
Princeton Plasma Physics Laboratory

Dr. James Irby
Research scientist and group leader, Plasma Science Fusion Center
Massachusetts Institute of Technology

Dr. David Johnson
Principal Research Physicist
Princeton Plasma Physics Laboratory

Dr. Arnold Kellman
Manager, Tokamak Operations
General Atomics

Dr. Michael Williams
Head Engineering and Technology Department
Princeton Plasma Physics Laboratory

<u>Sub-Panel VI. Magnet Performance</u>

Dr. V. Karpenko (Chairman)
Retired, Lawrence Livermore National Laboratory

Dr. Richard J. Briggs*
Senior Program Manager
SAIC

Mr. Charles Bushnell
Retired, Princeton Plasma Physics Laboratory

Dr. Karl Krause
Division Leader of the Engineering Research Division
Lawrence Livermore National Laboratory

Mr. Tom Peterson
Cryogenic Engineer
Fermi National Accelerator Laboratory

Dr. Clyde Taylor
Senior Scientist
Lawrence Berkeley National Laboratory

Sub-Panel VII. Neutron Irradiation Effects, In-vessel Components

Dr. Samuel D. Harkness (Chairman)*
Director R&D Operations
Westinghouse Science and Technology Operations

Dr. Tom J. Mcmanamy
Senior Engineer
Oak Ridge National Laboratory

Dr. Tom Shannon
Director Maintenance and Reliability Center
University of Tennessee

Dr. Dale Smith
Director of the Fusion Power Program
Argonne National Laboratory

Prof. Don Steiner
Institute Professor of Nuclear Engineering and Chairman of the Dept. of Environmental and Energy Engineering
Rensselaer Polytechnic Institute

Sub-Panel VIII. Cost Methodology and Schedule

Dr. Michael Knotek (Chairman)*
Dept. of Physics
Washington State University

Dr. Richard Callis
Manager, RF Systems
General Atomics

Dr. John R. Haines
R&D Program Manager, Fusion Energy Division
Oak Ridge National Laboratory

Dr. V. Karpenko
Retired, Lawrence Livermore National Laboratory

Dr. John A. Schmidt
Interim Director
Princeton Plasma Physics Laboratory

Dr. L. Edward Temple
Project Director, Advanced Photon Source
Argonne National Laboratory

Dr. Michael Saltmarsh
Fusion Energy Division Director
Oak Ridge National Laboratory

<u>Sub-Panel IX. Facilities</u>

Mr. John Davis (Chairman)*
Manager of High Energy Systems
McDonnell Douglas Aerospace

Dr. G. Hutch Neilson, Jr.
Principal Research Physicist, Acting Head of Advanced Projects Department.
Princeton Plasma Physics Laboratory

Dr. Michael Saltmarsh
Fusion Energy Division Director
Oak Ridge National Laboratory

*Also FESAC member

A.3 List of Panel Meetings

ITER Review Panel Meetings:

Following the FESAC meeting the ITER Review Panel met January 23-24, 1997 at General Atomics, San Diego CA to discuss initial reactions to the ITER reports and presentations and to establish the process to complete the review and prepare a report for FESAC's consideration. It was ensured that each of the key issues, in regard to the questions in the Letter of Charge was assigned for review Present: R. Conn, R. Hazeltine, I. Bernstein, J. Davis, D. Baldwin, J. Callen, M. Cray, J. Lindl, S. R. Goldston, S. Zweben, V. Karpenko, S. Harkness, T. Taylor, M. Knotek, R. Siemon, R. Briggs, J. Luxon, F. Najmabadi, W. Tang, E. Marmar. Also present to discuss the process were J. Sheffield, M. Rosenbluth, N. Uckan, B. Montgomery, and N. Sauthoff.

The ITER Review Panel met March 20-21, 1997 at UCSD, San Diego CA to coordinate the final stages of report preparation.

Sub-Panel Meetings:

Sub Panel I (Physics Basis) met Jan. 9 - 10, 1997 in San Diego. Attendees included sub-panel members: T. Taylor, J. Lindl, G. Bateman, G. Hammett, W. Houlberg, M. Kotzenreuther, R. Waltz and M. Zarnstorff; FESAC members: N. Sauthoff, N. Uckan, J. D. Callen and M. Rosenbluth and other attendees: D. Boucher (ITER/presenter), R. Perkins (ITER), J. Wesley (ITER), V. Mukhovatov (ITER), Y. Putvinski (ITER), and J. Kinsey (presenter).

Sub-Panel III (Disruptions/VDE's Blanket/Shield Attachment) held several meetings with various sub groups:

At PPPL on 12/3/96: Brad Nelson, Jim Bialek, Peter Bonanos, Phil Heitzenroder, Neil Pomphrey, Alan Reiman, Hutch Neilson, Eric Fredrickson, Bob Granetz, George Sheffield, Stewart Zweben

At PPPL on 12/5/96: Mike Ulrickson, Eric Fredrickson, George Sheffield, Stewart Zweben

At PPPL on 12/12/96: Rich Mattis, Hutch Neilson, Eric Fredrickson, George Sheffield, Phil Heitzenroder, Stewart Zweben

At PPPL on 1/6/96: Doug Loesser, George Sheffield, Stewart Zweben, Eric Fredrickson, and the PPPL ITER engineering term
At the San Diego ITER JCT office on 1/23/97: Ron Parker met with Stewart Zweben

Sub-Panel VI. (Magnet Performance) held several meetings with various sub groups:

At GA in San Diego on Jan 22: V. Karpenko, B. Montgomery, C. Taylor.
At LBL on Jan 16, 1997: V. Karpenko and Clyde Taylor
At LBL on Feb 10, 1997: V. Karpenko and Clyde Taylor
At LBL on Feb. 24-25, 1997: V. Karkenko and C. Bushnell

Sub Panel VIII (Cost and Schedule) met in Salt Lake City for three days (March 5-7, 1997) to discuss the general methodology used, and to look in detail at the development of the C/S for the magnets. In attendence from the sub-panels were M. L. Knotek (chair), John Schmidt, Ed Temple, Bob Simmons, Mike Saltmarsh, Victor Karpenko, John Haines, Rich Callis. In attandence from ITER-JCT: Bob Iotti (Raytheon), Tom James (UCSD), Paul Gregory (Raytheon), Joel Kirschner (Raytheon), Forest Kimball (Lockheed Martin), and from DOE: Stan Staten.

The other sub-panels communicated primarily via conference calls, and email transmissions.

Appendix A.4. FESAC Jan. 23-24 Meeting Agenda

The FESAC met on January 20-21, 1997 at General Atomics in San Diego to hear presentations on the status and prospects for the project from key ITER personnel. The agenda, and a list of public commenters at this meeting are listed below.

Agenda

General Overview of ITER	R. Aymar
Highlights of Past TAC Reviews	P. Rutherford
Contributions to EDA Activities	C. Baker
Physics Performance Overview	N. Sauthoff
D-III-D Tour	
Reception, GA cafeteria	
Divertor Physics Overview	R. Stambaugh
Magnet Systems Overview	M. Huguet
In-Vessel Systems Overview	R. Parker
Safety Overview	D. Petti
Cost and Schedule Overview	R. Iotti
Public Comment	
Adjourn	

People Giving Public Comment:

Dr. P. Politzer, General Atomic
Dr. W. Ellis, from Raytheon and Chairman of the US ITER Industrial Council
Dr. S. Dean from Fusion Power Associates
Prof. W. Stacey, Georgia Institute of Technology
Mr. C. DeVaney, Executive Vice President, Augusta Tomorrow Inc
Dr. T. Simonen, General Atomics
Prof. Mohammed Abdou, UCLA
Prof. R. Goldston PPPL
Dr. J. Gilleland, Chief Scientist, Bechtel Co.

APPENDIX B.

SWG-1 REPORT.

The ITER Special Working Group #1 Report is as follows:

“Preamble

- In accordance with Article 10 of the ITER EDA Agreement,
- with reference to Sections 1 and 2 of Protocol 1,
- in the light of the Guidelines for SWG-1 imposed by the Ist ITER Council Meeting (Attachment 1),
- on the basis of the ITER Conceptual Design Activities Final Report, ITER Documentation Series No. 16, and the documents referred to therein,

the Special Working Group has agreed as follows.

1. General Constraints

The ITER detailed technical objectives and the technical approaches, including appropriate margins, should be compatible with the aim of maintaining the cost of the device within the limits comparable to those indicated in the final report of the ITER CDA as well as keeping its impact in the long-range fusion programme.

ITER should be designed to operate safely and to demonstrate the safety and environmental potential of fusion power.

2. Performance and Testing

- *Plasma Performance*

ITER should have a confinement capability to reach controlled ignition. The estimates of confinement capability of ITER should be based, as in the CDA procedure, on established favorable modes of operation.

ITER should

- demonstrate controlled ignition and extended burn for a duration sufficient to achieve stationary conditions on all time scales characteristic of plasma processes and plasma wall interactions, and sufficient for achieving stationary conditions for nuclear testing of blanket components. This can be fulfilled by pulses with flat top duration in the range of 1000s. For testing particular blanket designs, pulses of approximately 2000s are desirable.

• aim at demonstrating steady state operation using non-inductive current drive in reactor relevant plasmas.

- *Engineering Performance and Testing*

ITER should

• demonstrate the availability of technologies essential for a fusion reactor (such as superconducting magnets and remote maintenance);

• test components for a reactor (such as exhaust power and particles from the plasma);

• test design concepts of tritium breeding blankets relevant to a reactor. The tests forseen on modules include the demonstration of a breeding capability that would lead to tritium self-sufficiency in a reactor, the extraction of high-grade heat, and electricity generation.

3. Design Requirements

The choice of parameters of the basic device should be consistent with margins that give confidence in achieving the required plasma and engineering performance. The design should be sufficiently flexible to provide access for the introduction of advanced features and new capabilities, and to allow for optimising plasma performance during operation. The design should be confirmed by the scientific and technological database available at the end of the EDA.

An inductive pulse flat-top capability, under ignited conditions, of approximately 1000s should be provided. In view of the ultimate goal of steady state operation, ITER should be designed to be compatible with non-inductive current drive, and the heating system required for ignition in the first phase should have current drive capability.

To carry out nuclear and high heat flux component testing at conditions relevant to a fusion power reactor:

• the average neutron wall loading should be about 1 MW/m^2.

• the machine should be designed to be capable of at least 1 MWa/m^2 to carry out longer-time integral and materials tests.

It is desirable to operate at higher flux and fluence levels. Within the engineering margins, the ITER designers should examine the implications and possibilities of exploiting a wider range of operational regimes. The design of the permanent components of the machine should not preclude achieving fluence levels up to 3 MWa/m^2. For the second phase of operation, the design should include the capability of replacing the shield with a breeding blanket.

4. Operation Requirements

The ITER operation should be divided into two phases:

ï The first phase, the Basic Performance Phase, is expected to last a decade including a few thousand hours of full DT operation. This phase should address the issues of controlled ignition, extended burn, steady state operation, and the testing of blanket modules. It is assumed that for this phase there will be an adequate supply of tritium from external sources.

  - Controlled ignition experiments in ITER will address confinement, stability and impurity control in alpha particle heated plasmas. Extended burn experiments will address, in addition, the control of fusion power production and plasma profiles, and the exhaust of helium ash.

  - The aim of current drive experiments in this phase should be the demonstration of steady state operation in plasmas having alpha particle heating power at least comparable to the externally applied power. Using the heating systems in their current drive mode, non-inductive current drive should be implemented for profile and burn control, for achieving modes of improved confinement, and for assessing the conditions and power requirements for the above type of steady state operation. Depending on the outcome of these experiments, additional current drive power may have to be installed.

  - Functional tests of blanket modules in this phase should consist of a few thousand hours of integral burn time, in parallel with the physics programme, including continuous test campaigns of 3-6 days at neutron wall loading of about 1 MW/m^2.

ï The second phase, Enhanced Performance Phase, is also expected to last a decade, with emphasis placed on improving overall performance and carrying out a higher fluence component and materials testing programme. This phase should address high availability operation and advanced modes of plasma operation, and may address reactor-relevant blanket segment demonstration. Operation during this phase should include continuous testing campaigns lasting 1-2 weeks, and should accumulate a fluence of at least 1 MWa/m^2.

A decision on incorporating breeding for this phase should be decided on the basis of the availability of tritium from external sources, the results of breeder blanket testing, and experience with plasma and machine performance.

The implementation of the Enhanced Performance Phase should be made following a review of the results of the Basic Performance Phase and an assessment of the relative value of the proposed elements of the programme.

5. Final Recommendation

The ability to achieve the above objectives and to comply with the "Guideline for SWG-1" provided by the ITER Council at its first meeting should be confirmed by the Director in the outline of the design referred to in that Guideline.”

Appendix C.

Charge Letters for the FESAC Review of ITER

This Appendix contains copies of the charge letters from the director of Energy Research to the chair of FESAC regarding the review of the ITER DDR.

Department of Energy
Washington, DC 20585

September 23, 1996

Dr. John Sheffield
Chair, Fusion Energy Sciences
Advisory Committee
Oak Ridge National Laboratory
Bethel Valley Road
Oak Ridge, Tennessee 37831

Dear Dr. Sheffield:

In its report to the Department of Energy on January 27, 1996, the Fusion Energy Advisory Committee recommended that there should be broad U.S. participation in the review of the technical results of the International Thermonuclear Experimental Reactor (ITER) Engineering Design Activities (EDA).

The decision-making process for future ITER activities involves several steps. The U.S. Government has entered into a noncommittal, prenegotiation Explorations process with our ITER partners to explore possible construction arrangements. If there is a successful outcome to the Explorations process, if the technical status of the ITER EDA is judged to be of high enough quality, and if the financial situation warrants, the U.S. Government could then agree to enter into a process with the other interested Parties to negotiate the terms and conditions for an agreement for the construction, operations, exploitation and decommissioning of ITER. Once such negotiations would have been completed, the Parties would then decide whether to sign the agreement.

Technical information will also have to be developed to support the decisions to enter into negotiations and to sign the agreement. This information is being developed by the ITER Director in the form of a Detailed Design Report (DDR), scheduled to be available in mid-December 1996, and a Final Design Report (FDR) scheduled to be available in mid-December 1997.

The Fusion Energy Sciences Advisory Committee should provide its view of the adequacy of the DDR as part of the basis for a United States decision to enter negotiations. Please provide your report on this charge by May 1, 1997.

I would also like the Committee to provide to me its view of the technical adequacy of the FDR as part of the basis for a United States decision to sign a negotiated agreement. The exact date by which this charge must be completed is not yet firm. I will inform you of that date as soon as possible.

Enclosed is a series of questions that the Committee should address for both the DDR and the FDR.

Sincerely,

Martha Krebs

Martha A. Krebs
Director
Office of Energy Research

Enclosure

Questions to be addressed by the Fusion Energy Sciences Advisory Committee are listed below. As appropriate for each question, consider both the existing knowledge base as well as results expected from R&D and experimental physics programs in the near future.

Are the ITER physics basis and engineering design sound?

Is ITER likely to meet its performance objectives as agreed upon by the four Parties and documented in the 1992 SWG-1 report?

Are the scientific and technological risks associated with the ITER design acceptable at this stage in the project?

Do the design and operating plans adequately address environment, safety, and health concerns?

Are the proposed cost estimates and schedules for the construction project and subsequent operations, exploitation and decommissioning credible, and are they consistent with the procurement methods and staffing arrangements recommended by the ITER Director?

Are there any cost effective opportunities for pursuing modest extensions of the current design features in order to enhance operational flexibility and increase the scientific and technological productivity of ITER?

Department of Energy
Washington, DC 20585
November 6, 1996

Dr. John Sheffield
Chair, Fusion Energy Sciences
Advisory Committee
Oak Ridge National Laboratory
Bethel Valley Road
Oak Ridge, Tennessee 37831

John
Dear ~~Dr. Sheffield~~:

The charge letter to the Fusion Energy Sciences Advisory Committee of September 23, 1996, on the review of the International Thermonuclear Experimental Reactor Detailed Design Report included six questions. I understand that in making your detailed preparations for the review, you have suggested combining the first original question on physics and engineering with the third original question on technology into a new first question and adding points of clarification to all of the questions. The revised questions, including points of clarification, are shown in the enclosure and are acceptable. The report submission date of May 1, 1997, remains unchanged.

I look forward to receiving your report. Please let me know if additional clarification is needed.

Sincerely,

Martha

Martha A. Krebs
Director
Office of Energy Research

Enclosure

Revised Questions to FESAC on ITER DDR Review
Including Points of Clarification

Are the ITER physics basis, technology base, and engineering design sound? Focus on the critical physics, technology, and engineering issues that affect the design while allowing for the R&D planned in each of these areas through the end of the EDA.

Is ITER likely to meet its performance objectives as agreed upon by the four Parties and documented in the 1992 SWG-1 report? Evaluate predicted performance margins, comment on the range of operating scenarios, and identify opportunities to improve the performance projections.

Do the design and operating plans adequately address environment, safety, and health concerns? Focus on the methodology used by the Joint Central Team to address the concerns.

Are the proposed cost estimates and schedules for the construction project and subsequent operations, exploitation and decommissioning credible, and are they consistent with the procurement methods and staffing arrangements recommended by the ITER Director? Focus on the methodology used to prepare the estimates.

Are there any cost effective opportunities for pursuing modest extensions of the current design features in order to enhance operational flexibility and increase scientific and technological productivity of ITER? Focus on areas where cost effectiveness of any design extensions would be high.

APPENDIX D.I

SUB-PANEL I: PHYSICS BASIS REPORT

Dr. Tony S. Taylor (Co chairman) *
General Atomics

Dr. William Tang (Co chairman)
Princeton Plasma Physics Laboratory

Prof. Glen Bateman
Lehigh University

Dr. Keith Burrell
General Atomics

Dr. Vincent Chan
General Atomics

Prof. Lui Chen
University of California, Irvine

Dr. Steven Cowley
University of California, Los Angeles

Dr. Patrick Diamond
University of California, San Diego

Dr. William Dorland
University of Texas_Austin

Dr. James Drake
University of Maryland,

Dr. Raymond J. Fonck
University of Wisconsin

Dr. Martin J. Greenwald
Massachusetts Institute of Technology

Dr. Gregory W. Hammett
Princeton Plasma Physics Laboratory

Prof. R. Hazeltine
University of Texas, Austiin

Dr. Wayne A. Houlberg
Oak Ridge National Laboratory

Dr. Joseph A. Johnson, III *
Florida A&M University

Dr. Stanley M. Kaye
Princeton Plasma Physics Laboratory

Dr. Michael Kotschenreuther
Institute for Fusion Studies

Dr. John D. Lindl *
Lawrence Livermore National Laboratory

Dr. Janardhan Manickam
Princeton Plasma Physics Laboratory

Dr. Kevin M. McGuire
Princeton Plasma Physics Laboratory

Dr. Stewart C. Prager *
University of Wisconsin

Dr. Mickey Wade
Oak Ridge National Laboratory

Dr. Ronald Waltz
General Atomics

Dr. Steven M. Wolfe
Massachusetts Institute of Technology

Dr. Michael Zarnstorff
Princeton Plasma Physics Laboratory

*- FESAC Member

The Physics Basis Report below was prepared by a 26 member panel representing a broad spectrum of the community. Each of the sections below was written and received serious consideration by only a subset of the subpanel: individual findings, conclusions, and recommendations may not be unanimously agreed upon by the full panel. This report was accepted by the full subpanel.

The report is organized as follows: Section I: Confinement and Performance Projections, IA) Data Base Projections, IB) Dimensionally Similar Discharges Projected to ITER, IC) Impact of 1-Dimensional Transport Modeling on the ITER Design; Section II: Edge Density and Temperature Pedestals; Section III: H–mode Power Thresholds; Section IV: Density Limit; Section V: Particle/Impurity/Helium Transport and Fueling; Section VI: Energetic Particles and Burning Plasma Physics; Section VII: Macroscopic Stability Boundaries; Section VIIA: Ideal Stability Limits, VIIB: Non-ideal Stability Limits and Long-Pulse ITER Discharges, VIIC: Disruption Physics, VIID: Sawteeth. Findings, and recommendations are contained in these subsections.

I. Confinement and Performance Projections

Despite impressive fundamental advances in recent years, a predictive understanding of tokamak confinement remains a grand challenge for physics research. There is insufficient knowledge to precisely predict confinement and energy gain (Q) in ITER. Moreover, specification of the uncertainty, or error bars, in the projection cannot be evaluated rigorously. Projections arise from a combination of empirical scaling and approximate physics-based models, with results which are partly subjective. Modeling from present data and using present methodology can not guarantee ITER performance, since ITER aims for regimes never before produced. A sense of the panel is that the uncertainty in the DDR projected confinement time of 6 seconds is in the range of 50% (rather than 30% as adopted in the DDR). Thus, the expected energy confinement time lies in the range of approximately 3 to 9 seconds, corresponding to Q between approximately 4 and infinity (ignition). Among subpanel members there is a variation of opinion on the most likely value of the confinement and the magnitude of the uncertainty. In addition to confinement issues, there are other sources of uncertainty in ITER's performance, such as the stability issues discussed in Section VII. However, there is general agreement that continued progress in theory and experimental understanding will help narrow this uncertainty.

A quantitative confinement-related goal of ITER is to attain ignition and extended burn. Two related, implicit goals are to study burning plasma physics and plasma confinement under conditions close to that of a reactor. The uncertainty in Q values judged by the panel implies that there is considerable uncertainty of whether ITER will reach its goal of ignition. However, it is highly likely that ITER will be able to study burning plasma physics under reduced conditions ($Q \gtrsim 4$), as well as provide fundamental new knowledge on plasma confinement in near reactor conditions.

Recent years have brought significant progress in understanding present tokamak confinement and in the development of techniques to make projections to larger devices. The ITER design activities have stimulated and contributed to this progress. For example, the design activity has motivated the development of a global energy H–mode confinement database, from which the ITER confinement scalings have been obtained. Also, a profile database has been established that has enabled physics-based models to be systematically tested against a large international experimental database. There has been focused attention on one-dimensional transport modeling studies, especially within the U.S. fusion program and major progress in understanding transport physics has been made. The design activity has also led to a focused efforts in the non-dimensional scaling area, especially with regard to comparisons between tokamaks.

The DDR defines three techniques that are potentially useful for projecting ITER confinement and performance: global database scaling, non-dimensional scaling, and one-dimensional (1-D) transport modeling. However, the quantitative projection to ITER is based only on the global database scaling. Applicability of the non-dimensional scaling method is limited because of the closeness of the ITER operating point to the H–mode threshold and the Greenwald density limit (n_G); as a consequence, there is uncertainty as to whether the scaling in ρ^* (ratio of gyroradius to machine size) can be extended from present machines to ITER. The 1-D transport models are not used quantitatively because there is still no community-wide consensus on the validity and applicability of the models. In the U.S. fusion community, it is generally agreed that the leading candidate to account for much of the core transport is the class of microinstabilities driven primarily by ion temperature gradients (ITG modes), but agreement has not been reached on the quantitative predictions.

<u>Findings</u>: The EDA has advanced fusion science by stimulating careful and creative work on tokamak confinement. There has been a dedicated effort to include the knowledge and

results of the worldwide community in this area. The JCT has attempted to account for the recommendations of the Confinement and Database ITER Expert Groups and the associated working groups in developing its design. However, a consensus in H–mode confinement projections remains an unachieved goal and many issues remain to be resolved.

The uncertainty in the projections to ITER confinement remains large. The most important reasons for this uncertainty are the complexity of transport in tokamaks, especially in the improved H–mode confinement regime; the proximity of the ITER operating point to predicted, but imprecisely known, limits (H–mode threshold, density limits, etc.); and the deviation of the ITER operating point from that represented by most present tokamak experiments. The DDR quotes a confinement time of 6 seconds, with a 95% confidence level of ±30%. Although a rigorous evaluation of the 95% confidence level is beyond the scope of this review, a value ±50% seems more appropriate, corresponding to a range in confinement of approximately 3 to 9 seconds and a range in Q of approximately 4 to infinity (ignition). At the lower end of this range ITER will not meet the controlled ignition and extended burn objectives as outlined in SWG-1. In addition, the present tokamak experience, especially as represented by the bulk of the H–mode database, is characterized by ($n_e < n_G$, v_φ/c_S significantly > 0, $P_{Transport} > P_{Threshold}$). Because the ITER operating point is characterized by ($n_e/n_G \varnothing 1$, $v_\varphi/c_S \varnothing 0$, $P_{Transport}/P_{Threshold} \varnothing 1$), a regime, associated theoretically and experimentally with reduced confinement, we consider the 6 seconds quoted in the DDR to be optimistic for the high-density ELMing H–mode operational scenario.

<u>Recommendations</u>. It must be recognized that these physics issues are complex and multifaceted, and not likely to be fully resolved prior to the FDR or the end of EDA. We therefore recommend a research program directed at their longer-term resolution. Nevertheless, we should expect progress in each confinement projection technique (database scaling, non-dimensional scaling and one-dimensional modeling), with the goal of using all three in the FDR. The use of physics-based projections is especially important for gaining acceptance in the broader physics community. We believe that reducing the uncertainties in the present data and in the projections will require focused effort; the differences in the data from different devices as well as in the projections from the three different techniques need to be understood. Progress in the following activities within the U.S. fusion program will be beneficial toward reducing the uncertainty in the projected confinement, and contribute toward an improved physics basis in the FDR:

1. Experiments that quantitatively test the stiffness of ITG-based transport models.

2. Theory and numerical simulations that clarify the origin of the stiffness in the ITG models.

3. Experiments, analysis, and theory to develop a better physics understanding of and predictive capability for the pedestal height (it is noted that 1-D core transport models require an appropriate model for the edge).

4. Experiments, analysis, and theory work to better understand and predict the H–mode power threshold and the proximity of the threshold to the ITER operating point.

5. Experiments which are more representative of the ITER operating point (T_i/T_e, n_e/n_G, v_φ/c_s, $P_{Transport}/P_{Threshold}$, radiative divertor), timely inclusion of this data in the global and profile database, and the development of scaling relations including recently available data.

Significant progress is expected in Items 3 and 5 before the end of the EDA, but full resolution of Items 1–4 will require a research effort extending beyond the end of the EDA.

The most solid and rapid progress in physics understanding is likely to occur in the context of the international tokamak program. Understanding differences observed on various tokamaks is likely to result from experience (experimental, data analysis, theoretical interpretation) coming from detailed comparative studies on the several tokamaks. Therefore we recommend that the U.S. fusion energy sciences program take a strong initiative in encouraging and promoting collaborative experiments on existing tokamaks. Scientists should be encouraged to engage in this collaborative endeavor, within the present framework of cooperation.

IA. Database Projections

<u>Summary of the DDR Physics Basis</u>. 0D empirical H–mode confinement scalings that have been developed by the ITER Expert Group on Confinement Databases and Modeling have been used to guide the ITER physics and engineering design and to estimate the confidence in achieving the physics mission of the device. Log-linear scalings have been developed from a large database of 0D H–mode confinement and discharge data primarily

from six tokamaks for ELMy and ELM-free discharges. A set of selection criteria of interest to ITER was employed to choose a subset of H–mode data to be used as a basis for developing the scalings. The Expert Group design recommendation was to use the ELM-free scaling (ITER93H) multiplied by 0.85 to take into account confinement degradation found in ELMy discharges. The ELM-free expression was used as a basis due to what was determined to be larger uncertainties in the ELMy dataset. The thermal energy confinement for ITER, using ITER93H, is predicted to be ~6 seconds (~2.6 ITER89P L–mode), with a 95% confidence interval in the range of 30%, if one assumes that the power law scaling is correct, that all the relevant variables are included, and that the effective number of independent measurements can be approximated as $N_{eff} = N/4$.

The Expert Group has been careful in the data selection and in detailing the methods used to develop the scalings and the uncertainties. There are, however, issues of the adequacy of the scaling, the methodology used to develop the scalings, and the estimates of the uncertainties that can affect the prediction of ITER's performance.

Assessment of the Physics Basis. In order to predict ITER's performance accurately, it is essential to base any empirical scaling on an ITER-relevant dataset. The present dataset has densities in the range from 40% to 70% of n_G while ITER plans to operate from 1 to 1.3 times this limit. While operation with good confinement at high density has been limited, confinement times of 1.8 ITER89P have been produced on DIII–D with pellet injection and divertor cryopump operation for up to 0.4 seconds at 1.5 n_G, indicating there is no fundamental impediment to obtaining good confinement at high density. More data with $T_i(0) \sim T_e(0)$ would make the dataset more ITER-specific. At present, the temperatures in the dataset were provided almost exclusively by JET (NB heating), and these data show no reduction in confinement relative to the respective scalings either for ELMy or ELM-free discharges when $T_i(0) \sim T_e(0)$. Atomic physics effects (divertor action, wall conditioning) can have a significant effect on confinement, as shown in ASDEX and PDX (early 1980s). Only PDX has provided information related to these effects, and the treatment of this information in developing the scalings was *ad-hoc*. Information on toroidal rotation and rotational shear is missing, and the impact of these effects on the ITER prediction and its uncertainties is not known. Finally, the scalings could be made even more ITER-specific by constraining the data further on q, elongation, and, when available, T_i/T_e.

The choice of regressor variables impacts significantly both the value and the uncertainty of the confinement time prediction for ITER. It has been argued that the H–mode data are "overfit" by the ITER93H set of regressor variables, and that dropping density and minor radius increases prediction accuracy but reduces ITER's τ_E by 30%. The uncertainty in the confinement time estimate is influenced through the variation of the data along their principal component directions and through extrapolation to ITER in each direction. The two least varying principal components for the set of regressor variables that constitute ITER93H (I,B,P,n,R,a,k,M) are relatively uncertain, yet they contribute to over half of the confinement estimate. In addition, tokamak to tokamak variation can impact the uncertainty [1]. Using data from five tokamaks in the database to predict the confinement of the sixth, and then appropriately averaging the RMS error, leads to a 95% confidence interval for ITER of 3 to 12 seconds.

Forms for the parametric scaling of confinement time other than a simple power law form can be considered as well. An offset-linear functional form that was developed gives confinement estimates that are more optimistic than those given by 0.85 ITER93H. Another functional form is a log-nonlinear one, which attempts to take into account a statistically significant "curvature" in the ELM-free dataset. The two weakest principal components for the set of regressor variables used in this scaling have a smaller extrapolation to ITER than those based on the ITER93H regressor set. The confinement times given by the log-linear and log-nonlinear scalings are comparable over the range of the database; for the ITER prediction, the log-nonlinear scaling drops by 25% relative to 0.85 ITER93H due to the curvature. It is therefore essential to determine the significance of the curvature in the dataset. Work by the Expert Group indicated that the curvature disappears in the ELMy dataset and is reduced in the ELM-free dataset when different energy estimates (i.e., diamagnetic, MHD) are used as a basis for the thermal energy estimate.

Given the full range of confinement time predictions given by the different forms for the parametric scaling and the range of stored energy estimates, many members of the Expert and larger Working Group felt that a more realistic range for the 95% confidence interval for ITER confinement time predictions is from 3.5 to 9 seconds, which is approximately a 50% rather than 30% range.

Recommendations. Empirical global scaling relations for the energy confinement time have served a useful role in fusion research, providing a summary of experimental trends and a tool for estimating device performance. The performance of a fusion device is a strong function of confinement time; a 40% change in confinement is the difference between ignition and Q = 5. The design requires, then, high accuracy in predicting confinement or enough contingency to handle the ramifications of large uncertainties in performance. There are clear steps that can be taken that will allow us to make progress towards better predicting the ITER confinement time and reducing its uncertainty. The first is to include available information on effects that are known to influence confinement (e.g., sheared E∞B flow, divertor/main chamber neutral pressure, current profile, density peaking, ELM severity). A quantification of conditioning and divertor action that is common across tokamaks can be developed and input to the database. More ITER-relevant data and constraints will allow us to develop more ITER-relevant scalings. Understanding the data subtleties better may lead to adopting the most appropriate scaling expression forms. Finally, objective criteria need to be developed for selecting the "best" set of regressor variables and scalings. This notwithstanding, it should be again stressed that the choice for the most appropriate functional form cannot be properly made until the most ITER-relevant dataset is assembled and used as a basis.

Given the present large uncertainties in confinement predictions, there is a need for the ITER experimental facility and research program to incorporate enough flexibility and enabling technologies to meet physics and engineering objectives in the event that the confinement is worse than the anticipated 6 seconds. The ITER project may need to consider various alternative operating schemes; pellet or CT injection for fueling, IBW or beams to induce sheared radial electric fields, and greater plasma shaping or operation at lower density (with peaked density profiles) for enhanced performance are a few.

IB. Dimensionally Similar Discharges Projected to ITER Ignition

Summary of DDR Physics Basis. *Dimensionally similar discharge projection to ITER ignition rests on a simple and powerful idea under the ideal circumstances.* The relative gyroradius $\rho*$ is the single most important dimensionless plasma physics parameter to be extrapolated in confinement scaling. Restricting to ITER shaped tokamaks dimensionally similar discharges with all dimensionless parameters the same except $\rho*$, going from DIII–D (or C-Mod) discharge pairs to lined up JET pairs is approximately a threefold scaling in $\rho*$, and from JET to ITER is a further threefold scaling. JET is the pivot point to ITER. *Ideally* then each machine separately (and together for a wider variation in $\rho*$) can determine the transport power (or confinement time) scaling law: $Pa^{3/4} \propto 1/\rho*^{\xi}$. It is very clear that ITER motivated studies of projected dimensionally similar discharges has greatly enhanced our understanding of the method. The DDR does give one example of a JET-ITER demonstration discharge and seems only to infer, but does not explicitly claim, the method works for ITER in the ideal. Instead the DDR stresses dimensionally similar discharge experiments as a precision check on the 0D empirical scalings.

Assessment of Physics Basis. In projecting to ITER, how far is the *actual* projection method from the *ideal* in terms of accuracy, proximity to hard limits where the confinement is expected to degrade, exact power laws, and hidden variables? We focus here on the DIII–D [2,3] and JET [4] ITER H–mode demonstration 1 T ∅ 2 T $\rho*$-pair studies. The projected DIII–D demo has such high β and density (also favorably peaked), that even Bohm scaled power will suffice, whereas the JET demo has low enough β that only gyroBohm scaling will suffice.

Inaccuracy in determining $\rho*$ scaling arises first from imperfections in meeting the similarity conditions within pairs and secondly from temperature measurement error. Typically the constancy on β is almost perfect but for example $\nu*$ can be off by a factor of 2 making the $\rho*$ scaling more or less optimistic depending on the actual collisionality dependence. Assuming ±10% error bars in temperature can change the scaling from gyroBohm $\Omega\tau \propto 1/\rho*^3$ to $1/\rho*^{3\pm0.7}$ (almost Bohm $1/\rho*^{2}$). Such dissimilarity and sensitivity explains why gyroBohm models can fit temperatures in Bohm-scaled experiments (or vice versa). *It is fair to say that 1.6-fold* $\rho*$ *variations can barely distinguish gyroBohm from Bohm and more examples of wider threefold DIII–D (C–Mod) ∅ JET intermachine variation is much needed.*

There is a further concern about power law scaling and "hidden" or secondary variables. An exact power law gyroBohm $\Omega\tau \propto 1/\rho*^3$ scaling results from assuming a complete scale separation between eddy size and plasma gradient lengths. In recent years we have learned that E∞B rotational shear stabilization, has a diamagnetic or $\rho*$ component and a Mach number (M) component. Theoretically this breaks the exact gyroBohm scaling. Theoretical models suggest the rotational effects may have 20% effects on the DIII–D demo discharges and that rotational stabilization (along with T_i/T_e stabilization) will be much diminished in ITER effectively changing the gyroBohm scaling to Bohm scaling [5]. Recent TFTR co-counter beam experiments may resolve the M dependence.

There are in fact hard limits in projecting existing dimensionally similar discharges to ITER. A well know limit is the Greenwald density limit. The similarity density relative to the density limit scales close to $1/\rho_*^{1/2}$. It turns out that the projected DIII–D demo does somewhat exceed the Greenwald limit. Perhaps a more worrisome limit is the H–mode power threshold (P_{th}) and pedestal scaling with $\rho*$. The dimensionally consistent empirical scaling ($P_{th} \propto Sn^{3/4}B$) can be written as $P_{th}a^{3/4} \propto (1/\rho_*^3)$ which is Goldston-like. Thus gyroBohm $1/\rho_*^{3/2}$ scaled or even Bohm $1/\rho_*^{5/2}$ scaled H–mode power tends to fall below the required threshold power in extrapolating to ITER at the high density operating point. Indeed this appears to be the case for the gyroBohm DIII–D pair [3]. The gyroBohm scaled transport power at 33 MW and even the Bohm power at 255 MW is likely at or below the threshold at the projected density which exceeds Greenwald. This may explain why the full field JET 1.7 T ∅ 3 T pair [4] has such poor Goldston scaling in proximity to the L/H threshold. Good H–mode confinement is dependent on the pedestal β which may have a pessimistic $\rho*$ scaling [6]. If one plots edge data for βq^2 versus ρ_{pol}/R for the DIII–D [2,3] and JET [4] pairs all together, a disturbing linear relation is clear over a threefold span in ρ^*. At ITER densities, this will extrapolate H–modes to L–mode boundary temperatures. However taking the pairs individually, there is no clear evidence of pedestal β degradation with 1.6-fold $\rho*$ variation within the pairs. Is $\beta q^2 < \rho_{pol}/R$ a hard limit, or is JET too under powered to operate at a higher edge β?

Recommendations. The FDR should be more explicit in discussing the limits to the method. *Confidence in ITER ignition could be significantly increased by actually making a full field, high powered, high-β JET "pivot point" pair similar to and actually lined up with a DIII–D pair with the same pedestal-β and scaling to ITER ignition.* Failing that for lack of JET power, efforts should continue to construct a derated DIII–D pair to match the existing intermediate JET demo pair verifying its requirement of gyroBohm scaling. It seems unlikely these extrapolations will operate clear of the density and threshold limits at the present design point and it would be useful to find backup derated driven burn points where the method works in the ideal.

IC. Impact of 1-D Transport Modeling on the ITER Design

The ITER Expert Group on the Confinement Database and Modeling assessed the status of local transport models and their ability to model the profiles and performance of present ITER-relevant tokamaks [7] in order to judge whether local models are sufficiently accurate to predict ITER performance. An ITER profile database was assembled with density and temperature profiles from a variety of machines in a variety of performance regimes. Tests were carried out with eleven models. The models were typically able to predict performance of present machines to within 15%–35% but differed widely on their predictions of ITER performance. The range of projections included both success and failure to achieve its ignition and high power burn objectives. The DDR therefore concludes that it is “not yet possible to clearly discriminate between these models” and no significant discussion of their implications for ITER performance or the assessment of the 0D scaling laws is presented.

Local Transport Models: An Assessment. There is no currently accepted model for anomalous transport in tokamaks. Nevertheless, it is widely acknowledged that there has been significant progress in the development of first principles models of ion energy transport. Particle and electron transport are less well understood. The progress in understanding ion energy transport is a consequence of the identification of the ITG mode as the likely dominant drive for transport in the core of many tokamaks, combined with the development of simulation techniques for studying the 3-D turbulence driving transport in realistic toroidal geometry. Of the eleven models tested by the Confinement Database and Modeling Expert Group, the four with the strongest theoretical underpinning [5,6,8,9] are built around models of ITG mode transport although Waltz's model [5] now includes dissipative trapped particle and ideal MHD modes. The multimode model [8] includes resistive ballooning modes in the colder edge plasma. None of the theoretical or empirical models properly describes the edge pedestal associated with H–mode barrier since there is at present no first principles theory of this pedestal. Thus, the models apply only to the core region inside of the edge pedestal during H–mode operation. Two of the ITG mode models [5,6] are completely theory based: the numerical values of the transport rates are based on comparisons with 3-D gyrofluid simulations with no adjustable parameters.

All of the local transport models predict the performance of the discharges in the ITER profile database to within about 30%. This should be considered a major success for the models which are nearly completely theory based. The multimode model [8] has the best fit to the data (within about 15%), although this should perhaps be expected in a model which has some empirical fitting parameters. Unfortunately, the performance projections for ITER differ greatly among the models and for most of the ITG mode models the performance is very sensitive to the assumed height of the edge temperature pedestal, pedestal temperatures of the order of 3–4 keV being required to achieve ignition. The sensitivity to the edge temperature arises because the ITG mode produces very large transport if the ion temperature gradient is significantly above the threshold for linear instability. As a consequence, the core plasma in the simulations tends to fall close to marginal stability and the temperature profile can to lowest order be obtained by simply using the marginal stability condition for the ITG mode to map temperature of the edge pedestal into the core. In such models the addition of more auxiliary power does little to raise the central temperature because the transport rates can increase sharply with little change in the temperature profile. The profiles are therefore “stiff”. The multimode model does not display the sensitivity to the edge temperature in the ITER projections in spite of its ITG mode foundation. This model predicts that ITER will ignite even with a pedestal temperature of the order of 400 eV. The transport rate in this model, unlike the other ITG mode models, includes a high inverse power dependence on the elongation of the cross-section κ. This dependence reduces the transport in the ITER projections and allows the profile to rise well above marginal stability.

Despite the lack of complete agreement between the transport models, they have a number of common features. In particular, many of the models predict improved plasma performance with increasing T_i / T_e, with more peaked density profiles, or with sheared rotational flow, and there is observational support for some of these predictions. Each of these effects are expected to be weaker in ITER than can be obtained in present experiments, due to ITER's reliance on α-particle heating, edge or near-edge fueling, and large size. Thus, our present understanding of plasma transport indicates that these parameters must be carefully controlled when extrapolating ITER's performance from present experiments. The present data mix these effects, casting doubt on the accuracy of the empirical projections of ITER's performance presented in the DDR.

Recommendations. A high priority should be assigned to resolving the discrepancies between the transport models, particularly those based on the same underlying physics, in order to narrow the range of predicted ITER performance. Examples include the dependence on κ, the stiffness of the profiles, the effect of flow shear, and the turbulence saturation levels. Experiments testing the points of discrepancy should also have priority and should be compared to the model predictions. Finally, because of the likely sensitivity of the performance projections to the edge conditions in H–mode operation, the development of a model of the H–mode pedestal or an empirical projection of the pedestal height should be vigorously pursued. With timely resolution of the discrepancies between models and development of an initial pedestal height prediction, 1-D modeling of ITER performance, with estimates of uncertainties, should be included in the FDR.

In addition, the empirical performance projection methods should be more carefully controlled for transport-influencing effects present in current experiments. Either the experiments contributing to the 0D database and dimensionless projections must be constrained to have ITER-like density and flow shear profiles and T_i/T_e, or experiments must be completed which pin down the impact of these variables on the confinement so that the ITER performance projections can be corrected.

II. Edge Density and Temperature Pedestal

The DDR estimates of the pedestal values are obtained by extrapolating a pressure gradient over a pedestal width. The gradient is generally taken to be limited by MHD, nominally the ideal ballooning first stability boundary. The width of the edge barrier is considered to either scale with R (resulting in constant β_{ped}), or with ρ, which would result in substantially lower pedestal beta in ITER than in present experiments. The pedestal density is taken to be close to the average value, close to the density limit. Implied values of the pedestal temperature are in the range of 200–900 eV if the width scales with ρ*. In the context of performance projections using 1D modeling, the reference value of T_{ped} is stated to be 2.5 keV, which is claimed to correspond to a "pedestal width" of 0.1 times the minor radius, using a non-standard definition of the width.

An ITER internal memorandum [10], generated after the DDR, attempts to improve the quantitative prediction for ITER as well as the physics basis of the projections. The tentative conclusions of this "preliminary" study for ITER show $0.4 < n_{ped} < 1 \infty 10^{20}$ and $1.5 > T_{ped} > 0.6$ keV, keeping $n_{ped}T_{ped}$ at the ballooning limit. The pedestal width is extrapolated from fits to available data and corresponds to values in ITER ranging from 3–6 cm; these empirical fits do not scale like ρ, and in some cases are not dimensionally correct. A critique of [10] is beyond the scope of the present review; it is mentioned here because it contains more quantitative results than the DDR and indicates an ongoing effort in this area.

There is theoretical and experimental support for estimating the pedestal height on the basis of a width scaling like ρ_{pol} and a gradient given by ballooning mode (1^{st}) stability, leading to a low pedestal estimate for ITER; however, there are exceptions in the data and plausible alternative theories and scalings exist. Data from JT–60U [11] shows excellent correlation with a ρ_{pol} scaling; however, data from JET [12] and DIII–D [13] show less scaling with current than would be expected for this dependence. Furthermore, an experiment on DIII–D in which β_{edge} and ν^*_{edge} were held fixed and ρ^* varied by about 50% showed no change in the pedestal width [14]. Yushmanov's analysis [15] of a DIII–D database indicates the pedestal width scales as a hybrid of ρ_{pol} and the width of a region of second stable access near the separatrix, but there are significant uncertainties. While a theoretical argument for a scaling with ρ or ρ_{pol} can be made [6], alternative models, such as one [16] in which the width scales as $(D_{neo}/\nu_I)^{1/2}$, where ν_I is the ionization rate, are also compatible with the data. The general problem of transport barrier width, of which the H–mode pedestal is a special case, must also account for the broad internal barriers encountered in VH–modes, which are not limited to a few ρ_{pol}. While type-I ELMy discharges typically have edge gradients close to the ideal ballooning limit, there are counter-examples. In many ITER-shaped DIII–D discharges [13,15] the edge has access to the second stable region. Moreover, in cases where the first stability boundary does exist, or where an extrapolated value is meaningful, there exist data [13] for which the experimental electron pressure gradient alone exceeds this limit.

While, given the present state of knowledge, we cannot provide a reliable estimate of the pedestal parameters in ITER, it must be stated that a pedestal temperature less than 1500 eV, perhaps much less, is a distinct possibility. The severity of the consequences of such a low pedestal depend on the validity of the stiff marginal stability transport models, which predict rather poor performance ($Q < 5$) for ITER if the pedestal temperature is less than 3–4 keV. The sensitivity of such theory-based projections highlights the importance of improving the pedestal predictions.

Recommendations. Physics basis improvement requires the generation and analysis of well-documented experimental pedestal data in existing divertor tokamaks. Theoretical models need to be elaborated to the point where experimental tests can be performed. Significant improvements in physics understanding, potentially providing better projections to ITER, could be achieved within the remaining one-year-plus of the EDA, though full resolution of these issues will take longer. The U.S., with experimental data from DIII–D and C–Mod, and theory development closely coupled to experimental work, could play a substantial role, provided significant resources are dedicated to this effort.

III. H-Mode Power Thresholds

Given the complexity of the physics involved, the ITER team has done a good job assessing the existing experimental data and projecting the H–mode power threshold for

ITER. The H–mode power threshold is a key issue for the ITER design, since ITER requires a significant energy confinement enhancement over L–mode to meet the goals established for the design and ELMing H–mode is the one reliable enhanced confinement mode with proven long pulse capability. The present design calls for operation with power through the edge near the nominal H–mode threshold.

An empirical model of the power threshold scaling is basically the only choice at present, given the complexity of the physics which can influence the power threshold. A fully validated, first principles theory which could be used to predict the threshold power is beyond the present state of the art. Such a theory would require, first, a complete theory of the divertor plasma, since the divertor physics sets the boundary condition on the separatrix. Second, a complete theory of L–mode transport in the plasma edge near the separatrix would be required. Finally, the transition condition would have to be theoretically specified in terms of the local edge parameters, which might have to include the amplitude and phase of the turbulent fluctuations. We have a semi-quantitative understanding of the formation of the H–mode transport barrier through the mechanism of E∞B shear stabilization of turbulence. However, at present, there is an evolving picture of the edge plasma conditions necessary to create the initial electric field and start the transition. Accordingly, empirical scaling based on data from various tokamaks is the only feasible approach.

The initial empirical threshold power scaling utilized engineering variables and found that, with a significant amount of scatter in the data, the plasma density and toroidal field were the key variables influencing the power threshold. Based on ideas of dimensionless scaling and assuming the plasma physics variables are the key, the ITER Expert Group on Confinement Database and Modeling has obtained the power threshold scaling relation used in the DDR. Unfortunately, no experiment has yet demonstrated that only plasma physics variables are important in determining the H–mode power threshold.

It is clear from the large scatter in the H–mode power threshold data that key variables are probably missing from the threshold scaling relation employed in the ITER design. This scatter leads to the large range in the prediction of the power threshold given in the DDR.

A key recent realization embodied in the DDR is that ITER can meet its goals even if the power threshold for the forward (L-H) transition and the back (H-L) transition are equal at the nominal forward transition level. This means that a power hysteresis is not required. However, if the H–mode power threshold is at the maximum given in the DDR, the H-L threshold would have to be about 1/2 that maximum for successful operation. It appears that the greatest problem with a high threshold would be the need to purchase more auxiliary power to produce the initial L to H transition. This power would be required only for a short time to trigger the transition. Only if there were no hysteresis and the power level required is at the maximum would there be a fundamental problem.

Because of the potential serious impact on ITER operational scenarios, the uncertainty in the H–mode power threshold projection and the hysteresis range should be reduced. The U.S. fusion program can help reduce the range of uncertainty in the power threshold projections for ITER in several ways:

1. Provide the L to H and H to L threshold scaling data requested by ITER through the Expert Groups.

2. Continue physics-based investigations of H–mode threshold and transition physics, as called for in the TAC 11 report, stressing local measurements in the plasma edge.

3. Directly test the idea that the H–mode power threshold depends only on plasma physics variables.

The H–mode power threshold scaling, especially the scaling with size is a critical research topic for ITER, as is noted in the DDR. Increase resources are needed in this area, since the same limited manpower now available to address this issue is also being asked to provide support for other important ITER Urgent Tasks, such as H–mode pedestals and ELMs. The most solid and rapid progress in this area will be provided by greater theoretical insight.

IV. Density Limit

<u>Assessment of Physics Basis</u>. The DDR overstates the theoretical understanding of the disruptive density limit, downplays the observed robustness of the Greenwald limit [17], and emphasizes those relatively unusual experiments where the empirical limit is exceeded. While there is general agreement about the density limiting mechanisms, current theories

do not successfully predict when this will occur with respect to global parameters. In particular, theories of a radiative collapse predict stronger dependence on input power and impurity content than is seen in experiments. Further, they suggest that Marfes and/or divertor detachment should occur just before the limiting density is reached, a prediction that is contradicted by experiment [18].

The DDR analysis of the H-L transition as an effective density limit also has shortcomings. We note that the existing H–mode data base is conformal to the Greenwald density limit, n_G when plotted in the I_p/a^2 vs. n_e plane, data from “good” H–modes are bounded by a line parallel to the density limiting line. The DDR suggests that the pedestal density should scale like $1/\mathrm{sqrt}(T_{edge})$. The limit calculated in this manner has not been experimentally verified, and we are unaware of any experimental evidence supporting the DDR prescription of increased plasma shaping (within the limits of the present design) or divertor baffling for improving the H–mode density limit. Nevertheless, the notion that the pedestal temperature and the pedestal density are qualitatively inversely related has experimental support, and compounds concerns that the DDR expectation of high-performance operation with high density is optimistic, particularly with flat density profiles.

Although flat density profiles are expected for ITER, it is noted in the DDR that experiments with strong central fueling and peaked density profiles have shown the ability to exceed the empirical limit. Newly reported data from DIII–D [19] have shown operation with $n/n_G \sim 1.5$ and $H \sim 1.8$, $1.5 < n_e(0)/\langle n_e \rangle < 2.3$. So far, these results have only been obtained at low input power — central pellet fueling and peaked density profiles are harder to achieve with high input powers. Furthermore, most of the existing database is from plasmas with neutral beam heating which often makes an important contribution to plasma fueling. Discharges with predominantly ICRF heating tend to have lower densities with respect to the limit. At best, ITER may employ high energy beams which will make a smaller contribution to overall fueling.

High density operation has not been a priority for the fusion program. The result is an experimental database which is only lightly populated at high densities. Operation above 0.85 times the Greenwald limit accounts for about 1% of records with H–mode confinement. Since exploration of the density limit has not been a goal for most experiments until recently, this does not necessarily rule out high density operation in the future. However without further successful experiments, plans for operation at or above the empirical limit evidently entail substantial risk to the ITER mission.

Recommendations. Firstly, we recommend a vigorous experimental campaign aimed toward producing high density discharges with high confinement, especially in the ITER similarity configuration. This should include studies where central fueling is negligible in order to determine if an acceptable operating regime exists without the need for deep fueling. Secondly, we should explore techniques relevant to ITER for core fueling and for producing peaked density profiles in high performance plasmas. Thirdly, we recommend that the ITER team assess scenarios with $n/n_G = 0.85$, in case the first two activities are unsuccessful. Finally, a concerted experimental and theoretical effort to understand the physics which underlies density limiting phenomena should be undertaken.

V. Particle/Impurity/Helium Transport and Fueling

The performance projected in the DDR simulation is based on three primary assumptions regarding particle transport: 1) a flat density profile, $D_e = D_{He} = D_{fuel} = \chi_e$ and neoclassical particle pinch; 2) He density is self-evolving given the above transport properties, alpha particle source rate, and assumed τ_{He*}/τ_E for recycle and 3) impurity profile shape the same as the electron density profile. Although there is little theoretical justification for these choices outlined in the DDR, most of the assumptions appear to be consistent with experimental observations. Plasma fueling is primarily via gas injection with pellet injection or some other means of central fueling as a backup.

Given that the fuel density profile for ITER is a crucial element of the simulations, it is surprising that little physics justification is given for the choice of a flat density profile, other than that it is considered conservative in terms of performance projections. Few 1-D models (with the multimode model being a notable exception) presently incorporate predictions of the density profile although it is possible to determine particle transport properties from the various turbulence models. The DDR prediction that helium transport in the core plasma will not be a serious cause of ion dilution in ITER appears to be well founded and based on helium transport and exhaust measurements. Theoretical models for turbulent contributions to impurity transport are almost non-existent (excepting the multimode model, which cannot predict impurity source strengths). The credibility of argon injection to radiate a large fraction of the power is therefore difficult to assess with modeling. The fueling issue seems to be somewhat avoided in the DDR though the JCT has been active in pushing for shallow pellet fueling experiments. In this regard, pellet fueling appears to be the primary choice since there appears to be little likelihood that compact torus fueling will be investigated on a major facility any time soon.

Recommendations. The assumptions made in the DDR with regard to particle transport and fueling are reasonable, given the present set of information available. Experimental study of the fueling efficiency of pellets versus penetration depth in ELMing H–mode plasmas, including an assessment of inside launch is needed. This would lend credence to the assumption that shallow pellet injection will be sufficient in ITER. An assessment of shallow pellet fueling could be completed prior to the end of the EDA. Secondly, a better characterization of the expected impurity level, especially for argon or other high-Z impurity, would help to reduce the uncertainties in the projected performance: this evaluation would likely not be completed prior to the end of the EDA.

VI. Energetic Particles and Burning Plasma Physics

In an ignited D-T plasma, the alpha-particle power must be transferred to the thermal plasma before it is lost to the vacuum vessel wall. Energetic alpha-particles are the main source of the plasma heating in ITER and hence a good alpha-particle confinement is important for achieving ignition. The main issue for the ITER design is possible mechanisms of energetic particle loss and estimates of local heating of the first wall.

Alpha-particle confinement and loss has been measured in TFTR using unique and novel alpha-particle diagnostics. It was demonstrated during D-T experiments that alpha particles are well confined in MHD-quiescent plasmas on TFTR, the measurements showed classical behavior of alpha heating and ash buildup as predicted by code calculations. The theoretically predicted toroidicity-induced Alfvèn eigenmode (TAE) has been seen on TFTR and no alpha loss was observed with $\beta\alpha$ ♠ 0.1%.

The DDR has used the predictive capability obtained over the last few years to model the behavior of energetic particles and alpha-particles. They have studied possible mechanisms of energetic particle loss and estimated local heating of the first wall. Clearly a number of significant issues need still to be resolved in order to have high confidence in achieving ignition in ITER.

Recommendations. Probably the most significant problem for ITER in the energetic particles and burning plasma physics area will be major disruptions. Alpha particle loss at disruptions is very localized, and this could lead to serious problems with the first wall heat loads. Alpha loss caused by the instability of TAE, kinetic ballooning modes (KBM), kinetic TAE (KTAE), beta-driven Alfvèn modes (BAE) and beam modes (EPM) could still be a problem for ITER. More effort in theory and experiment is needed to address this area. A working high-nq stability code which treats the energetic-particle and the core-ion FLR physics non-perturbatively is needed. The progress on such a code has been slow and may need further international/national coordination. The TAE stability needs to be assessed for the advanced tokamak scenarios, due to the TAE sensitivity to the q-profiles which has been observed in TFTR.

The DDR modeling results show that ripple loss of the alpha-particle and the NB ions could be significant in some of the ITER reference plasma scenarios. Suggestions and methods to reduce the ripple loss are discussed in the DDR report, but a decision is needed on the best course of action for the ITER design.

VII. Macroscopic Stability Boundaries

Realistic adherence to macroscopic stability constraints is arguably the most important and fundamental of the requirements for the design of any magnetic confinement device. The starting point of scoping studies is the examination of ideal MHD equilibrium and stability — fortunately an area where the physics basis with respect to conceptual formulation and the necessary computational tools are well established. Using the profiles generated from PRETOR simulations, ITER equilibria and ideal stability have been systematically analyzed worldwide and have led to the DDR conclusion that ideal MHD stability is not expected to be a performance-limiting issue. With respect to the results from the analyses carried out to date, the panel concurs with this statement. However, there remain significant questions regarding the choice of more realistic profiles. This is, of course, related to the challenge of predicting such profiles from confinement and transport projections — a major topic featured elsewhere in this report. On the time scale relevant to the assessment of the FDR, the panel recommends that (i) for the ITER profiles of choice, systematic current and pressure profile variations be analyzed and more robust ideal stability diagrams be generated. It is also recommended that (ii) in order to help make more compelling the arguments for more flexibility in the ITER design to accommodate improved performance tokamak scenarios, dedicated experiment/theory initiatives should be pursued to quantitatively assess the benefits of increasing triangularity, reversed shear, and high internal inductance.

In dealing with the key issue of beta limits in long-pulse discharges lasting several energy confinement times, the DDR notes that the achieved beta limit is around 40% below the ideal threshold. The panel concurs with the assessment that gaining the physics understanding necessary to project the actual non-ideal stability constraints on ITER should receive the highest priority and strongly recommend support for theory/experimental initiatives to devise means to expand such stability bounds. An important goal in this area is to quantitatively predict the critical widths and parametric dependencies of seed islands (from, e.g., sawteeth, ELMs, fishbones, etc.) which can trigger resistive modes such as neoclassical tearing instabilities. Possible dynamic control of these modes could be demonstrated, for example, by a feedback scheme to replace the missing bootstrap current inside the island via local current drive (e.g., using ECCD or LHCD systems). In general, avoidance of non-ideal beta limits is needed because violating such bounds can produce a soft beta collapse with fractional confinement degradation and the occurrence of locked modes followed by a hard disruption. Improving the non-ideal beta limits is also expected to have a beneficial effect on reducing the disruptivity.

The presence of disruptions (fast plasma terminations) is arguably the most critical issue for tokamaks in general and for ITER in particular. As such, it is the sole topic of another subpanel of this FESAC ITER Review. Here we address the physics basis for this key subject and would conclude that the level of understanding is mostly empirical. With regard to the impact of disruptions, the DDR has produced a relatively clear picture of the most important physical processes including halo currents, vertical disruption events (VDEs), and runaway electrons. Identifying as key design parameters the maximum magnitude and the toroidal distribution of the in-vessel halo currents, the ITER design team has done a good job of deducing the associated scaling trends from a large, multi-machine, international database. It is recommended that their plans for further theoretical/experimental investigations to narrow the range of uncertainty for these key design parameters should receive high priority and would likely lead to significant improvements on a time-scale to impact the FDR.

The frequency of disruptions is another major issue that is far less well addressed. At the present time, the relevant experimental database for studying this problem is inadequate and the theoretical tools needed to complement the analyses are largely undeveloped. The panel has found no acceptably-documented data from any of the leading tokamak experiments which would compellingly support the present DDR position that the disruption frequency near ITER-relevant stability bounds can be expected to be 30% or less. It is strongly recommended that dedicated experiments on DIII–D, C–Mod, JET, JT–60U, and ASDEX-U be carried out which systematically examine whether discharges operating at ITER-relevant values of ν^* and near ITER-relevant beta and q limits (not individually but simultaneously) can successfully avoid disruptions for long pulses (greater than 2 seconds). Additionally, disruptivity in these devices should be evaluated near the density limit at ITER-relevant values for beta and q. We believe that significant new results on the needed timescale are likely to materialize with a “major-push” effort of this kind.

The DDR has pointed out that the presence of an electron runaway current during disruptions can adversely modify the disruption dynamics and possibly lead to very localized heat deposition on the first wall. The panel concurs with the assessment that runaways could seriously damage in-vessel components and supports their recommendation that the expeditious development of an integrated disruption/VDE model with runaways is essential and should receive high priority with respect to both experimental run-time and theoretical focus.

Concern about sawtooth activity in ITER arises primarily because of the potentially adverse effect on confinement of thermal and fast-alpha particles and because they might trigger disruptions. On the basis of semi-empirical modeling studies, the DDR is cautiously optimistic that major difficulties will not occur. At present, none of these sawtooth models have received the level of testing against data given to some confinement models, and there remains controversy over the requisite physics elements in them. The panel concludes that based on the results from analyses using presently available data and models, the DDR position on sawteeth appears reasonable but on the optimistic side. It would clearly be desirable to have a more realistic physics-based model which could yield more substantive predictions of alpha loss and the possible coupling of sawteeth to modes that might degrade confinement outside the mixing radius. More systematic experimental validation of the predicted mixing radius (e.g., better documented evidence supporting a large mixing radius in JET) would also be very valuable.

VII.A. Ideal Stability Limits

The DDR statement that "ideal MHD stability is not expected to be a performance-limiting issue" is essentially correct for the series of profiles analyzed. The basis of this statement is the analysis of ITER equilibria based on PRETOR profiles [20] which included work done by several groups around the world. For the PRETOR profiles studied, there was general agreement that the β-limit is at β_N ♠ 3.4 and is set by the high-n ballooning mode. This would appear to provide a substantial margin of stability, since the β-limit for the n=1 mode is even higher, at $\beta_N = 4.5$. However there are significant questions associated with the choice of profiles. For example, the choice of q_{axis} greater than 1, combined with the fact that the pressure profile is broad contribute strongly to the stabilization of the m/n = 1/1 mode. If q_{axis} falls below one, even with the broad pressure profile, the β-limit drops below the ITER operating value, $\beta_N = 2.5$. If the pressure profile is more peaked, even with q_{axis} greater than 1, the β-limit would drop. On the other hand, experimentally, many discharges with q_{axis} less than one exceed the m/n = 1/1 β-limit. Because of this and because of the relatively high ideal β_N limit for the model profiles, it seems likely that ideal MHD instabilities will not be a critical limiting issue.

The DDR has addressed a specific scenario, which should be expanded to address the possible variations. The only profile scenario considered is based on a model where sawteeth benignly provide the two key elements, q_{axis} greater than one with an extended low-shear region, as well as a broad pressure profile with virtually no pressure gradient in the low-shear region. Since the likelihood of regularly achieving such conditions is rather questionable, it would be very worthwhile to consider both q- and p-profile variations, to determine a robust stability diagram in plasma parameter space.

Apart from the β-limiting role, ideal instabilities may also play a role in limiting the performance of ITER. This could occur through ELMs, driven by kink or ballooning modes, and giant or monster sawteeth, driven by the m/n = 1/1 mode. The latter is a strong possibility because of the predicted large q=1 radius. The correlation of ELMs with ideal instabilities needs to be thoroughly documented so that an ITER-relevant model can be properly developed.

With respect to possible ideal stability advantages gained from advanced modes of operation, reversed shear with an internal transport barrier is seen as a promising option for ITER. However in order to actually access this regime stabilizing the external kink is essential. At the present time, ITER still needs to develop a credible plan to achieve this stabilization. In the context of the 1/1 mode, theoretical analysis shows that triangularity is known to significantly enhance stability. There is also increasingly prominent experimental data, (e.g. from JT–60U), that higher triangularity improves the performance. If more flexibility in the ITER design were possible, the benefits of increasing the accessible range of triangularity could and should be seriously pursued. Another possibility for improved performance with respect to ideal MHD constraints is the high-l_i mode with q_{axis} greater than unity. The difficulty here is the ability to sustain the current profiles for a long time. This may require enhancements in the RF heating and current drive capability. A design extension or upgrade for such a scheme should be explored. In general, the prospects for the ITER advanced mode operation are significantly more challenging than the conventional mode. Even though there is much evidence that tokamaks in this regime behave in conformance with ideal theory, the associated requirement for stabilization of the external kink is nevertheless quite challenging. It would be highly desirable for the US to develop a comprehensive program to stabilize external kink modes using an external feedback scheme such as the Fitzpatrick-Jensen model. This would also provide benefits in the key area of disruption control.

VII.B. Non-Ideal Stability Limits: Long-Pulse ITER Discharges

While ideal MHD limits adequately describe the upper bound of achievable beta values in present day tokamak discharges, these limits have only been sustained for short durations. For discharges which lasted over several energy confinement times produced in ASDEX-U, COMPASS-D, DIII–D, JT–60U and TFTR, and specifically for long-pulse ITER demonstration discharges, the achieved beta limit is significantly lower. The typical value of β_N is in the range 1.5–2.5 in ITER demonstration discharges, approximately 40% below the ideal limit. The onset of the non-ideal beta limit is accompanied by what appears to be resistive modes with low m/n mode numbers. The beta limit can result in a soft beta collapse with fractional degradation of confinement, the occurrence of locked modes, and a hard disruption. While ITER only requires β_N ε 2 to be able to ignite, the presence of the non-ideal limit close to the marginal value is of serious concern. Accordingly, sufficient understanding of the physics responsible for the non-ideal limit is essential both to project the stability performance for ITER and to devise approaches to expand the stability boundary.

Studies in this high priority area have indicated with some certainty that the beta limit is not due to lower ideal limits when the profiles are not optimized. Profile diagnostics and stability calculations have improved in recent years allowing quantitative validation of ideal MHD stability theories. Results from ITER demonstration discharges indicate: (i) ideal stability with respect to external kink modes and to ballooning modes in the core; and (ii) conventional resistive and neoclassical modes may be responsible for the beta limits. The growth time of the low m/n modes observed before the beta collapse is consistent with the resistive time. They occur in the presence of sawteeth, ELMs, or fishbone activities which can provide the seed islands to trigger the instabilities. When the instability is excited, examination of the sign of Δ' is one way to distinguish between conventional (+) and neoclassical (-) tearing using tools such as the PEST-III code. However, for some neoclassical regimes where such codes do not have enough accuracy to resolve nearly singular behavior, other scaling arguments need to be used.

Neoclassical tearing modes are driven by pressure gradient or β_P. Theory predicts a β_P threshold above which the mode becomes unstable and the mode width increases with β_P. Also, because of the dependence on collisions (more stable at higher collisionality), the evolution of the island width with β_P has a hysteresis behavior. Both features are in qualitative agreement with experiments. Data from TFTR (also DIII–D and others) have shown that after neutral beams (NB) are turned off, the observed island decay tracks the bootstrap term very well. Also, during the NB phase, a beta-collapse event reduces the island as expected. All these are clear examples of phenomena outside the realm of ideal MHD theory. In order to project ITER performance close to beta limits, it is necessary to understand the origin and magnitude of the seed islands (from, e.g, sawteeth, ELMs and fishbones) which are required to trigger neoclassical tearing instabilities. The requisite theory for quantitative predictions of critical seed island widths and their dependencies on machine and plasma parameters is incomplete. The dependence of the stability limit on triangularity suggested by JT–60U results are also not well understood theoretically. Competing models include those based on finite thermal conductivity and on ion-polarization current effects. A strongly focused effort combined with experimental validation would be needed to resolve this issue within a reasonable time.

Various proposals to raise the beta limit by controlling non-ideal modes have been proposed. Even though increasing the collisionality would stabilize the main driving terms, this cannot extrapolate to ITER conditions. Sawteeth can be removed by raising $q(0) > 1$, but it is difficult to provide the profile control needed for long-pulse. The understanding of ELMs is incomplete, and ELMs control is in its infancy. Ideas for controlling neoclassical tearing modes falling in the global current profile control category include making Δ' more negative and producing negative central magnetic shear. Both should have stabilizing effects which are insensitive to the details of the mechanisms. Possible dynamic control of these key instabilities could be accessible by a feedback scheme to replace the missing bootstrap current inside the island via local current drive produced, for example, by ECCD or LHCD. Although the efficiency of this approach appears to be qualitatively reasonable, demonstration experiments involving, e.g., off-axis ECCD in existing experiments will be needed for validation. The associated theory needs to be developed to help provide the basis for designing experiments to test these schemes on existing tokamaks.

VII.C. Disruption Physics

The severity and frequency of disruptions (fast plasma terminations) are obvious major issues for tokamak reactors. They can cause high electromagnetic forces and thermal loading. Associated vertical disruption event (VDEs) and runaway electrons can have very serious effects on the in-vessel components and supporting structures. Loss of plasma vertical position control in ITER would result in a VDE which causes plasma wall contact and ex-plasma ("halo") current flow that close poloidally through conducting in-vessel components. This in turn produces forces on these components and their supporting structures. In assessing the severity of disruptions, the DDR has identified the maximum magnitude and the toroidal distribution of the in-vessel halo current as key parameters. Utilizing a large multi-device database, the design team has done a good job of examining the scaling trends and extrapolation to ITER-like conditions. In order to narrow the range of uncertainty about the maximum halo current magnitude and the toroidal peaking parameters during disruptions and VDEs, it will be necessary to further enhance the experimental and theoretical R&D in this key area.

In planning for the ITER plasma facing components, vacuum vessel, and magnetic system to withstand several thousand plasma disruptions, the DDR allows for a major disruption frequency of up to 30% and ~3,000 disruptions with a range of severities during the 11,000 discharges which occur in the physics phase. However, there has been no compellingly-documented data from the leading international tokamak experiments which supports the contention that the disruption frequency near ITER-relevant stability bounds can be expected to be less than 30%. Dedicated experiments on devices such as DIII–D, C–Mod, JET, JT-60U, and ASDEX-U urgently need to be carried out which demonstrate that discharges operating at ITER-relevant ν^* and close to the beta and q limits (not individually but simultaneously) can successfully avoid major disruptions for long pulses (greater than 2 seconds). It would also be necessary to examine the disruptivity near the density limit in these devices at ITER-relevant values for beta and q. The design basis for assessing ITER disruption and VDE-related design issues are well documented in the DDR document and in published literature. The DDR includes a discussion of the MHD phenomena that lead to onset of a disruption and then the rapid loss of the plasma thermal energy confinement. The thermal quench, which results in cooling of the plasma to ≤ 100 eV, is followed by a subsequent current quench and in an elongated tokamak like ITER, onset of a VDE. This analysis has been well done by the ITER design team and there is a lot of experimental data and modeling to verify the results.

For ITER disruption scenarios, the substantial conversion of plasma current to runaway electron current with a very long lifetime is another very serious issue discussed in the DDR. Secondary runaways can be produced by knock-on avalanche multiplication during the "thermal" and "current" quench phase of disruptions due to the sudden increase of the inductive electric field. Similarly, it can be produced during the fast disruption shut-down proposed for ITER using high-Z impurity pellet injection. The presence of runaway current can modify the disruption dynamics and pose a concern for very localized deposition of heat on the first wall. While this is a major issue for ITER, it should be noted that since the knock-on electron avalanche multiplication mechanism predicted to be responsible for this conversion is important only for plasma currents ε10 MA, significant runaway conversion does not usually occur in present "low-current" disruptions in existing tokamaks. A model which can quantitatively predict runaway electron production under realistic disruption conditions is essential for ITER design. A theory which accounts for knock-on avalanche effects has evolved from heuristic models, to semi-analytic calculations, to detailed Monte-Carlo and Fokker-Planck studies. Given the temperature and density evolution and assuming that the runaway electrons are confined in a flux surface, theory can calculate accurately runaway magnitude and energy spectra. However, this calculation can not yet be coupled to other theories which simulate the dynamic evolution of the 3-D MHD equilibrium and presence of fluctuations during disruption. The task of coupling all the phenomena is a challenging one because of the strong nonlinear interactions. An alternate approach of building simplified models validated by comprehensive codes would more feasibly provide answers on the time scale needed to impact the FDR. In general, the development and validation of an "integrated" disruption/VDE model with runaways should receive substantial experimental and theoretical resources in the timeliest way to address this critical design issue for ITER.

VII.D. Sawteeth in ITER

Simulations of the ITER reference design indicate that sawtooth oscillations are expected to have a mixing radius out to 65% to 70% of the plasma half-width, with central temperature swings of 35% to 45%, and a sawtooth period at least several times the energy confinement time. There is concern that such large sawteeth might have an adverse effect on thermal and fast alpha confinement, or that they might trigger plasma disruptions. However, the DDR is cautiously optimistic about the effects of sawteeth because: (i) several independent sawtooth models indicate that the sawtooth period is expected to be at least several times the energy confinement time and that there is enough time for the plasma to recover and for ignition to continue through successive sawtooth crashes if the sawtooth period is in fact this long; (ii) modeling results indicate that alpha loss during sawteeth may be small; and (iii) both JET and DIII–D discharges with low q_{95} have operated with frequent sawteeth and relatively flat central profiles without degraded confinement outside the sawtooth mixing radius. When this projected extreme sawtoothing condition is used with transport models, which assume no confinement inside the mixing radius and no degradation outside the mixing radius, the DDR points out that ignition in ITER can be achieved. However, they also note the danger of sawtooth-triggered disruptions and confinement degradation which could result from coupling of the m = 1 mode to other poloidal harmonics. These effects could be large because of the lack of flow, the strong shaping, and the large q = 1 radius. This coupling may also trigger bootstrap-driven islands, ballooning modes, and edge modes.

All of the currently available models of sawteeth are semi-empirical, and none of these models have received the extensive testing against data that has been given to some confinement models. Although several have been compared with data, the physics contained in the models differs and have led to debates over key issues such as: (i) Is the trigger purely the crossing of the linear stability boundary for the m = 1 mode, or does it involve some nonlinear physics such as the suppression of the ideal energy? (ii) Does the rapid transport in the crash come from triggering a secondary instability [1], or from a full reconnection followed by a re-reconnection to restore the q profile? What then are the magnetics profiles immediately after a crash? At the present rate of progress, these questions are not likely to be answered within the next few years. The development of a more realistic physics-based model for the sawtooth crash is needed to yield solid predictions of the alpha particle loss and the coupling to modes that might degrade confinement outside the mixing radius. Until this is done, the currently available models for fast alpha loss, which yield a safe margin, are at least plausible. Nevertheless, if the alpha loss were large during each sawtooth crash, it would be much harder for ITER to ignite. For example, since fast alpha stabilization of the sawteeth is critical in the ITER reference model, the predictions would likewise be affected. Also, the expected lack of rotation in ITER suggests possible complications that do not appear in current experiments. For example, the m = 1 might be driven by locked mode perturbations to nonlinear amplitude. This could be problematic if the mode is metastable (like the bootstrap-current-driven modes). While experiments suggest that regimes can be found with benign sawteeth or even sawtooth-free cases (e.g., for reversed shear plasmas), the viability of robust sawtooth-free regimes (or active stabilization) for ITER is still a matter of debate.

Vigorous pursuit of a clear physics-based understanding of sawteeth is strongly encouraged. Such an effort should include: (i) writing a code for finding the detailed kinetic stability boundaries of the n = 1 mode and using it in self-consistent simulations to answer Question-(i) above; (ii) measurements of the microturbulence and flows in sawteeth to help address Question-(ii) above; (iii) development and testing of improved purely empirical models of the period to provide guidance analogous to confinement scaling guidance for transport models; (iv) further development of mode-coupling theory in weakly rotating plasmas; and (v) measurements of fast ion losses during sawtooth crashes in elongated tokamaks with broad sawteeth.

References

[1] R.J. Goldston, R.E. Waltz, *et al.*, Fusion Technol. **21**, 1076 (1992).

[2] C.C. Petty, *et al.*, Phys. Plasmas **2**, 2342 (1995).

[3] C.C. Petty and T.C. Luce, Nucl. Fusion **37**, 1 (1997).

[4] J.G. Cordey, *et al.*, Plasma Phys. Control. Fusion **38**, A67 (1996).

[5] R.E. Waltz, *et al.*, Plasma Phys. and Contr. Nucl. Fusion Research, IAEA, Vienna, 1996, paper D1-6.

[6] Kotschenreuther, *et al.*, "First principles calculation of tokamak energy transport," 16th IAEA (Montreal, 1996) IAEA-CN-64/D1-5.

[7] J.W. Connor, *et al.*, Plasma Phys. and Control. Nucl. Fusion Research, IAEA, Vienna, 1996, paper FP-21.

[8] J. Kinsey, *et al.*, Plasma Phys. and Contr. Nucl. Fusion Research, IAEA, Vienna, 1996, paper DP-7.

[9] H. Nordman and J. Weiland, Nucl. Fusion **32**, 151 (1992).

[10] Igitkhanov, *et al.*, “ITER operational space for density limits and H-mode operation,” ITER Memo G 19 MD 1 96-12-12 W0.1 and S 17 MD 1 96-12-12 F1, (12/9/96).

[11] Kamada, *et al.*, “High triangularity discharges with improved stability and confinement in JT–60U,” 16th IAEA (Montreal, 1996) IAEA-CN-64/A 1-6.

[12] Lomas and JET Team, “High fusion performance ELM-free H-modes and the approach to steady operation,” 16th IAEA (Montreal, 1996) IAEA-CN-64/A 1-5.

[13] Osborne, APS 1996 and (unpublished 1997).

[14] C.C. Petty, *et al.*, 1996 (unpublished)?

[15] Yushmanov, *et al.*, APS 1996.

[16] P. Diamond (January, 1997) (unpublished).

[17] M. Greenwald, *et al.*, Nucl. Fusion **28**, 2199 (1988).

[18] B. Lipschultz, *et al.*, J. Nucl. Mater. **220-222**, 50 (1995).

[19] R. Maingi, *et al.*, to be published in Phys. Plasmas 5/97; also GA-A22521.

[20] M. Rosenbluth, *et al.*, 1994 IAEA.

APPENDIX D.II

SUB-PANEL II:
HEAT FLUX COMPONENTS, FUEL CYCLE

Dr. Earl S. Marmar (Chairman)*
Massachusetts Institute of Technology

Prof. Ira B. Bernstein*
Yale University

Dr. Bastiaan J. Braams
New York University
Courant Institute of Mathematics

Dr. Katherine B. Gebbie *
NIST

Dr. John R. Haines
Oak Ridge National Laboratory

Dr. Dave Hill
Lawrence Livermore National Laboratory

Dr. Charles Karney
Princeton Plasma Physics Laboratory

Dr. Bruce Lipschultz
Massachusetts Institute of Technology
Plasma Fusion Center

Dr. Stanley Luckhardt
University of California, San Diego

Dr. Peter Mioduszewski
Oak Ridge National Laboratory

Dr. Gary Porter
Lawrence Livermore National Laboratory

Dr. Charles Skinner
Princeton Plasma Physics Laboratory

* FESAC Member

Dissipative Divertor

The proposal by ITER is to utilize a vertical-plate geometry divertor and to have an operational mode with a partially detached divertor (PDD). This leads to reductions in peak power and particle flows by factors of 5-10, which may allow for reasonable engineering and physics design goals (steady state wall loading below 5 MW/m^2 and core Z_{eff} below 1.8). Partial detachment leaves the remaining attached flux surfaces farther away from the separatrix to provide the needed neutral baffling in the divertor. Neon (or possibly other gases) are injected to enhance divertor radiation and detachment. A Ne concentration of ~.5%, leads to 100 MW of radiation. There is reasonable experimental evidence and 2-D fluid modeling to support most of this proposal. In particular, all divertor tokamaks have achieved some version of detachment. Vertical-plate geometry not only lowers the detachment threshold density but results in partial detachment (Alcator C-Mod). Experiments at DIII-D and ASDEX-U both show impurities collecting in the divertor (compression), in a manner similar to that seen for deuterium, indicating that injected impurities will be primarily contained there. One danger of this mode of operation is the potential for transition to full detachment, whereby the cold plasma region would expand to fill the whole divertor chamber, and the baffling would be lost. Control would be achieved by strong pumping in the divertor in conjunction with an adjustable gas feed, but we don't believe that either experiment or modeling has yet shown satisfactory control of detached operation. The present modeling results indicate that this control problem will be difficult in a quiescent plasma, and, in the presence of ELMs, such control may be impossible.

To date, the experimental results are mixed. On the positive side, in DIII-D and ASDEX-U, the amount of injected impurity required for detachment is low, leading to a small increase in core Z_{eff}. In contrast, results from Alcator C-Mod and JET find that the required impurity injection rates lead to unacceptably large core impurity levels. The causes of these differences are under investigation. In the meantime, we cannot reliably predict the impurity level inside the separatrix of ITER. A related issue is the effect of radiation inside the last closed flux surface on core energy confinement. In JET, Alcator C-Mod and JT-60U, the energy confinement begins to degrade significantly when P_{rad}/P_{tot} exceeds about 0.5.

Experiments at C-Mod and JET, directed at understanding the effects of divertor geometry on detachment, have successfully shown that the vertical plate geometry lowers the detachment threshold and aids in PDD operation. DIII-D, ASDEX-U and JT-60U are presently changing their divertor geometries, in order to continue these studies.

Experiments on all the devices are being expanded in scope to include the effects of geometry on impurity transport, detachment control and the role of neutrals in producing the PDD state.

Another important issue is the role played by intrinsic impurities. Be, C and W have been chosen for the first wall, divertor entrance and highest heat flux divertor regions, respectively, and the resulting levels of these impurities in the divertor and core plasmas are open questions. If these levels are sufficiently high, injected impurities may not be needed for increasing P_{rad} and the effects of intrinsic impurities on the core confinement and Z_{eff} become the issues. The level of intrinsic impurities cannot be externally controlled and impurity levels could get too high. Perhaps most importantly, intrinsic impurities do not allow for feedback control of the detachment characteristics, which is required for the stability of the partial detachment.

Another issue relating to the interaction of the core and divertor plasmas, but in the other direction, is the effect of ELMs. The present assumptions for ITER are somewhat conservative in this area, namely that type 1 ELM's will be dominant. Based on data from ASDEX-U, DIII-D and JET, an empirical scaling has been derived, which indicates that the fractional loss of stored energy, $\delta W/W$, for each ELM, should be ~1.5%. This is marginally acceptable from an erosion standpoint. Further data are needed with respect to $\delta W/W$, ELM frequency and the divertor profile of the heat load, particularly under operating conditions at or near the density limit, with separatrix power flux just above the H-Mode threshold.

We expect significant experimental progress in these areas over the next 1-2 years, which, when coupled to progress in understanding impurity entrainment, may increase our confidence that a radiative divertor solution, compatible with the core, is possible. In terms of density and power, the existing database would say that operation near the Greenwald limit with a vertical target will almost certainly guarantee achieving the PDD condition and low target plate heat flux, as long as there is sufficient impurity density in the divertor. As to the question of whether good confinement can be maintained at such high densities, there is still considerable uncertainty; however, by the end of the EDA there should be more progress on the confinement issues related to density and radiation.

While the design presented in the DDR is credible, we are not convinced that it is well optimized. We agree with the proposal to take advantage of partially detached operation, but feel that this scenario requires more modeling attention by the ITER team. One aim should be to optimize an alternative divertor geometry for partially attached high-recycling operation in conjunction with a radiating mantle. We expect that this optimization might result, for example, in a pumping location on the outside of the divertor leg, and possibly also in a reduction of the distance from x-point to the strike-point. The present JCT position on helium exhaust is that the problem is solved, since experiments measuring helium transport show that core transport in ELMing H-mode should be adequate to flush the helium ash from the core. However, 2-D SOL modeling for ITER suggests that pumping from the private flux side of the divertor will be inefficient for helium exhaust, and so it is not clear that sufficient helium exhaust can be obtained with the present geometry. On the outside, the plasma and neutral densities are substantially higher than on the inside. A pump location on the outside, in addition to improved helium removal, should also provide the fastest hydrogenic and impurity pumping, both of which appear to be required for controlling a radiating mantle solution. More experiments on existing machines are needed to determine if the 2-D effects seen in the models really exist; it is not clear that this topic will receive much attention by the end of the EDA.

At this time we cannot rely on a radiative mantle/shallow divertor concept, mainly because of the possible effects of the radiating mantle on core energy confinement as well as the uncertainty in the H-mode power threshold. Demonstrations on diverted, high density H-Mode plasmas are also required. Therefore, the statement in the 1995 TAC-8 report (Section 4.1.3), that the present amount of space for the divertor should be retained, remains valid. Nevertheless, we recommend that the engineering implications of a shallower divertor (e.g., 1.0m between x-point and plate instead of 2.0m) be explored before the end of the EDA. We would naturally want to reap the benefits of a shallower divertor by enlarging the plasma volume, raising the current, and perhaps increasing the triangularity. It is not clear that the present poloidal field system is suitable to control such modified configurations. A related question is: can we anticipate the possibility of modifying the divertor cassettes to handle a shallower divertor and larger plasma without having to replace the first wall and shield as well? These issues should also be addressed in the final design report.

The present ITER divertor design presents one concept for the divertor configuration required to achieve low target plate peak heat flux simultaneously with high confinement and low Z_{eff}. It may not be optimum for an ITER-sized machine. By the end of the EDA, present experiments will have explored operation with similar divertor configurations. These experiments will show whether such a deep divertor offers significant advantages (e.g., better impurity entrainment and greater heat flux reductions with lower Z_{eff}) over more compact and/or more open designs, and may point out unforeseen adverse effects.

Comparison of Modeling and Experiment

The ITER divertor design is necessarily based, in large part, on modeling results, and the feasibility of the design relies heavily on those results, which come primarily from 2-D plasma fluid simulations. It is thus very important to demonstrate the viability of the physics in the models which are being used. One source of confidence in the results of the codes derives from their successful use in modeling existing experimental results. The B2/EIRENE code has been compared extensively with results from the ASDEX-U experiment; the EDGE2D/NIMBUS code with results from JET; and the UEDGE code with results from DIII-D and C-MOD. The modeling successfully reproduces many of the experimentally observed phenomena in each case. The ITER team is to be commended for their efforts to urge the divertor community to apply these modeling codes to their experimental results.

One of the principle unknowns in all of these experiment/modeling comparisons relates to the perpendicular diffusion coefficients. All models assume this diffusion to be anomalous. There has been some work assuming the diffusion coefficients scale as Bohm, and a little work with more theoretically based transport coefficients. The simplest models assume the particle diffusivity, electron thermal diffusivity, and ion thermal diffusivity are each spatially constant throughout the calculational domain. The value of the diffusivity is determined by matching to some experimentally measured radial profile. This simple model is remarkably successful in reproducing experimental results. In some cases, a particle pinch is introduced to simulate narrow divertor ion current profiles. The scalings of these coefficients have not been adequately studied. The uncertainty in the value of these diffusivities is very worrisome for the ITER design, and it is important that the ITER team continue to pursue these experimental comparisons to obtain a better idea of scaling and to demonstrate the sensitivity of their design to the choices for these uncertain parameters.

While significant effort has already gone into the 2-D modeling of the edge, along with extensive 1-D modeling of the core transport, relatively little attention has been paid, so far, to the synergistic effects at the interface between the core and the scrape-off layer. Critical issues related to the interactions of the H-mode threshold and edge barrier transport with the dissipative divertor dynamics are intimately tied to this interface region. We believe that significant effort to model these effects, along with detailed comparisons to experimental results, will be required as part of the overall divertor optimization process.

Engineering Issues

The ITER divertor design and R&D team has made great progress on addressing engineering related issues during the EDA. They have identified the most critical items, defined an R&D plan, and conducted, or are in the process of conducting, most of the R&D tasks. Prototypes of all elements of the divertor cassette will be fabricated prior to the completion of the EDA, so that manufacturability issues should be adequately addressed. Similarly, remote handling of full-scale, prototypical divertor cassettes will be examined during these final stages of the EDA.

One unresolved engineering concern relates to the use of beryllium and tungsten for the plasma facing components. As has been recognized in the DDD, there have been difficulties in finding a braze technique for attaching either Be or W to the required copper substrate that will survive in the ITER radiation environment. This problem is under investigation in the current R&D, and some radiation testing results should be available before the end of the EDA.

The remaining issues that are of critical importance are related to integrated, synergistic, or large-scale effects. For example, high heat flux testing of an entire divertor cassette using irradiated materials would be very desirable. Another example of an activity that would be extremely useful is refurbishment of a cassette that has been previously irradiated to establish whether swelling or the presence of helium seriously impacts the ability to transport the cassettes, make and break mechanical attachments, or reweld coolant system connections. While it is clear that these tests would greatly increase the confidence in the selected design approaches, the costs are likely to be prohibitive. For this reason and because of the uncertainties associated with the lack of understanding of the plasma edge physics and its coupling to the bulk plasma, it is essential that the divertor be easily handled and that ITER be designed with the flexibility to accommodate different divertor configurations. The ITER design team has addressed this issue by (1) ensuring that the divertor can be replaced frequently (eight divertor changeouts over the lifetime of ITER) and quickly (changeout of all 60 divertor cassettes is designed to take about 50 days), and (2) providing a large volume to accommodate future, i.e.~unknown, divertor configurations.

The rest of the outstanding engineering concerns regarding the divertor are listed below. It is possible that most, if not all, of these issues have already been adequately addressed by the project, but this panel, with its limited documentation and interaction with the design team, has been unable to obtain satisfactory resolution. Refurbishment (for example, rewelding of previously irradiated coolant connections) of the divertor cassettes in a hot cell appears to be a very difficult task, especially given the alignment/positioning requirements involved in the target assemblies. With the recognized erosion problems, this refurbishment will be required; the DDD goes into some detail on how this will be done, but questions remain on the time scale required. Also, is this process cost effective compared to disposal of irradiated cassettes and replacement with new ones? There appears to be a disconnect between the physics studies aimed at determining the peak heat flux on the divertor and the design requirement for the peak heat flux. The plasma edge modeling efforts are focused on achieving a toroidally-averaged peak heat flux of 5 MW/m^2. The engineering design and R&D program are based on a requirement to handle an absolute steady-state peak heat flux of 5 MW/m^2, with transient excursions up to 20 MW/m^2. Given that the modeling efforts have yet to make a completely compelling case for achieving even the toroidally averaged peak heat flux value of 5 MW/m^2, it seems reasonable to ask whether or not the design requirement can be increased. This does not relieve the pressure on the physics community to find improved divertor operating regimes, but does place an added, but realistic and reasonable, burden on the engineering design and R&D programs.

Particle Control: Density Control, Fueling, Pumping

The nominal gas throughput can best be derived from the helium exhaust requirements. The total fusion power of 1.5 GW will produce helium ash at the rate of 2 Pa $\supseteq$ m^3/s. In steady state, this is the required helium exhaust rate. The nominal helium concentration in the core plasma is 10%. The recommended “enrichment" factor in the divertor is 0.2, i.e. the concentration in the divertor is assumed to be 2%. With a required helium exhaust of 2 Pa $\supseteq$ m^3/s, the corresponding nominal DT exhaust is 50 Pa $\supseteq$ m^3/s, given by the helium production and divertor enrichment factor. The nominal divertor pumping speed is 200 m^3/s. At the exhaust rates stated above, this translates into a helium partial pressure of 0.01 Pa and a DT partial pressure of 0.25 Pa. Pressures of this magnitude have been observed in today's machines, but extrapolation to ITER is not straightforward, because the pressure not only scales with fluxes and pumping speeds, but also with the details of divertor and pumping duct geometry. Based on contemporary experience and the results of computer modeling, the assumed pumping parameters appear plausible.

The gas puffing system has some flexibility, provided by the divertor and main chamber locations as well as by the toroidally distributed gas valve system, and by the available fueling rate of 200 Pa $\supseteq$ m^3/s, which is a factor of four larger than the nominally required steady state rate. During density ramp-up, the rapid build-up of the plasma inventory, as well as transient wall pumping, may require up to 500 Pa ⊇ m^3/s, according to the DDR. Since a maximum of 200 Pa ⊇ m^3/s is available for fueling, this issue needs to be addressed. Due to the large plasma size and high edge densities, the gas fueling efficiency is likely to be very low. To avoid very high gas puffing rates, core fueling by pellet injection with two centrifuge injectors is foreseen. Continuous pellet fueling of 50 Pa ⊇ m^3/s for T and 100 Pa ⊇ m^3/s for DT seems to be adequate for a tailored fueling scenario that minimizes the tritium wall inventory. Impurity gas puffing is foreseen at rates of 10 Pa ⊇ m^3/s for N_2, Ne, Ar, and Kr. Actual impurity puffing rates, necessary to lower the divertor power to 150 MW or less through edge and divertor radiation, are as low as 0.25 Pa ⊇ m^3/s. With a divertor pumping speed of 200 Pa ⊇ m^3/s, this rate results in a steady state impurity density of 3.3×10^{17} m^{-3} corresponding to an impurity concentration of 0.5% in the SOL. Within the existing uncertainties of the atomic data, this is sufficient neon, as calculated with B2-EIRENE, to reduce the divertor power flux to 5 MW/m^2. The missing part is a self-consistent treatment including intrinsic impurities such as carbon, which has been observed to be the main radiator in many present experiments.

The effective pumping speed is quoted as 200 m^3/s. We assume that this pumping speed is independent of the pumped species, applying also to helium. Differential pumping is not currently planned: it might be possible first to remove the hydrogenic species by cryo-condensation panels, with the helium subsequently pumped by cryo-adsorption panels. This scheme would minimize the required active charcoal in the divertor pumps and may give added flexibility for density control. In present devices it also has been found beneficial to have at least a small pumping capacity of turbomolecular pumping available for recovery of a choked cryopump etc.

Density control is of crucial importance, because it is the main means for active feedback control of the fusion power. The density response of the plasma is controlled by two factors: the particle confinement time τ_p, and the global recycling R of the system. In order to achieve fast density control, (1) particle transport to the plasma edge has to be sufficiently fast, and (2) the global recycling has to be low enough to remove particles from the edge at sufficient rates. The density response time is usually given by the global particle containment time, $\tau_p^*=\tau_p/(1-R)$. The particle confinement time is assumed to be about twice the energy confinement time, i.e. τ_p ~12s, and global recycling has been shown experimentally to be reduced by divertor pumping to values no smaller than about R=0.9. The resulting value for the global particle containment time is τ_p^*=120s. This time constant appears to be too long for effective density control. Assuming constant particle confinement time, an effective control of the global recycling is needed to control the density. This needs to be demonstrated by computer modeling and subsequent validation on present divertor machines.

In steady state, the fueling and pumping rates are identical, and equal to 50 Pa ⊇ m^3/s. The neutral gas leakage rate from the divertor to the main chamber has been specified to be 30 Pa ⊇ m^3/s. This is a source rate which is almost as large as the external fueling rate and, since it is not under external control, may have a substantial effect on density control. It also seems to defeat the efforts for strong neutral confinement in the divertor. It should be demonstrated, by computer modeling, whether the midplane neutral pressure resulting from this leak rate is tolerable.

It has been shown in DIII-D that forced flow of fuel particles in the SOL and divertor, generated by strong gas puffing, can improve the impurity retention in the divertor. To take advantage of this effect, the ITER gas puffing capability has been increased to 200 Pa ⊇ m^3/s, but it has been argued that this might not be sufficient. However, a major increase of this gas puffing rate might not be justified at this stage for the following reasons: (a) it is not obvious how the gas puffing rate scales from the 20 Pa⊇m^3/s used on DIII-D to that required for ITER, (b) in ASDEX-U, JET and Alcator C-Mod, impurity entrainment has been observed as being due to local recycling fluxes without strong external gas puffing, (c) the present limit of 200 Pa ⊇ m^3/s is determined by the tritium processing plant and can not be raised by simply increasing the gas puffing rate. Therefore it should be considered sufficient until clear evidence is available that higher rates are needed.

Tritium Retention, Erosion and Dust

Tritium fueling, retention, and removal, together with erosion and dust removal are ‘housekeeping’ issues fundamental to the development of magnetic fusion as a safe, environmentally sound energy source. Much has been learned about these issues. However, for ITER, the constraints of available tritium supply and safety limits in allowable tritium inventory, together with high predicted erosion rates in partially detached divertor operation, pose severe engineering challenges. These problems are recognized in both the DDR and DDD documents, but, at the moment, clear paths to their solutions are not always apparent.

Co-deposition of tritium in carbon plasma facing components (PFC’s) is a serious issue and could severely limit the operation of ITER. This problem is particularly acute in detached or semi-detached plasma operation, where up to 20g of tritium may be co-deposited in a single pulse and the 1 kg inventory limit may be reached in only 50 pulses. Tritium retention in beryllium may be lower than previously feared. Recent results from the US Home Team have indicated a saturation effect at ITER-like conditions; this needs to be confirmed. Carbon and oxygen impurities also affect the level of retention in beryllium, and predictions for ITER are dependent on knowledge of the concentration of impurities in the ITER plasma. Predictive models of tritium retention in tokamaks need to be further developed and tested against the experience of existing tokamaks. We also recommend that gas balance studies be carried out on operating tokamaks to extend the database on deuterium retention.

Erosion of PFC's by sputtering, disruptions and slow transients is a major issue for ITER. Sputtering erosion rates 10x higher than previously foreseen are predicted by recent calculations by the US Home team that have been benchmarked with DIII-D measurements. In the resulting scenario, most of the 4 cm thickness of the carbon fiber composite divertor target plate in high heat flux regions would be eroded over 3,000 pulses and close to 1,000 kg of carbon converted mostly to atomic C, C_3 and CH_4. This carbon would primarily be co-deposited with tritium, rapidly raising the tritium inventory and creating tritiated, radioactive, chemically active and/or toxic dust and flakes that may accumulate in regions that are difficult to access. The use of silicon doping can be used to reduce chemical sputtering, but physical sputtering alone would remain as a serious problem. We recommend that further measurements of erosion be undertaken on an existing tokamak, for conditions as close as possible to the ITER PDD operational mode.

The in-vessel tritium inventory limit of 1 kg is a very small fraction of the several hundred kg total fueling over the lifetime of a divertor cassette. Minimal tritium retention is thus required, together with the development of efficient and rapid tritium removal techniques that are orders of magnitude beyond the experience of current tokamaks. Assurance that the in-vessel inventory is below the safety limit will also depend on the development of new ways to detect tritium buried in carbon flakes and dust in difficult-to-access areas inside the vessel.

CFC materials were introduced as divertor target plates when it was recognized that slow transients could impose heat loads up to 20 MW/m^2 for up to 10 s. However this choice will only be practical if solutions are found to difficult issues in the areas of tritium retention, removal, erosion and in-vessel tritium diagnostics, issues that become more acute in detached or semi-detached divertor operation. Given these difficulties, it is essential to minimize the use of carbon in ITER. ITER tasks T226 and T227 have been set up to coordinate efforts in the US Home Team to address these issues. Validation of the proposed solutions, by bench-marking with experience on existing tokamaks, is essential.

Wall Conditioning, Baking and Discharge Cleaning

The DDR specifies that the first wall components in ITER can be baked to 240C. This may slow initial conditioning after vents, and perhaps of more importance, disruption recovery may be particularly troublesome, (as in TFTR and Tore Supra) due to the presence of graphite in the device. Also, because glow discharge cleaning will not be available between shots, because of the need to turn off the toroidal field, this cannot be used to aid in post-disruption recovery of the wall conditioning. While it is true that the fraction of the wall covered by graphite is small, data from ASDEX-U and experience from Alcator C-Mod clearly show that carbon migrates to all places in the machine, so that much of the W and Be on the PFC's will be coated with carbon. If we knew at what temperature the disruption recovery becomes easy, then we could confirm the DDR design point, but a study of how disruption recovery depends on bakeout temperature is not likely to happen before the end of the EDA, if ever. It is well known (and described in the DDD) that 240C is inadequate for water removal from CFC. An increase of the bakeout temperature capability, up to perhaps 350C, would probably require excessive water pressure, or a switch to steam coolant; however, if CFC plasma facing components remain in the design, this option should be re-examined. In parallel, an experimental study of post disruption cleanup using ECRDC on a tokamak using carbon PFC's would be very useful.

Choice of Plasma Facing Materials

In the present design, CFC is chosen for the highest heat flux regions of the divertor, with tungsten used on the rest of the divertor surfaces and beryllium on the remainder of the plasma facing components. The choice of carbon, in particular, poses many additional difficulties, including tritium codeposition and retention, dust, insufficient bake temperature and related conditioning issues, and possibly impurity radiation. A potential solution would be to eliminate carbon altogether, using tungsten (or perhaps some other sputter resistant metal) for the entire divertor surface, and we recommend that the project seriously revisit this option.

APPENDIX D.III

SUB-PANEL III: REPORT ON DISRUPTIONS/VDES AND BLANKET/SHIELD ATTACHMENT

Dr. S.J. Zweben (Chairman) *
Princeton Plasma Physics Laboratory

Dr. Robert S. Granetz
Massachusetts Institute of Technology

Prof. Hans Fleischmann
Cornell University

Dr. Arnold Kellman
General Atomics

Dr. Eric Fredrickson
Plasma Physics Laboratory

Dr. George Sheffield
Princeton Plasma Physics Laboratory

* - FESAC Member

Summary of Findings and Recommendations:

This sub-panel considered the effects of plasma disruptions on the ITER first wall and divertor structures. Because there is a large uncertainty in the frequency and severity of disruptions in ITER, their effects may range from benign (gradual erosion of the divertor targets) to catastrophic (detachment of a blanket/shield module from the backplate). The physical understanding of the dynamics of disruptions and their enormous thermal, electromagnetic and mechanical transients is incomplete and evolving. The ITER engineering team has tried to keep up with the changing physics and remote handling requirements, but the present bolted blanket/shield design is marginal for the one detailed EM load case analyzed so far. The main recommendations of this sub-panel are for:

a) improvement of the bolted blanket/shield design until it can be shown to be able to withstand at least 500 full-power, full-current disruptions of various kinds, including the effects of VDE's, non-axisymmetries, halo currents, and post-disruption runaway electron impacts

b) continued research into the physics of tokamak disruptions, including experiments on the thermal and electromagnetic transients, modeling of the plasma and runaway beam dynamics, and studies of the plasma-surface interactions at high heat flux

c) increased efforts to develop and test disruption avoidance and mitigation techniques based on a closer interaction between tokamak experimentalists, theorists, and ITER engineers.

1. Disruption Frequency (Stewart Zweben)

The DDR specifies a design guideline of 500 full current, full-power disruptions (followed by VDEs) for the 11,000 pulse (10 year) Basic Performance Phase (DDR Table 4.2-2). In addition, a larger number of partial- current and partial-power disruptions are anticipated, leading to a total disruption frequency design guideline of 30% for the BPP. However, the disruption frequency is assumed to be only ~10% for the 5000-shot full-performance phase of the BPP, and ~ 3% for the later EPP (extended performance phase).

This design guideline in the DDR is roughly consistent with empirical statistics on disruption frequencies in present tokamaks, as reviewed by the ITER Expert Group in this field [1-2]. However, it is well known that this frequency is not a constant of Nature, but depends crucially upon how close the plasmas are to their beta, density, and q(a) limits. Therefore it is not clear whether these empirical statistics are a reliable guideline to predict the disruptivity of full-performance ITER plasmas which will be challenging all of these limits simultaneously. There is even less of a theoretical basis for such a prediction.

Another shortcoming of this design guideline is that it apparently does not take into account the factors which make ITER *different* from present experiments, such as its very long pulses and the relatively uncontrolled nature of alpha heating near ignition. For example, the disruption frequency in Alcator C-Mod is ~ 1 sec^{-1} at q(a) ~ 3, which could imply that *every* ITER discharge will disrupt. Thus it is not yet clear how the present experimental results can be scaled to the 1000 sec ITER pulse lengths. It is also possible that the dynamics of alpha-heated profiles at ignition might naturally evolve toward some MHD-unstable state, potentially leading to a higher disruption frequency than present externally-heated experiments.

Given this large uncertainty, it seems highly desirable to extend the engineering limits on the allowable number of full-power, full-current disruptions in ITER. There seem to be at least three such limits: the erosion of the divertor target plate by the high heat flux during a thermal quench, the mechanical strain on the blanket/shield module attachments due to the current quench, and the possible damage to the first wall due to a disruption-induced runaway electron energy (see other sections of this sub-panel report for details). However, the EDA documents do *not* yet present sufficient analysis to define how many full disruptions the machine can handle before a first-wall or divertor module needs to be replaced (although some modeling of the divertor erosion has been made).

Therefore, we suggest that future ITER EDA activities in this area focus on efforts to better define and improve the engineering limits for the allowable number of disruptions, and to encourage new experiments and modeling on disruption avoidance and mitigation techniques. Some potential "fast plasma shutdown" schemes have already been suggested by the ITER team [2, 3], but they need further experimental testing and modeling before being adopted in the ITER design.

It should be noted that the ITER team has done a good job on several of the engineering systems related to the effects of disruptions. The replaceable divertor cassettes seem to be very well designed, the in-vessel metrology system should be able to monitor disruption-induced damage, and the safety systems have been designed to handle an accidental break of a water cooling tube in the first-wall or divertor region, which is a potential failure mode due to a major disruption. However, the consequences of such a failure are still severe, i.e. a 1-2 month downtime for replacement of the leaking module.

2. Halo Currents and Non-Axisymmetries (Bob Granetz)

Partly in response to ITER concerns, a large dataset on halo current characteristics from disruptions in many divertor machines has been compiled over the last two years [2]. This dataset defines the envelope and distribution of I_{halo}/I_{p0} and toroidal peaking factors (TPF's), and it is being factored into the engineering constraints for the ITER backplate and blanket modules. The range of I_{halo}/I_{p0} seen at q_{95}=3 and k=1.6 is roughly 0.1 to 0.3, with most of the uncertainty arising from machine-to-machine variation in the experimental dataset (not understood at all), as well as lack of a detailed understanding of the physics of halo currents. The TPF's seem to scale generally in an inverse relationship with I_{halo}/I_{p0}, with relevant envelope limits somewhere in the range of $0.50 < \text{TPF} \bullet I_{halo}/I_{p0} < 0.75$.

Exactly how to apply this data to the ITER design is not clear. The ITER projections given in the DDR use the data by normalizing it into dimensionless parameters (I_{halo}/I_{p0}, for example, but nothing involving machine size), which is all we know how to do at this time. The previous design, which called for welded attachment of the blanket modules, was able to accommodate halo current amplitudes and asymmetries typical of the mean of the experimental dataset ($\text{TPF} \bullet I_{halo}/I_p = 0.50$) (TAC-11 Report, sec. 3.1.3) . However, taking the worst-case limits ($\text{TPF} \bullet I_{halo}/I_p = 0.75$) and translating them directly to ITER parameters led to local stresses which probably could not have been accommodated with the welded design. A new bolted attachment design for the blanket/shield modules has recently replaced the welded design. However, as of this time (3/97), the bolted design has only been analyzed for a fast radial disruption, so the potential effects of halo currents and asymmetries have not yet been included. But are the worst-case scenarios from the dimensionless dataset even relevant to ITER? It is not known how well the observations in present-day machines predict the expected values in ITER, because we do not have a good physical understanding for the variation seen in the toroidal asymmetry or in the halo current fraction. In particular, the fact that the large machines (JET and JT-60U) generally see lower halo current fractions than smaller elongated tokamaks hints that there may be a favorable size scaling in the dataset. As an aside, the present dataset on halo currents would be even more useful if it had data on I_{halo}/I_{p0} from JT-60U at relevant elongation, and TPF's from JET and JT-60U.

The US Home Team has proposed trying to get a better understanding of the expected halo currents and asymmetries in ITER through a major modeling effort, benchmarked by well documented disruption data from DIII-D and C-MOD (a medium size and a small size tokamak respectively). In addition, some theoretical progress has already been made in explaining the toroidal asymmetries, including the most general features of the TPF scalings in the dataset [4], and this work should be continued.

Even with continued work on present machines, the resolution of these worst-case issues may have to await actual ITER operation. Hopefully, sufficient experience will be obtained during the hydrogen phase of the BPP (possibly at reduced plasma current) to allow for identification of the areas of operational parameter space which lead to unacceptable disruption loads, if any. It is imperative, of course, that ITER diagnostics include sufficient instrumentation to measure halo currents and toroidal asymmetries. ITER has specified that the ground strap currents in a number (10) of divertor modules will be measured. This will give good information on the toroidal asymmetry of the bulk of the halo current (the remainder flows in the vessel wall). Consideration has been given to measuring the poloidal distribution by instrumenting the ground straps of a number of the new insulated blanket modules, but nothing definite has been decided.

Finally, one comment concerning disruption mitigation scenarios (i.e. very fast plasma shutdown) should be made. A lot of the difficult problems handling disruptions in ITER may be helped by killer pellets or other fast quenching actions. As far as halo currents are concerned, recent killer pellet experiments indicate that although these actions can reduce or eliminate halo currents in the divertor region (because the plasma dies away before it gets to the divertor), they may not help at the inboard midplane. Data from Alcator C-MOD show that midplane disruptions also have halo currents, with normalized magnitudes comparable to VDE disruptions [5]. Since fast termination will result in a lot of quenches at the midplane, it would be desirable to be able to measure halo currents there.

3. Disruption Effects in the Divertor (Arnie Kellman)

The major effects of disruptions on the divertor hardware involve the forces and heat flux and resulting erosion or melting of the divertor components. Considerable work is being done by both the central team and the home teams to address the issues; however, large uncertainties remain on some critical topics, especially in the area of PFC lifetime.

The present divertor design has vertical target plates, a dome in the private flux region and wings on the private flux side of the divertor legs. The tentative material choices (final choices will depend on successful results of ongoing tests) are CFC (40 mm thick) for the lower vertical target plate and dump plate, and W for the dome, upper vertical target, wings, and first wall coating on the baffles.

There is considerable uncertainty in the calculated lifetime of the divertor plate, although much of that uncertainty results from sputtering yields during steady state (SS) ELMing operation. In all models, a 10% disruptivity is assumed with a heat deposition to the divertor of 100 MJ/m^2, which is a conservative estimate. Calculations indicate an erosion loss of 30 micron per disruption. Coupled with one of the sputtering models, the expected lifetime of the divertor plate is 5800-8200 shots, with disruptions accounting for 18-25 of the 38 mm target loss. Based on these estimates it appears that ITER can meet its objective of surviving in excess of a few thousand plasmas before divertor target plate replacement. However, there are many reasons for the uncertainties in this lifetime estimate.

First, we are depending on the existence of a vapor shield to reduce disruption erosion. Vapor shields are seen in disruption simulation experiments with plasma guns, but have not been observed in present-day tokamaks because the disruptive heat flux is too low. The vapor shield models for CFC do not all go to 100 MJ/m^2 so require extrapolation, and the different models have factor of 2-4 differences. Similarly, simulation experiments do not extend to 100 MJ/m^2 and pulse durations are generally less than 1 ms, with longer disruptions not yet examined..

Second, experiments on plasma gun disruption simulators show that if the plates are tilted (as in ITER) with respect to the magnetic field, the erosion increases significantly, but the modeling does not quantitatively match the experiment yet. The modeling does show however, that radiation by the vapor shield away from the target plates will be large enough to affect nearby elements.

Third, the survivability of W during disruptions is questionable because of possible melting and melt layer loss, but again there are no experimental results in tokamak disruptions. In addition, no model can yet predict how much of the melt layer will be lost. Thus, the lifetimes of W under high heat flux is highly uncertain if melting occurs.

Fourth, the characteristics of the disruptive SOL plasma have large uncertainties and no models exist to extrapolate to ITER heat flux levels. ITER specifications are that the SOL broadens by 3 times during a disruption, but the peak heat flux location remains unchanged. However, data from DIII-D and JT-60 show that during disruptions, the heat flux peaks can move significantly away from the nominal separatrix location. This may affect whether the baffle or dome location receive heat fluxes sufficient to melt the tungsten.

An additional uncertainty is related to the side and back heating of the other divertor components from the intense radiation of the vapor. While this effect is known to the ITER group, its effect is still being evaluated. Finally, there are still uncertainties in the choice of the materials (especially W on baffle and dome) pending test results on thermal stress, fatigue, melt layer loss, cracking, and embrittlement.

In the TAC-JCT Informal Technical Review (5/95), they identified a rather comprehensive set of outstanding issues relating both to erosion, surface contamination during disruptions, and mechanical load calculations. Given sufficient support for the simulation effort, it is likely that all mechanical issues related to eddy/halo current forces can be resolved. On the other hand, it is still too early to say whether the current optimism for the use of CFC or W will be supported by further testing and experimental results.

There is an urgent need for disruption simulation experiments with higher energies and longer durations (up to 100 MJ/m^2 and variable durations in excess of 1-10 msec) All experiments should be performed with a strong and inclined magnetic field. In addition, there is a strong need for both modeling of melt layer and for tokamak experiments to test melt layer and erosion models.

The design of the baffle must take into account possible disruptive heat loads due to wider than expected SOL or peak heat fluxes away from the SS strike points. Additional physics input on disruptive heat flux patterns should be sought. In addition, both the baffles and divertor dome should be designed to survive runaway electron impact. Even the small plasmas that would contact the dome can have runaway electrons remaining.

A full set of TSC scoping studies must be performed to allow self-consistent force calculations on the divertor components to be done including halo currents. The time dependent effects on the halo currents must be included in the dynamical analysis of the various divertor components. A viable plan must also be developed for a between shot cleaning method that will permit adequate recovery from disruptions.

4. Disruption-Induced Runaway Electrons (Hans Fleischmann)

Concerning the effects of runaway electrons (RA) during disruptions, sizable uncertainties still exist at present. The generation of strong runaway beams - with particle energies in the 10's MeV region - has been reported (and somewhat investigated) for disruptions in a number of large tokamaks, in particular JET, TFTR and Tore-Supra [6]. Also significant is that strongly localized wall damage has been ascribed to runaway in some cases. Present theoretical models [3,7] - not yet well confirmed (nor opposed) experimentally, but grounded on very basic facts and assumptions - indicate a strong possibility that in ITER disruptions such runaway beams will be strongly enhanced through collisional avalanching - with enhancement factors of up to 10 - 15 orders of magnitude. Thus, even very small densities of seed runaways existing or generated at the start of the current quench phase could to lead to runaway beams carrying a large fraction of the initial discharge current. In this case, it might be expected that a rather large percentage of disruptions will show strong runaway beams.

The present design assumes a total energy content of 50 MJ of 20-MeV runaways (equivalent to a ring beam current of about 10 MA) hitting the wall - or the divertor - distributed over an area of 10 m^2 (i.e. 5 MJ/m^2). Assuming a spatially-uniform pulse deposition in a 0.5 cm shield of Be or graphite, the shield will be heated to only about 300 °C. However, neither the expected depositable beam energy, the deposition area nor the deposition profile are well established at present. The results of additional - in part presently proceeding - analyses will be required to obtain a more sound assessment. Specifically, the needed analyses should include at least the following areas:

(a) Total Runaway Energy: Under some circumstances, in particular with fast current quenches, a sizable part of the inductive energy of a runaway beam may be transformed into relativistic runaway energies during the decay. In this case, the present assumption may be an underestimate by possibly an order of magnitude.

(b) Deposition Area: The presently assumed 10 m^2 are equivalent to a uniform band of about 20 cm width around the circumference. The realism of such an assumption carries a sizable uncertainty: while the actual beam deposition may be more spread out vertically in a VDE, the toroidal distribution may be much more peaked - depending on the toroidal symmetry of the field perturbations during the disruptions etc. Obviously, a narrower deposition area could raise the wall deposition density to seriously unsafe values. A sound assessment will need more detailed analyses of beam dynamics and beam particle orbits under disruption conditions relative to the - possibly rippled - wall surfaces.

(c) RA-Generation: (i) Field-Closure: RA generation and avalanching will require the existence of sizable regions with closed field lines, while open field surfaces seem required for the fast thermal quench. While such field closure after the thermal quench is indicated in the present experiments by the existence of RA's, its scaling needs to be assured, and will depend on the actual plasma conditions in ITER disruptions. So far no calculations of field closure appear to exist. (ii) The times needed for closure and the respective plasma conditions also may sensitively determine the amounts of seed electrons for avalanching, e.g. from trapped high-temperature electrons [7]. The relevant plasma conditions and field configurations need to be known in more detail, and might also be used to suppress RA generation.

(d) Specific Potential Avoidance Procedures: In addition to specific plasma parameters, runaway generation/avalanching might be suppressable by certain specific procedures. Present ideas/investigations using pellet-and/or jet injections or/and pulsed magnetic perturbations need to be followed to permit an assessment of their potential for ITER. Also, other methods may have to be looked for.

(e) Energy Deposition Depth Profiles in Walls/Divertor: Also, a much wider analysis of expected deposition profiles and their dependence on beam angle, and of the related heating - direct or subsequent - of the graphite/Be shield and the copper substructure is needed. At present calculations appear to exist only for a rather narrow parameter range. More surface-oriented profiles may lead to enhanced, partial surface erosion. Inversely, the higher RA-stopping power of copper could lead to a direct melting of the underlying cooling structure, if too large a part of the beam reaches that structure. According to present estimates, repair/replacement of a broken water line will take 1-2 month which could unduly hamper (or limit) experiments if such disruptions prove to be common.

5. Blanket-Shield Attachment (George Sheffield)

The JCT switched from a blanket/shield attachment design that relied on welding to one that made the attachment with bolts. This bolted design has been evolving throughout the time that this sub-panel has been reviewing the design. It is good that the design process is evolving because the initial embodiment had areas of concern. However, this evolution has made it hard for the sub-panel reviewers to be working with relevant information.

The basic concepts of this new design follow. The bolts were substituted for the welded attachment to simplify the remote removal and reinstallation of the modules. However, since the bolted connection is not as strong as the welded connection several modifications were made to reduce the loads.

The welded design could carry the loads generated by the differential thermal expansion between the modules and the back plate caused by the nuclear heating gradient. This load is moderated in the bolted design by incorporating a set of connections that are flexible in shear.

The welded design could sustain the electromagnetic, EM, loads generated by the eddy currents paths in the modules and back plate. These loads were reduced for the bolted design by introducing insulated surfaces to increase the length of eddy current flow paths.

Though the bolted approach would appear to be easier to handle remotely, the sub-panel has some concern that the tighter tolerances require for the bolts to engage and the possibility of nuclear damage may present a new set of problems. Also, the increase in complexity (many more parts) of the bolted design will increase the probability of a failure that will require remote repair. The JCT may want to do some failure effects analysis.

The addition of insulated surfaces does indeed reduce the EM loads that must be carried through the bolted attachment but finding an insulating material that will sustain the high rad dose in the shield region and, at the same time, withstand the impulse loads from a disruption may not be easy. Failure of the insulated surfaces could substantially increase the EM loads.

Also, the EM loads presently being used may not be the worst case loads for the attachments. The present design has only been analyzed for the fast radial disruption. To assess the load situation, the JCT needs to carry out a complete loads survey which evaluates both moving and stationary events over the whole range of time scales and mechanisms: thermal quench 0.5 to 3.0 ms; disruptions 10 to 300 ms; VDE's 500 ms; and halo currents.

Also, the dynamic mechanical response of the module and its attachment to the back plate should be calculated to determine its resonate mechanical frequencies. EM events at the same frequencies could ring the structure causing high stresses.

A lot of very good, creative engineering work has gone into developing the bolted attachment design. The technique for calculating disruption EM loads on a specific geometry is well understood and operational. However, the procedure followed by the JCT to pick the worst case EM load is inadequate.

There is no dynamic mechanical modeling of the module and its bolted attachment. Therefore, the interaction of the EM load frequencies and the mechanical frequency of the structure has not been determined. The insulating material to be used in the bolted design has not been selected. Its performance in the hostile nuclear environment of the shield module under impulse loads constitutes the largest uncertainty.

The sub-panel feels that the bolted attachment design is maturing but it has not demonstrated its ability to handle the forces and environment of the shield /blanket area. However, this process has benefited from work done by the U.S. Home team and would benefit from an increase in staffing for this area. Also, as back up, the welded design should be brought along in its maturity.

[1] F. C. Schuller, Plasma Physics and Control. Fusion **37**, A135 (1995)
[2] J. Wesley et al Proc. 16th IAEA Fusion Energy Conf. IAEA-CN-64/FP-25
[3] M. Rosenbluth et al, ibid IAEA-CN-64/FP-26
[4] N. Pomphrey and W. Park, APS DPP Meeting Poster, 1996

[5] R. Granetz et al, Nucl. Fusion **36**, 545 (1996)
[6] e.g. R.D. Gill, Nucl. Fusion **33**, 1613 (1993)
[7] H. Fleischmann, et al Princeton Plasma Physics Report PPPL-2914 (1993);
R. Jayakumar, et al, Physics Letters A **172**, 447 (1993)

APPENDIX D.IV

SUB-PANEL IV:
ADVANCED MODES, FLEXIBILITY, AND HEATING

Dr. Farrokh Najmabadi (Chairman)
University of California, San Diego

Dr. Steven Allen
Lawrence Livermore National Laboratory

Dr. Nathaniel Fisch
Princeton Plasma Physics Laboratory

Dr. Richard Freeman
General Atomics

Prof. Michaele E. Mauel
Columbia University

Dr. Dale Meade
Princeton Plasma Physics Laboratory

Dr. Miklos Porkolab
Massachusetts Institute of Technology

High performance tokamak discharges with enhanced stability and energy confinement time can be obtained through the control of current and pressure profiles. These advanced tokamak (AT) modes operate at higher poloidal β, higher edge safety factor, and, usually, higher normalized β than those of ITER reference scenario. Operation with a high poloidal β results in a lower plasma current and a high bootstrap current fraction which are the necessary ingredients for steady-state operation.

The AT modes have been identified only in the last decade and the physics of and experience with advanced tokamak operating modes are developing rapidly. Experiments are now planned to (1) investigate all aspects of wall-stabilization of high normalized β operation and control of the resistive wall mode, (2) investigate and compare several advanced operating modes (such as high-l_i and reversed-shear) for possible use in ITER, (3) investigate the sensitivity of high-performance discharges to variations to the plasma and current profiles, (4) investigate the use of on-axis and off-axis current drive to sustain and control the long-time evolution of advanced scenario plasmas, (5) explore non-conventional fueling and pressure profile techniques, *etc*. These experiments will produce a considerable database, guide further theory work, and lead to the eventual development of a predictive ability to model MHD stability, current drive, and profile control of AT modes.

A major portion of AT physics can be explored in existing and future dedicated facilities. However, the fusion α-particle heating in a burning plasma significantly alters the plasma pressure profile and AT operation in a burning plasma discharge should be demonstrated and studied. The DDR states that the next generation large tokamak devices should explore the advanced tokamak modes in order to investigate steady-state or long-pulse discharges with high fusion power and reduced current-drive requirements. What is not made clear in the DDR is that advanced operating modes may also provide flexibility and operational margin to insure successful completion of other important goals of the ITER project. ITER can contribute enormously to the development of attractive steady-state tokamak power sources by retaining the flexibility to investigate these advanced modes. ITER represent an unequaled facility to conduct the needed investigations of advanced operating modes at the appropriate regimes for future fusion power sources. It is critical that operational flexibility be retained in the ITER device so that it can explore the present and “future” AT modes of operation.

Although the MHD design issues for the ITER reference scenario are addressed using relatively well-established extrapolations from companion discharges in present-day tokamak devices, no comparable discharges exist for ITER advanced mode scenarios. At present, the predictive uncertainties make it difficult to adopt specific advanced modes in ITER and/or specify the exact hardware components needed for AT scenarios. The Subpanel, therefore, has focused its effort on examining the capability of ITER in exploring the AT parameter space and only used one AT scenario (reversed-shear) as an example. This mode of operation requires the following approximate conditions to be satisfied: q_0 = 2.5-3.5, q_{min} = 2.1 at r/a = 0.7, and q_{95} = 4.5. The AT scenario for ITER corresponds to a reduced size plasma (a = 2.3 m) and a total current of 12 MA (called the Steady-State Operation Mode in the DDR). This mode is also characterized by 70%-80% bootstrap current, normalized β of 3.6-3.8 (and ultimately 4.5-5.0), central temperatures of 33-25 keV, and an average current drive figure of merit, Γ = 0.21 X10^{20} m^{-3}A/W. Corresponding average densities are 0.7-1.0 X 10^{20} m^{-3} for producing 1000-1500 MW of fusion power.

The primary mission of ITER is fusion burn in a long (1000 s) discharge (see SWG-1 report). While steady-state operation has been maintained as a goal for the later part of ITER experimental activity, work on advanced modes in ITER (which are necessary for steady-state operation) began recently and is almost entirely done by the US Home Team for the ITER program. While significant progress has been made in this area, considerably more work is needed to ensure and enhance the flexibility of ITER to examine AT scenarios. We also note that certain hardware can be added to ITER later when the need for additional flexibility arises while other components such as the poloidal-field (PF) system would be difficult to replace. These “permanent” components require close examination to ensure that sufficient flexibility exists.

Our review is based on Draft ITER Detailed Design Report (DDR), the ITER Technical Advisory Committee (TAC) reports, material presented to the Design Review Committee for the DDR, and discussions with US ITER Home Team members working in this area. Section 1 summarizes the MHD issues. The physics of heating and current drive for AT operations are reviewed in section 2 and the heating and current drive system hardware is reviewed in section 3. The divertor issue are summarized in section 4.

1. MHD EQUILIBRIUM AND STABILITY

Advanced operating scenarios operate at higher normalized β, higher poloidal β, and higher edge safety factor than the ITER reference scenario. The higher performance resulting from the enhanced stability and energy confinement of advanced operating modes allows a significant reduction in plasma current without loss of fusion power. For this reason, advanced operating modes have become the preferred scenario for steady-state operation.

The reduced plasma current during advanced operation also significantly expands the variety of plasma equilibria possible with ITER. Several example equilibria have been extensively studied, and this has demonstrated the flexibility of the ITER poloidal field (PF) system at reduced current. For instance, calculations for the 12 MA reversed magnetic shear (RS) scenario show triangularity as high as 0.47 and elongation as high as 2.2. (shaping flexibility may be considerably reduced if the monolithic central solenoid is retained.) Furthermore, all of the basic MHD equilibrium calculations performed for ITER reference scenarios have also been performed for the RS scenario. The RS equilibria are consistent with the PF system; the PF power supplies are sufficient for dynamic control of the plasma vertical position and the maintenance of edge plasma position in the divertor. Equilibrium restoration following realistic repetitive plasma disturbances do not overheat the cold-structure. These design calculations indicate that no major MHD equilibrium design issues exist which might prevent the basic plasma operation of a variety of advanced performance scenarios within ITER.

Although the ITER design is able to maintain and control plasma equilibrium during advanced mode operation, significant challenges exist. These challenges include (1) controlling plasma instability at high normalized β, (2) controlling both plasma current and plasma pressure profiles at high poloidal β, and (3) preparing plasma equilibrium at low-plasma current in the presence of large toroidal field ripple.

Attractive advanced operating modes maintain high fusion power at reduced current by operating at elevated normalized β. This may excite additional MHD instabilities within ITER. In the reference scenario, fast nonaxisymmetric MHD instabilities are stable, and the MHD instabilities of greatest concern are internal modes: neoclassical tearing modes, sawteeth, and edge-localized ballooning modes. In some advanced operating modes (like the RS mode), ITER will obtain stability of fast MHD instabilities by wall-stabilization. However, wall-stabilization of fast external modes creates conditions for excitation of slowing growing external instabilities (called the resistive wall modes). Resistive wall modes produce a growing helical distortion of the plasma equilibrium as nonaxisymmetric poloidal fields resistively penetrate the blanket assembly and the vacuum vessel. As resistive wall modes grow, they may destroy plasma edge confinement, couple to internal tearing modes, and/or lead to major disruptions.

Advanced operating modes at low plasma current require operation at high poloidal β. Indeed, high poloidal β is a major attraction of the advanced scenarios since this increases the bootstrap current fraction and reduces the auxiliary power required for steady-state or ultra-long-pulse discharges. However, the presence of high fractions of bootstrap current changes the dynamic response of the plasma to disturbances. Changes in the plasma pressure profile lead to changes in the plasma current profile. Long-time equilibrium control will require both pressure as well as current profile control. Additionally, high-performance operating modes have profiles near instability boundaries. If a disturbance rearranges plasma pressure (for example, creating a more peaked pressure or current profile during RS operation), fast MHD instability could result leading to disruption. The disruption dynamics at high performance may have different characteristics than the disruptions occurring during the reference scenario. Because of the sensitivity of advanced modes to plasma instability and the coupling of the pressure and current profiles, both long-time and dynamic equilibrium control of plasmas in advanced modes is more difficult than the control of the reference scenario.

Finally, fast-particles confined to plasmas with reduced toroidal current are more susceptible to toroidal ripple. Calculations of alpha ripple loss for representative RS scenarios indicate fast α-particle loss rates ranging from 5% to 16% of the α-particle production rate depending upon the details of the current profile and the position of the plasma. If ripple-induced α-particle loss is not addressed, the ability to make use of low-current advanced modes for fusion power production will be severely limited.

Although advanced modes present significant challenges to ITER operation, these challenges can probably be resolved through continued progress in the scientific understanding of plasma instability and profile control and through additions or modifications to the ITER design. Several possibilities exist for control of the resistive wall mode. Recent experiments have shown that when the plasma rotation at the outermost dominate resonant surfaces, Ω, exceed a few times the inverse wall time constant, $\Omega\tau_W$ • 4-10π, the resistive wall mode is stable. Some theoretical calculations indicate that faster rotation rates may be required for stabilization. In the event that sufficient plasma rotation cannot be induced, theories indicate that an array of relatively low-power error-field coils can be used to feedback stabilize the resistive wall mode in non-rotating plasmas. Finally, modulated currents at rational values of the internal safety factor induced by various current drive options may also stabilize the resistive wall mode due to poloidal coupling. These stabilization options are not incompatible with the draft ITER design.

Plasma profile control options exist which may be used to improve the dynamic and long-term maintenance of advanced modes. These techniques include non-conventional fueling (high-field side injection of pellets or compact toroids) and control of pressure through static or non-static creation of internal transport barriers. These techniques of controlling the pressure profile are active areas of present research; however, the ITER design does not appear to prevent their use as a design modification.

Finally, the issue of toroidal field ripple is important and addressed in more detail in other areas of the design review. ITER high toroidal field ripple presents serious challenges to α-particle confinement at low current as well as the creation of locked internal tearing modes which may prevent the plasma rotation required for plasma stability and the formation of internal transport barriers. Nevertheless, techniques have been identified which significantly reduce the effects of toroidal field ripple. For example, ferromagnetic inserts have been shown to reduce ripple by nearly a factor of three. This ripple reduction would reduce fast α-particle losses and allow advanced modes to operate with DT and during conditions of high fusion power. The ripple reductions required to prevent the onset of locked modes is not well understood at this time.

Although the MHD design issues for the ITER reference scenario are addressed using relatively well-established extrapolations from companion discharges in present-day tokamak devices, no comparable discharges exist for ITER advanced mode scenarios. The startup and quasi-long-pulse operation of ITER demonstration discharges provide us with general MHD characteristics which may be expected in ITER, but, at the present time, it is much more difficult to predict the general MHD behavior of advanced modes in ITER--A situation that will probably be resolved within a few years given the rapid pace of advancement of experimental and theoretical understanding of advanced operating modes.

In summary, to preserve advanced mode options in ITER, consideration need be given to design modifications or additions in the areas of (1) PF flexibility and optimization, (2) resistive wall mode control, (3) pressure profile control, and (4) toroidal ripple reduction.

3. HEATING AND CURRENT DRIVE PHYSICS

We concur with the ITER TAC conclusion that no one heating and current drive method can satisfy all the physics needs for ITER — start-up assist, heating to ignition, burn control, magnetohydrodynamic control, current drive on- and off-axis, and driven rotation. In addition to inductive drive, four current drive techniques are considered for ITER in the DDR: neutral beam (NBI), fast-wave (FW), electron cyclotron (EC), and lower-hybrid (LH). The four heating and current drive candidates have been developed to a substantial level with good progress in many areas, *e.g.*, coupling, source and window development, and current drive capability. All four candidate methods need further R&D to meet the ITER specifications. The TAC has also endorsed the JCT position that a selection of one or more preferred methods is neither necessary nor desirable at this time. This section reviews the physics as well as specific launching issues as appropriate for current drive requirements. We will also comment on applications to the reversed shear (or negative central shear) mode. Heating and current drive system hardware is reviewed in the next section.

3.1 Neutral Beam Current Drive

For ITER applications, 1.0 MeV negative ion beams are being developed primarily by our Japanese partners. Such energies are necessary for penetration to the plasma core. At the highest densities considered for ITER even higher energies would be desirable, especially for tangential injection for current drive applications. Additional functions of NBI include momentum transfer for plasma rotation to stabilize external kink modes, especially at the high normalized β of AT scenarios (although low energy beams are better suited for this). The efficiencies for current drive presented in the DDR are reasonable, and they have been calculated by the highly reliable ACCOME code. Typical efficiencies given are $\Gamma = 0.17\text{-}0.25 \times 10^{20}\ m^{-2}A/W$. This implies driven currents of 1.1 MA for the reference case at an average density of $1 \times 10^{20}\ m^{-3}$, and 1.6 MA in the smaller plasma characteristic of the AT (steady state) mode of operation. The NBI-driven currents in the smaller plasmas of the AT scenario have relatively broad profiles, satisfying the requirements of reversed shear and high central q values (of the order of 2.5-3.5, as required in the most promising AT tokamak q-profiles). The efficiency in the “reference” case is lower since the tangency radius of injection is at 6.5 m, and part of the beam particles end up with trapped orbits. The “steady state scenario” has a beam tangency radius of 7.5 m, and therefore the trapped ion fraction is minimized, resulting in increased current drive efficiency.

In the AT mode of operation, the NBI system has to provide a central seed current of about 1.1 MA. This is likely to be provided by about 30-40 MW of NBI at 1.0 MeV beam energies. The rest of the power is needed to heat the plasma and/or to provide rotation for stability (this latter task is of dubious value with MeV beams). If there are 3 beams injected through 3 ports and they are they all co-injection beams, in the low density regime too much current may be driven on-axis, lowering q(0) to unity, rather than say 2.5-3.5, the target values. Therefore, the beam energies may have to be lowered toward 0.5 MeV. It should be noted that the q profiles given for the AT scenario in the DDR were obtained by a zero-dimensional code. Therefore, the numbers given can only serve as guidelines and requirements, not necessarily achievable design values for systems. A self consistent design, with a full current drive and heating complement of subsystems, with self consistent density, temperature and current profiles, and their temporal evolution at full parameters is not presented in the DDR (even without self-consistent transport). To our knowledge, there was one such study with the ACCOME code, using LHCD and NBI-CD; however, at the present time this is not part of the DDR, and in any case, coupling to LHCD is a major issue. Nevertheless, in this study the optimal beam energy was found to be 0.5 MeV in the AT small plasma scenario rather than the 1-MeV beam which is necessary for good penetration in the reference scenario. In summary, "variable-energy" NBI-CD would work well for most applications requiring central current drive, with a desirable smooth and broad central current profile.

3.2. Fast Wave Current Drive and ICRF Heating.

Fast wave heating (ICRF) is well demonstrated on existing experiments at high power levels, including minority heating (H and ^{3}He), 2nd harmonic tritium heating in TFTR, (and ^{3}He heating in other machines - same physics scenario), as well as mode conversion heating (TFTR, C-Mod and JET, as well as possibly on other tokamaks). Thus, the 57 MHz frequency is well chosen for ions in ITER at 5.7 T (second harmonic tritium cyclotron resonance, or ^{3}He minority resonance if needed at low temperatures for startup).

Fast wave current drive in the ICRF regime has been demonstrated recently on several large tokamaks, in particularl on DIII-D (see Fig. 7.3-1 in the DDR, and a more recent reference is the paper presented by R. Prater *et al.* at the IAEA Conference in Montreal, paper F1-CN-64/E-1). The current drive efficiency for four strap antennas on DIII-D is 0.04 at 6 keV, and more importantly, the efficiency is shown to be a linear function of T_e. Hence, in ITER this extrapolates to a current drive efficiency of 0.20 at 30 keV for four strap antennas, a perfectly acceptable value. Higher efficiencies would be expected for an 8-strap antenna, however this is not likely in the present ITER design if the antenna has to fit in a port of 1.6 m toroidal extent. At the present time 4 such ports have been reserved for ICRF antennas in ITER. The antenna voltages corresponding to 12.5 MW power per port, are 38 keV for 90 degree phasing of four strap antennas (two arrays poloidally), an acceptable upper limit if ELM activity is mitigated. Fast wave driven current is found to be centrally peaked in all experiments, as expected from theory, and hence it is not useful for off-axis profile control. However, it may be deployed as a reversed central current drive technique for central q(0) control, especially in the AT scenario.

The main FWCD scenario proposed for ITER is at 57 MHz, at a toroidal field of 5.7 T. However, this is also the main heating scenario, utilizing the harmonic of the tritium cyclotron frequency. Thus, the absorption mechanisms will compete, and the current drive efficiency will suffer. The exact power split between electrons and tritium ions may be strongly affected by the antenna spectrum and exact magnetic field. The calculations are quoted from the reference of Kimura *et al*, JAERI Report 95-070 (1996). This report was prepared in 1995, and used an 8-strap antenna of 3.0 m toroidal extent as the reference case. This antenna design has since been abandoned in favor of the 1.6 m, 4-strap antenna design (P.M. Ryan and D.W. Swain, ORNL/TM-13370, Feb. 1997), and therefore the Kimura report conclusions do not necessarily hold. As a consequence, the ICRF/FWCD section in the DDR suffers from inconsistencies, and the physics modeling remains to be redone with the new antenna design. It may impact some of the operating scenarios, in particular an acceptable current drive scenario.

There are also other issues that need to be reconsidered, such as parasitic H minority absorption in front of the antenna in the 5.7 T case (it will be difficult to reduce the hydrogen concentration in ITER below say 0.5%). Another issue is α-particle particle absorption on the high field side (at 57 MHz and 5.7 T the α-particle cyclotron resonance is located on the high field side, at the location of the deuterium resonance, about half way out radially). For the 8-strap antenna, the Kimura report finds no impact, however, given the width of the resonance at the α-particle energies, for the 4-strap antennas this may not be true. The 20-30 MHz scenario mentioned as a backup may not be acceptable since for the restricted antenna geometry the parallel index of refraction will be above 10 in the plasma center, resulting in strong electron trapping and reduced current drive efficiency; in addition, parasitic absorption by mode conversion into shear Alfven waves near the plasma edge must be considered since this may also be a drain on the available power for current drive at the center. Another back-up, the ^{3}He minority current drive (75 MHz scenario) has been found inefficient during the TPX studies, hence such a scenario has to be documented convincingly for ITER applications.

There may be attractive mode-conversion CD scenarios (*e.g.,* mode-conversion current drive, MCCD) that would be very useful for profile control, but they are discussed in detail in the DDR. Admittedly, experimental verification is still lacking, although efficient mode-conversion electron heating has been demonstrated recently in TFTR and C-Mod. Also, other magnetic fields, such as 4.8 T have been found to be promising in limited ACCOME FWCD code studies (not included in the DDR), which removes the harmonic tritium cyclotron frequency from the axis so that most of the power is absorbed on electrons, yielding much better current drive efficiencies; however, parasitic minority hydrogen absorption on the low-field side must be considered in this case also.

In summary, the ICRF/FWCD part of the DDR is incomplete at best, with many inconsistencies and omissions. This observation is further supported by the recently changed antenna design. Given the rich variety of experimental results observed world-wide, and the low-cost of immediately available technology, this range of frequencies deserves a better physics evaluation than what is presented in the DDR. There are also other potential applications of ICRF, such as α channeling and mode-converted IBW shear flow thermal barrier formation, none of which is discussed in the DDR.

The steady-state scenario as presented in the DDR will require moving the plasma away from the outside wall area by substantial amounts and this will introduce serious problems for ICRF antenna coupling and hence the power delivered. This has not been discussed in the DDR. In addition, full power CO-FWCD may be a problem, leading to very low values of q(0) as found during the TPX studies. Hence, a mixture of current drive and minority heating may be preferred. Its control would depend on the exact value of the toroidal field as well as antenna phasing. Interaction of the wave fields with the incident NBI ions, or with α-particles may be a problem and must be analyzed.

3.3. Electron Cyclotron Heating (ECRH) and Current Drive (ECCD)

Technologically, this is the most promising technique for both electron heating and current drive in reactor grade plasmas. This is based on the observation that no launching structure in contact with the plasma is needed for ECRH/ECCD. Furthermore, efficient wave-plasma coupling in both the "reference scenario," as well as in the "steady-state" AT mode of operation is equally feasible. Current profile control is possible by steering launching mirrors in the toroidal plane (section 4.). Single pass absorption in the O-mode of radiation is more than adequate all the way from a few keV to 30 keV. The central current drive efficiencies quoted in DDR (0.19) are reasonable and are based on accurate modeling with the TORAY code (which has been benchmarked with the CQL-3 Fokker Planck code). Central heating and central current drive have been optimized by choosing a frequency slightly in excess of the central resonant cyclotron frequency. However, besides central heating, one of the key missions of ECRH should be profile control, and in particular off-axis current drive. This is particularly true for the RS scenario, or for neoclassic tearing mode (island) stabilization in finite β reference plasmas (*i.e.,* above a normalized β of 2.0). In particular, there may be issues with current drive efficiency for the required currents driven at the large minor radius (r/a = 0.7). Basically, the 1.1 MA off-axis current quoted in the DDR with the 50 MW of ECH power is based on highly optimized poloidal launching angles (20 degrees in this case). Other modeling with straight midplane launch (which is more consistent with the present launcher design) may yield only 0.7 MA. Thus, the available power is a factor of two to three too low to achieve the AT scenario presented in Fig. 6.6-1, Chap. III, DDR, where off-axis currents of at least 2 MA may be required. Therefore, if the main scenario for ITER is NBI-CD/ECCD, or ICRH/ FWCD/ECCD, more ECH power may be required, perhaps as much as 100 MW to achieve the AT scenarios presented even for the optimized case. Thus, further optimization of launching geometries and scenarios is required.

Another way to increase the off-axis currents may be to lower the frequencies toward 150 GHz (there is some support for this from recent work of European theorists). This would reduce the central current drive efficiency, but near central heating would be still possible for normal incidence at 17 cm off axis on the low field side, at r/a = 0.06, about the same amount as is the present case with 170 GHz on the high field side (on-axis resonance for perpendicular incidence is 160 GHz at 5.7 T). In all cases heating would be achieved well inside the q = 1 surface, and we know from present day experiments that the heating efficiency remains the same in such circumstances. The central current drive efficiency may suffer somewhat, but this may be of secondary consideration since strong centrally peaked co-current drive would produce undesirably low q(0) values. Therefore, we are recommending further modeling studies from the US theory community of these critical ITER issues. Additional studies of tearing mode stabilization are also highly desirable.

The MW steady state gyrotron sources still need to be developed, including windows, unfortunately the US industrial development work has been terminated to the detriment of ITER prospects, as well as to the US base research program. Further high power experimental testing of these concepts is also necessary, including fundamental O-mode testing.

3.4. Lower Hybrid Current Drive

This concept is not part of the official complement of powers. However, the Europeans continue work in this area, and US modellers have also carried out one detailed study of the AT scenario, using the highly reliable and benchmarked ACCOME code (ISCUS presentation by P. T. Bonoli and M. Porkolab, August, 1996, Livermore meeting). It was found that a very attractive scenario could be produced with 50 MW of NBCD and 50 MW of LHCD at 5.5 GHz, leading to the current profile predicted by Fig. 6.6-1 in Chap. III of the DDR. The bootstrap current fraction was above 70%, and the driven off-axis lower hybrid current was 2.0 MA, as required, leading to a q_{min} at r/a = 0.7. There are two figures and one Table reproduced from the ISCUS report, showing the favorable predictions for the reference AT case using only 50 MW of LH and 40 MW of NBI power. There is little doubt that for “edge current-drive” LHCD is the most efficient current driver owing to its characteristics of carrying the current with mildly energetic electrons with a dominant parallel energy component.

The biggest issue here is to solve the antenna technology, and to couple to the reduced size AT plasma [for good coupling, the antenna (grill) interface needs a plasma density of approximately 2-3 X $10^{17}m^{-3}$]. Another issue might be to provide the precise spectral control using the multi-junction grill for accurate q-profile control.

4. HEATING AND CURRENT DRIVE SYSTEMS

The heating systems identified for the ITER mission are ion cyclotron heating and current drive (ICH and ICCD), electron cyclotron heating and current drive (ECH and ECCD), lower-hybrid current drive (LHCD) and negative ion neutral beam (N-NBI) heating and current drive. The ion-cyclotron resonant heating (ICRH), electron-cyclotron resonant heating (ECRH), and lower-hybrid current drive (LHCD) systems use standardized in-port concepts. The LHCD system is being designed by the EU Home Team on a voluntary basis. System parameters have been adequately chosen to fulfill ITER requirements for localized heating and central current drive (ECRH, ICRH, N-NBI), ion heating (ICRH, N-NBI), off-axis current drive (ECRH, LHCD), start-up (ECRH, ICRH), and discharge cleaning (ECRH, ICRH). The designs are based on operating experience with comparable systems in present-day tokamaks.

A brief assessment of each system based on the operational experience base and the technology status is provided below. Integration into ITER, together with reliability, remote maintenance and safety issues, will play a key role in the ultimate choice of the heating and current drive systems.

4.1. Negative Ion Neutral Beam Heating and Current Drive System

Positive ion neutral beam heating systems have been the workhorses of the fusion program. Effective heating and current drive have been demonstrated at high power in many devices. The classical nature of the heating process provides confident extrapolation to higher energies. The main physics issue related to the MeV energy beams required for ITER appears to be the possibility of energetic ion losses due to MHD modes driven by the fast particles. The only major experiment deploying high power energetic negative ion beams is JT60-U and only preliminary results are available.

The technology development required to provide MeV beams for ITER at the 50 -100 MW power level needed is a significant extrapolation from the JT60-U beams in terms of energy throughput and thermal management. Given even the most optimistic projections of the neutralization efficiency, the heat load on the dumps is substantial and they must operate for long pulses. Since the beams extend the tritium envelope, all maintenance must be performed remotely, so reliability is essential. A few key technology concerns that were identified in earlier reviews, in addition to the tritium and thermal management issues, are the magnetic shielding required at the neutralizer and its impact on the tokamak error fields, the large torus valve required which is beyond present experience, the ability to fabricate ceramic insulators in the large diameters required and their ability to maintain voltage stand-off under the neutron and radiation flux environment, and the ability to produce a cw source of negative ions reliably. None of these design issues represents a show stopper, but sufficient resources must be applied to resolve them. Substantial advances that have been made in the past two years in the performance of negative-ion-based neutral beams. The operation of the new negative-ion-based neutral beam system on JT-60U at the multi-MW level is a very encouraging milestone in the application of this technology, but substantial further progress must be made to qualify this technology for ITER.

As identified by TAC, the principal development tasks remaining are (i) to demonstrate, simultaneously, for ITER pulse lengths the required energy, current density, and electron suppression, (ii) to develop the technology for the production of the very large (3 m diameter) alumina ceramic insulators, which are needed to handle the high voltages in the presence of the expected radiation field, (iii) to ensure remote handling compatibility, and (iv) to refine the interfaces to the tokamak and other systems, while ensuring adequate radiation shielding. Since the R&D plans for the N-NBI systems was not available to the reviewers, it was not possible to accurately assess the likely success of these efforts.

4.2. Ion Cyclotron Heating and Current Drive System

Ion cyclotron systems are in wide use throughout the worldwide fusion program, and nearly all major tokamak experimental devices employ these systems. The application of ICH to ITER will extend the experience base to discharges with much larger neutron and thermal loads, and experience is already available from TFTR, and in the future from JET, with reacting plasmas. The ability to heat with ICRF is well established, and TFTR has recently demonstrated ITER heating scenarios utilizing tritium. The JET system has a source power of 32 MW and the TFTR system has a source power of 14 MW, so the experience base exists for high power utilization. The experience base with current drive is more limited, but DIII-D, TFTR, and Tore Supra have used fast waves in the ion cyclotron range to drive plasma current by damping directly on the electrons with resulting current drive efficiencies consistent with the theoretical predictions. In addition, ion cyclotron current drive (ICCD) has been demonstrated on JET and mode conversion heating has been demonstrated on TFTR and C-Mod. As discussed in section 3., these alternate ion cyclotron wave current drive schemes may be applicable to the mode stabilization and off-axis current drive missions for ITER.

The antenna design approach is conventional and appears robust against the expected heat loads and disruptions. In order to maintain frequency flexibility, a somewhat novel sliding contact is provided inside the vessel; this has not been deployed on any present day machines, so it represents a risk. The power systems and transmission components required are generally available from commercial vendors, although fusion systems tend to operate at unit powers higher than off-the-shelf equipment.

A major concern for the extrapolation of ICH and ICCD systems to ITER is the power handling capability. The presence of ELMs causes wide variations in the plasma loading and may impact the power coupled to the plasma. The ability to deliver the design power reliably may also be impacted by the large antenna-plasma separation projected because this forces the design antenna voltages to be chosen at the upper end of the experience base. The situation is further stressed by an environment where particle and energy fluxes will be beyond current levels; hence, it would be prudent to develop the folded waveguide concept, a combline, or similar alternative to facilitate coupling with the large antenna-plasma gap.

Despite the concerns expressed, the ICH system represents a conservative choice since its basic missions have been demonstrated in present experiments and the technology needed does not represent a large extrapolation from present practice.

4.3. Electron Cyclotron Heating and Current Drive System

The physics of ECH for heating and current drive is well established, although the power levels utilized in experiments has not been at the levels used in neutral beam or ion cyclotron experiments. Previous T-10 results demonstrated effective heating at MW levels, and several multi-MW ECH experiments are expected to operate over the next few years. The efficiency of off-axis current drive with EC waves must be quantified.

A key advantage for ECH compared to other RF schemes is the launcher structure can be remote from the plasma since the waves can propagate in a vacuum. An antenna or variable frequency source capable of dynamically tracking islands for stabilization applications must be developed and demonstrated. Evacuated transmission lines allow extremely high power density transmission over large distances and minimal port space is required. However, the development of high power, long pulse sources at 150-170 GHz remains the major issue for ITER. Gyrotrons operating at 0.5-1 MW have been demonstrated at 110 GHz and 140 GHz for pulse lengths of a few seconds and scaling to 170 GHz is not expected to be a major issue. In fact, the primary limitation in the gyrotron development is the lack of a suitable window for a MW high power, long pulse gyrotron. Several promising window concepts are being explored and a solution appears imminent. Unfortunately, CPI (formerly Varian) is the only developer which has demonstrated a true steady-state gyrotron operation, so the reduction of US ITER support for ECH leaves the development effort in a much weaker position.

4.4. Lower Hybrid Current Drive System

The ability of Lower Hybrid Waves (LHW) to drive current with high efficiency and to modify the current profile has been demonstrated in several machines. In ITER, LHW can fulfill several tasks, including off-axis current profile control during burn, driving a high fraction of the total current in the advanced scenarios and assisting the formation of the discharge during start up. LHW will not be able to penetrate the central core of the ITER plasma due to the high electron temperature, but from a physics point of view their strong first-pass damping and high current drive efficiency makes them an attractive candidate for off-axis current profile control. In addition, feedback stabilization of tearing modes, although not yet experimentally tested, has been recognized as a very attractive feature of LHW since it would require modest power (~5 MW), owing to the high electron temperature and the large size of ITER.

The optimum frequency is 5 GHz. This choice of frequency takes into account issues associated with high power klystron amplifiers, transmission line components, a convenient size for the waveguide array couplers, and the avoidance of the α-particle, perpendicular Landau damping.

A major concern is the coupler survivability in the proximity of the plasma: the Europeans have developed a design which would allow cooling of the waveguide septa. The condition that a minimum clearance be maintained between coupler and plasma might impact coupling, which requires a density of the order of 3×10^{17} m^{-3}: in this case tailoring of the plasma density at the coupler face might be necessary, either with local gas puffing or RF ionization.

5. DIVERTORS

The reference ITER case uses a Partially- or Fully- Detached Divertor (PDD or FDD) which means that the ion flux either near the separatrix (Partially) or all across the plasma

(Fully) is reduced at the divertor plate. The plasma pressure also drops along these field lines from the midplane to the divertor plate. Radiation from deuterium and impurities in the divertor, along with an edge mantle, is used to reduce the heat flux to acceptable levels at the divertor plate (5-10 MW/m^2). The divertor structure is long (2m) and is closed. Sixteen cryopumps are used for particle exhaust. Carbon-Fiber composite material is used in the high heat flux areas, the divertor is tungsten, and the remainder of the first wall is beryllium.

There is a substantial amount of experimental and modeling work that has been performed with this overall concept, and it seems credible for the baseline case. The divertor subpanel has noted concerns with the effects of radiation on the H-mode threshold and core confinement, potential difficulties in controlling the detachment, and the fact that effects due to Edge Localized Modes (ELMs) are uncertain.

The experimental and modeling database for this divertor concept with AT operation is much more limited. JT-60 has operated successfully some discharges with reverse central shear (RCS) and a "radiative divertor." In double null operation, DIII–D has noted that the sensitivity of the confinement is related to the x-point height (in the current open divertor geometry); a high x-point is advantageous (note that the x-point to strike point distance and the x-point height on DIII–D is ~20 cm, while the divertor length is nearly 2m in ITER, and a baffle has been used in the "private flux" region below the x-point). DIII–D currently has installed a more closed, high-triangularity divertor with pumping and a campaign of experiments focusing on the coupled divertor-core problem, including AT modes, is scheduled for the 1997 run period. Preliminary work with UEDGE and EIRENE models do not indicate any particular problems with high-triangularity operation, but self-consistent calculations with RCS-core and PDD-divertor plasmas have not been performed. ASDEX-U is currently considering modifications that will allow higher triangularity divertor operation in the future.

Perhaps the biggest uncertainty is the compatibility of the high-density divertor operation with the lower-density, high-confinement, optimized current profiles of the Advanced Modes. Fortunately, most tokamaks are currently focusing on better isolation of the divertor plasma. That is, nearly every machine is currently modifying their divertor for more closed operation, which should more effectively decouple the core and edge plasmas. Another concern is that Advanced Mode operation requires lower density and therefore better particle control. While the ITER pumping system is large (200 m^3/s) and seems adequate for the baseline case, there must be sufficient particle flux at the divertor in the Advanced Mode (at lower density) that can be used to control the core density (*i.e.*, particle exhaust is the product of the pumping rate times the pressure) and the helium exhaust must be maintained. Fortunately, experiments (JT60 and DIII–D) and modeling (UEDGE-EIRENE, B2-EIRENE) are currently in progress to answer this key question.

A more obvious issue is whether the location of the divertor strike points and x-point will be maintained in the optimum location during operation in the Advanced Modes. Advanced Mode equilbria (*i.e.* high triangularity) have been calculated that are well-matched to the ITER divertor. Dynamic analyses have indicated that the divertor can be controlled without excessive cold structure heating. It is also important that the divertor configuration is maintained during startup and when there are changes in the current profile. (In the TPX case, the inner divertor was left fairly open as the inner strike point would move up and down in response to changes in internal inductance.) Changes in the core plasma can also change the flux expansion in the divertor, which effectively changes the "width" of the divertor chamber (at least for charged particles). The coupling to the RF antennas near the midplane must also be simultaneously maintained by the control system. So far, the work that has been done (some may not be in the DDR) indicates that the control is adequate. A specific plasma "controller" has not been written for the Advanced Modes, (but does exist for the ignited mode), but a sensitivity test was performed by moving the plasma centroid by 10 cm and the system had to restore the perturbation before the centroid reached 15 cm. It was assumed that the divertor response was proportional. A detailed check of the strike point and x-point in these calculations would be useful.

The divertor operation with large ELMs is problematic (although the control scenario for the ignited mode was checked for response to large ELMs), and ELMing H-mode is envisioned for the base case.. There are currently development paths for both ELMing and ELM-free Advanced Modes in current machines. If achievable, the ELM-free scenarios would remove the transients from the divertor (of both particles and power), but an efficient particle control scenario needs to be developed. This will come out of experiments and modeling which are scheduled on JT-60 and DIII–D in the next year.

Overall, the ITER divertor should be able to handle AT equilibria (*i.e.* high triangularity). There could be, however, limitations to the achievable pulse length or the operational window. The fact that the divertor volume is relatively large and has a modular and flexible design means that future upgrades or modifications can be made to optimize the design specifically for AT modes, if required.

The proposed “Partially Detached Divertor” (PDD) scenario is credible for the baseline operation (as noted by the main subpanel on the Divertor). However, further experimentation and modeling are required to determine if the composite issues of current profile control, density control (at lower density than the baseline), heat flux reduction (with PDD, a high-density divertor), and enhanced confinement are compatible with operation in advanced modes.

Effective control of the divertor strike points and the flux expansion must be maintained for the divertor to operate properly. This could be more difficult to achieve in Advanced Modes because of the more dramatic plasma shaping (with a relatively closed divertor) and there will be changes in the current and density profiles, particularly during startup. The ITER design has looked at some of these issues (some results may not be included in the DDR), and there does not seem to be a major problem.

APPENDIX D.V

SUB-PANEL V:
OPERABILITY AND SAFETY

Dr. James Luxon (Chairman)
General Atomics

Dr. Richard Callis
General Atomics

Dr. Tom Casper
Lawrence Livermore National Laboratory

Ms. Melissa Cray *
Los Alamos National Laboratory

Dr. John DeLooper
Princeton Plasma Physics Laboratory

Dr. Richard Hawryluk
Princeton Plasma Physics Laboratory

Dr. James Irby
Massachusetts Institute of Technology

Dr. David Johnson
Plasma Physics Laboratory

Dr. Arnold Kellman
General Atomics

Dr. Michael Williams
Princeton Plasma Physics Laboratory

* - FESAC Member

Introduction

The operation of ITER with control, diagnostic, safety, and data systems more complex than any experienced on tokamaks to date, represents a noteworthy challenge. After a period of operations, interaction with the hardware will be done remotely. The understanding of tokamak physics will evolve during the design and construction of ITER. As a result, every effort must be made to maintain flexibility in the ITER design. The segmentation of the ohmic heating solenoid and the adoption of diagnostic port plugs are examples of flexibility (although the loss of V-sec in the solenoid change is a potential reduction of flexibility).

This section covered the specific areas of: Machine Operability; Plasma Control; Diagnostics; Computer and Data Handling; and Environment, Safety and Health. It is evident that a great deal of work has been accomplished in design of ITER. The ITER Team should be commended for this effort. However, this review team has identified issues which should be addressed in order for ITER to be a scientifically productive experiment. These issues include:

¥ Responsibility for operations
¥ Number of allocated plasma pulses
¥ Availability goals
¥ Allocation of system reliability goals
¥ Tritium retention in the vacuum vessel
¥ Wall conditioning
¥ Hydrogen operations

- Plasma shape and position control
- Error fields and correction coils
- AC losses and off-normal/transient events
- Diagnostics design progress/visibility
- Diagnostic access
- Diagnostic R&D
- Diagnostic plasma control

¥ Survivability of optical components
¥ Remote operations
¥ WAN speed connections
¥ Advanced data storage
¥ Evacuation criteria
¥ Release limits

- Worker ALARA issues
- Assumptions
- Tritium accountability
- Machine operations
- Safety of design
- Waste products

Each issue is documented with recommendations in the subsections of this report.

Machine Operability

ITER has the potential to operate effectively and achieve its program goals. No concerns were identified which would *a priori* preclude this. However, a number of areas must be addressed in greater depth by the ITER team before one can say with confidence that their operational goals will be achieved.

Many of the issues that need to be addressed in order to ensure effective operations don't appear to be getting addressed on a project-wide basis. Reliability and availability goals aren't getting propagated down to the system level. Inter-group design issues don't seem to get addressed appropriately. An area of concern is the development of an effective plasma control system and the resolution of the many inter group issues inherent in its design.

Recommendation. A group should be identified to take responsibility for operations related issues of the design including assuring that the facility will be able to meet its reliability and availability goals.

The total number of pulses allocated for operation during the basic performance phase (BPP) would significantly limit the scope of the research and technology program and should be increased. The total number of plasma pulses of all kinds provided in the BPP is 15000 over 10 years, or 1500 per year. ITER studies have indicated that this number of pulses is adequate to complete the requirements of the BPP phase assuming that the physics program is restricted to that specified in the reference plan for preparing for D-T, and that the BPP phase is not extended. These assumptions would constrain the research program since time is not clearly identified for activities ranging from checkout of diagnostics to the development and characterization of new (unanticipated) burning plasma physics operating modes.

Recommendation. The non-replaceable major components of ITER (toroidal field coil, vacuum vessel, power systems, etc.) should be designed for 30,000 to 50,000 shots in the BPP.

The availability of ITER and related goals are defined in a manner which is difficult to interpret and implement. Availability is defined as "the ratio of the product of the number of pulses and their duration in an operation run period if the device is operational at its acceptable planned performance level, to the product of the number of pulses and the average duration which could be achieved during that run period in the absence of component failures." This definition is similar to that used by existing tokamaks albeit without the time weighting. The goal for ITER availability under this definition increases from 4% during the 4th and 5th years of the BPP to 10% at the end of the BPP. This goal is important during the design because it identifies to the engineering staff the subsystem requirements.

Recommendation. An appropriate goal for the availability of ITER (defined as shots completed divided by shots planned) would be 80-90% in the last years of the BPP.

A formal reliability-availability-maintainability (RAM) program does not appear to exist. While some detailed calculations of mean-time-to-failure have been performed as part of the safety analyses and some difficult issues associated with remote handling have been considered in detail, a standard does not exist to judge if a system has met its objectives. It is not possible to assess whether ITER will meet its availability objective.

Recommendation. Implement a reliability, availability, maintainability (RAM) program to assure ITER design meets the established performance objectives.

ITER will use large quantities of tritium. Because tritium is a radioactive gas and a sensitive nuclear material, special considerations are required for inventory control. Tritium retention in the co-deposited carbon films due to erosion and redeposition of plasma facing components is restricted to 1 kg-T. The increase in the co-deposition inventory in ITER is expected to be in the range of 1-20 g/1000 s pulse and according to TFTR experience possibly even higher. These estimates are recognized to be quite uncertain but could limit the operating phase to 50 shots (or less if based on the TFTR data) before active techniques are implemented to remove the tritium from the vessel. A method does not exist to determine the tritium inventory inside the vessel to verify compliance with the 1 kg-T safety limit. A method to remove the tritium from the vessel has not been established. Most of the techniques proposed will significantly decondition the machine and require substantial time to restore the machine to high performance. If the tritium retention is high, machine availability could significantly decrease.

Recommendation. A more complete design element to address tritium retention, removal and assessment in the vacuum vessel is required along with appropriate supporting R&D.

Proper wall conditioning has been crucial to good tokamak operation. The long pulse nature of ITER will require that new techniques be developed since techniques used today will not be applicable in ITER because the toroidal field will not be turned off between discharges. One of the most effective techniques in early preparation is the use of high temperature in situ baking to temperatures of at least 300 C. The baking temperature in ITER is presently limited to 200 C to 240 C.

Recommendation. ITER should fully define the techniques to be used for wall preparation and carry out the supporting R&D. The baking temperature should be increased to at least 300 C.

The ITER plan calls for a three year hydrogen phase followed by a brief deuterium phase and then deuterium and tritium. Presently there is very little data available on operations in hydrogen or for projecting from operation with one isotope to another.

Recommendation. Research should be carried out in the home team on hydrogen plasmas and the isotope dependence of plasma operation.

Plasma Control

A considerable amount of high quality work has been performed by the ITER team in this area. However, there are a number of outstanding issues, especially in plasma shape and position control, error field correction and AC losses that must be addressed more thoroughly.

Plasma Initiation There are no fundamental problems remaining in this area. Based on experience of present devices and ITER specific modeling, the requirement of ECH for reliable breakdown and burnthrough is justified.

Recommendation. Evaluate the level of installed ECH power needed to provide the required 3 MW of absorbed power with high reliability. In addition, validate the underlying assumptions in establishing this requirement. Two areas that warrant further analysis include the effect of the 3D structure and the ferritic inserts in the toroidal field on the error fields at breakdown.

V-sec Consumption Both the monolithic central solenoid and the proposed segmented solenoid options should provide sufficient flux for over 1000 second burn requirement at nominal ITER parameters. It should be noted, that the flux available in the segmented option is lower and while the reference value of internal inductance of li(3)~0.9 is reasonable, if it is exceeded by more than 0.1, the 1000 second burn requirement will likely not be satisfied. Similarly, if confinement is degraded by more than 30% from nominal (H=2.0) the burn time will likely be only 700-800 seconds.

Plasma Shape And Position Control A wide variety of plasma equilibria have been examined and the PF system provides considerable flexibility for ITER. Work remains especially in the evaluation of the newly proposed coil set, although initial results indicate that it provides improved control capability.

Recommendation. The combined effects of the static reconstruction errors and the effect of the eddy currents on the equilibrium reconstruction need to be incorporated into the evaluation of the control system performance.

Error Fields and Correction Coils The error correction coils proposed in the IDR and the DDR are presently being redesigned based on recent physics input concerning the importance of correcting multiple modes. While the new design provides increased flexibility, many questions remain in determining TF and PF coil placement accuracy, the effect of cool down, techniques to accurately measure error fields prior to and during machine operation, and the effect of Incoloy in the coils and ferromagnetic inserts in the TF coil.

Recommendation. Confirmation of the importance of multi-mode correction must be provided by Physics R&D. It is essential that one group be responsible for coordinating the full task of minimizing and correcting error fields. The use of modest amounts of neutral beam injection for rotation should be evaluated given the uncertainties in the error field correction.

AC losses and Off-Normal/Transient Events A variety of off-normal and transient events have been specified and study of the control system response to those events is on-going. However, control systems response to off-normal events presents potentially the most serious obstacle to ITER achieving its operational requirements of acceptable plasma control and pulse duration. The biggest uncertainty with the largest impact occurs when the control system tries to respond to large repetitive changes in the plasma, e.g. the ELM. The ELM characteristics input to the analysis are purely empirical with large error bars and are then extrapolated to ITER. When applied to plasma control, the large error bars translate into uncertainty in the power and time derivatives of power required to control the plasma. For AC loss calculations, the uncertainty in the ELM specification results in a large uncertainty in the maximum pulse duration that the cryogenic system can support. In the worst case, this has the potential to limit the pulse duration of ITER below the 1000 second specification.

Recommendation. Better specification of the ELM and the other transient and off-normal events by the physics programs is needed. ELMs should be included in the choice of transient events in the Reference Control Action described in DDD 4.7 (2.1.4.10). It is essential that there be close coordination with and guidance to the control group in order to determine if the control can be sufficiently "softened" to reduce AC losses to acceptable levels while maintaining adequate protection of the first wall.

Diagnostics Evaluation

Progress in Diagnostics Design. The diagnostics design effort for ITER has made steady progress. Measurement concepts were developed with adequate community input and with openness to the wide ranging interests of the community. Concepts are based on successful experience from existing tokamaks and, with careful implementation on ITER, will fulfill the physics requirements. The number of diagnostics with reasonable design effort has been greatly expanded. However, the EDA effort in diagnostics will not result in descriptions detailed enough to be ready for fabrication. For a small number of the 30-40 systems being considered, the design will be at a level appropriate to begin detailed drawings. However, in many cases, it may be impossible to say at the end of the EDA that the diagnostics will be adequate to meet the mandated requirements.

Recommendation. The level of effort for the diagnostic design should be elevated, with highest priority given to those diagnostics needed for machine safety and plasma control.

Visibility of Diagnostics Effort. The machine interface for diagnostics is far more demanding than in present day tokamaks. Diagnostics play a pivotal role in machine protection and plasma control, and design compromises in other critical systems (e.g. blanket shielding or remote handling) may be needed to accommodate some diagnostics.

Lack of definition in the blanket design and top port design has hampered diagnostic design efforts. It is not clear that there is adequate access allocated for the diagnostics in categories I (Measurements for Machine Protection and Plasma Control) and II (Measurements for Performance Evaluation and Optimization). Neutronics modeling is needed to verify that shielding is adequate in the case where many diagnostic penetrations share a single diagnostic port. The proposed diagnostic usage of the mid plane ports not needed for heating and blanket use in the early phases of ITER operation is endorsed. The visibility of diagnostic issues within the ITER design effort is too low (critical design decisions may never be addressed before designs are frozen), as are the resources allocated to diagnostic design. The concept of "phasing in" diagnostics as time and resources allow will be difficult to apply to ITER, since there is little time between first plasma and DD operation to make such changes, without relying on remote handling.

Recommendation. The visibility of diagnostics interfaces must be increased so that they are seen as part of the overall design, not as a separate entity.

Diagnostics and Plasma Control. There are inconsistencies between the diagnostic needs for plasma control in the physics assessment sections of the DDR, and the classification of required measurements in the diagnostics section. The control needs would indicate that more diagnostics should be in the class designated “for machine protection and plasma control”. Also, clarification would be helpful in the definition of these categories and their relation to plasma control. Development of techniques to measure critical control quantities such as n_D/n_T for burn control, J(r) for equilibrium control, and a substitute for magnetics for steady state position/shape control are appropriately designated high priority R&D issues, as credible methods have yet to be identified.

Recommendation. Clearly define those diagnostics necessary for machine protection and plasma control.

Diagnostic R&D Issues. Many significant diagnostic R&D efforts have been appropriately identified, although it is not clear that they can be addressed before the completion of the EDA.

Recommendation. It is important that each Home Team concentrate on completing the most urgent Diagnostics R&D tasks.

Survivability of Optical Components. The highest risk technical concern for diagnostics, the survivability of optical components, has been identified as a high priority R&D area. Most optical diagnostics will be dependent on mirrors very close to the plasma. The maintenance of adequate optical quality for these mirrors in the presence of neutral particle bombardment and radiation is questionable. This is particularly true in the divertor region, where contamination of mirrors by tile material during disruptions is a serious concern.

Recommendation. Additional evaluation of the survivability of optical components is necessary.

Computer and Data Handling

ITER experimental operations present a complex, interactive environment that places considerable demands on computations systems supporting controls, data acquisition, integrated remote operation, and scientific analysis. The proposed hierarchical, distributed network coordinated by a supervisory system is motivated by requirements for real-time interaction, the volume of machine and scientific data to be acquired, and support for remote operations. This approach naturally extends to Wide Area Network (WAN) access but requires attention to access security and network connectivity performance. A reasonably well-posed philosophy defining the overall structure is developing. Much of the detailed design has been delayed until a later phase. Given the current, rapid development of networks and computer technology, this approach seems reasonable.

ITER remote experiments will require the combined use of systems and teams from ITER and remote sites. Certain key functions will be available only at the ITER site. Physics operation will be supported from a single remote site at any one time. Subsystems can receive discharge parameters and pre-programmed waveforms from the remote site with consistency checked both by remote and ITER site supervisory control systems. The ITER site maintains final approval authority for the discharge.

Recommendation. Detailed definitions of on- and off-site functions should be made available to determine impact on ITER participants. ITER should seek input on remote operations from existing experiments to determine specifics for off-site access to the control systems and to all data.

Networks are separated by function to accommodate disparate requirements for controls, instrumentation, interlocks and audio/video data. The proposed ITER site aggregate network bandwidth of 1Gbps should support the operational requirements. The proposed T1 (1.5Mbps) remote site connections are insufficient to support off-site requirements for high speed data connections, real-time updates of machine status, operations displays and audio/video data.

Recommendation. Adopt higher speed WAN connections to support integrated use of off-site resources: 45Mbps (T3) is routine now and technology supports speeds in excess of 100Mbps.

The data system must support both fast, control room oriented analysis and detailed post-operations analysis. Planned resources for on-site computational power appear to be sufficient to support a significant amount of analysis. The ITER site should have a general purpose computer for analysis that communicates off-site while connected with controls and data acquisition networks. Plasma parameters calculated from related groups of diagnostic data will be provided by the diagnostic control supervisory computer or by computers in the data analysis system. A supervisory computer will coordinate the operation of all the subsystems. Some detailed analysis tasks to be performed by the supervisory computer may put a significant processing load on it.

Recommendation. Define the general purpose computer and its role in computations and communications. Consideration should be given to minimizing computational tasks done by the supervisory computer to keep it free for real-time coordination and safety monitoring tasks.

The need for continuously acquired data has been recognized with the database creation to be done continuously. The detailed database structure must be generalized to also accept continuously generated scientific data during steady-state operation. Dedicated diagnostic subsystems provide data acquisition, reduction and conversion to plasma parameters at the local controller to the extent practical to reduce network data traffic. Intelligent, dedicated control systems are required for real-time plasma control, monitoring and machine protection. The diagnostic control system supervisor will calculate global plasma parameters from combinations of diagnostic data and feed these results into the plasma control system in real-time. A multi-CPU architecture for high performance and reliability is specified for this operation.

Recommendation. Generalize the “shot data file” to record continuous scientific data for the steady-state operation. Develop smart front end processors for real-time processing optimized for speed and robustness and a plan for correcting recording and processing errors or bad calibrations.

Environment, Safety and Health (ES&H)

ITER will be a complex, experimental device. ES&H issues are being taken into consideration during the design process in a satisfactory systematic approach. However, the following specific concerns have been identified during the review:

Public ALARA Issues. The 50 mSv criterion for no evacuation is a factor of five higher than the US DOE Fusion Safety Standard. Postulated accidents are expressed in grams of tritium released rather than off site doses using a reasonable hypothetical site.

Recommendation. 10 mSv should be the "no evacuation" criterion. Releases from postulated accidents should be defined as off site doses to assure that the "no evacuation criterion" is met.

Projected releases of airborne and waterborne tritium are high. Discharge of tritiated water vapor can impact tritium groundwater concentrations due to deposition. Waste water tritium concentrations would exceed the USEPA drinking water standard. Water Detritiation System concentration discharges up to 1,000 mCi/m^3 would exceed the 2 mCi/m^3 ITER limit for tritium-contaminated waste water.

Recommendation. More conservative limits should be established for tritium releases to the environment.

Worker ALARA Issues. Several examples indicate worker ALARA policy is not fully implemented: 1) high tritium concentrations in the HTS loops (1E9 -1E12 pCi/l) could impact maintenance: 2) surface contamination levels < 8Bq/cm^2 for "green" zones are about five times higher than the US Standard and about 50 times higher than used by the TFTR; and 3) the Detail Design Report assumes that dust effluent from vacuum vessel port openings would be < 0.3 g/a (removal of 10 small windows on the TFTR resulted in over a gram of dust).

Recommendation. These limits and assumptions which affect worker safety need to be re-evaluated for ALARA considerations.

Assumptions. Many assumptions and "to be determined" items exist in the NSSR-1 which is understandable based on the current level of design.

Recommendation. A process should be in place to capture assumptions and assure that they are addressed before operations begins.

Tritium Accountability. The NSSR-1 description does not provide enough detail to assure that safety limits are not exceeded and that tritium is not diverted for unlawful purposes.

Recommendation. Where and when tritium is to be measured and reconciled, with measurement tolerances, needs to be better defined to assure operations within the safety envelope.

Operations. The NSSR-1 description does not define the operational hardware and minimum staffing levels to assure an operational safety envelope. While the size and planned performance of ITER will be unprecedented for a fusion device, the operations knowledge that has been gained by large fusion experiments, including D-T experiments, is relevant to ITER ES&H issues and should be fully exploited.

Recommendation. Define operational criteria now so that its impact on machine availability can be determined. Existing fusion facility operating experience should be integrated into ITER design and planning.

Safety of Design. The following topics need to be further documented in the NSSR-1: 1) fire detection and suppression systems since some can have adverse affects on machine components and/or tritium systems; 2) the safety hazards of, and measures for using diborane (a highly reactive, explosive and toxic gas); 3) the handling of HTO in the condensates of HVAC Systems for potentially tritium-contaminated rooms; 4) the effect of NBIs pumping tritium onto their cryopumps including the HVAC impact (negative rather than atmospheric pressure in NBI cell); 5) the SL-2 peak ground acceleration of 0.2g with a return period of 10,000 years (benign when compared to sites considered to have low seismicity); 6) potential failures in the vacuum vessel pressure suppression system since the system appears to mitigate some accident scenarios; 7) determine whether loss of primary heat transfer systems functions result in radioactivity releases; and 8) use of SF6 in neutral beams for worker safety and tritium cleanup systems impact.

Recommendation. These issues should be analyzed and documented in the NSSR-1.

Waste Products. The production of significant amounts (~6,000 tonnes) of highly radioactive, fairly long-lived waste (>1 rem/hr contact dose for 100+ yrs), has implications for public acceptability of fusion.

Recommendation. The project should attempt to minimize waste products whenever possible.

APPENDIX D.VI

SUB-PANEL VI: MAGNET REPORT

Dr. V. Karpenko (Chairman)
Retired, Lawrence Livermore National Laboratory

Dr. Karl Krause
Lawrence Livermore National Laboratory

Dr. Richard J. Briggs *
SAIC Corporation

Mr. Tom Peterson
Fermi National Accelerator Laboratory

Mr. Charles Bushnell
Retired, Princeton Plasma Physics Laboratory

Dr. Clyde Taylor
Lawrence Berkeley National Laboratory

* - FESAC Member

Introduction/Summary

In preparation of this response, the panel has reviewed data supplied by the U.S. Home Team and material supplied by D. B. Montgomery. (A detailed list of the documentation reviewed may be found in Table 1). The panel is impressed with the detail presented in this material and feels that the design team has not only progressed on schedule, taking in account the requested directions of TAC, but reached a level of design that is compliant with the requested TAC directions and that will meet and perform to the engineering requirements of the General Requirements Document (GRD).

The panel concludes that while the present design is more than satisfactory for *the Detail Design Report* (DDR), that the full efforts of the design team should now be focused on the completion of: (1) the design details, and (2) critical R&D items. Both of these latter subjects are dealt with more thoroughly in the *questions* discussed below. In addition the panel has suggested a prioritized list for consideration by the US Home Team for action during the conclusion of the EDA and prior to *Construction Start* (See Table 2).

Answers to the Questions directed to the Panel

Q-1 **Is the presented engineering design sound and does it meet the requirements stipulated for the basic machine parameters?**

In the reviewed documents, the design at the *Detail Level* is well conceived, presented and laid out. The requirements are clearly stated and the criteria are reasonably specified. There are areas where further work of supplying details are required and the needed details are generally identified by the Design Team. We feel that the period of time remaining between now and final design to accomplish these tasks is sufficient and the effort required is well within the capabilities of the JCT and the Home Teams. The DDD's (with Appendices) clearly show that the stipulations of the GRD are being correctly taken into account.

The areas that we feel require further effort during the final design stage are:

Structure cooling - design and analysis (particularly the cool down and warm up scenarios)

Gravity supports (it would be gratifying if a *Table 2* type document could be prepared for these elements) including cooling design and analysis

PF Coil supports, mechanical, electrical and cooling details

CS Coil structure cooling

Coil Terminal boxes and Break boxes

CS Coil (+PF Coils) Hybrid Design Option (effort requested by TAC) -- Due to the increased level of complexity of this option, the loss in machine performance and the modest cost benefits, we strongly question the advisability of further work in this area.

Integration of load driven and thermal deflections along with manufacturing tolerances into the design with the resulting definition of the requirements for the correction coils.

Q-2 **Is there a RAM plan and is it being followed?**

The DDD clearly specifies the requirements and cross references the design information linked to each requirement. The document specifically addresses; 1.) Design, 2.) Assembly, 3.) Maintenance, 4.) Safety, 5.) Operations and 6.) Remote Maintenance. In addition, Appendix G of the TF DDD addresses safety (failure modes and effects and the design features that mitigate them.)

While there are some missing sections in the above presentations, in the main, they involve the detail design areas that are mentioned in *Q-1* above. The working of these *details* and the integration of them into the design features appropriately, we believe, can be accomplished by the design teams if they are allowed to focus on them during the time remaining in the EDA. In particular, to complete the RAM plan, they must address the quantified reliability/availability requirements found in the IDR.

<u>Q-3</u> **Is the R&D Program adequate and timely to support construction decisions? Are the results available at the end of the EDA?**

Model Coils:

The basic information on the Model Coil Program is found in a document that is now more than one year old. By digging into the present schedule information, it becomes apparent that there has been a non trivial erosion of the Model Coil schedules during the past year and significant reduction of smaller R&D programs to supply needed data to the design effort. It is our belief, based on the present situation, that data from the Model Coil Programs (fully verified and understood by all Parties) will probably not be available until mid to late in >1999 for the CS and mid to late in > 2000 for the TF. It is possible that complete cyclic data on all the proposed inserts in the CS Model Coil will be even later than > 1999! It is clear that information will not be available at the end of the EDA. This is a success oriented schedule. It does not allow for any significant problems with any of the coil tests. Any modification to coils and retest will result in a substantial delay to the schedule.

Conductor Strand:

Adequate performance of strand has been well established. We believe that it is critical to the program to place on the top level schedules a major milestone that commits to the purchase of 10% (approx. 100 tons) of the conductor strand -- a long lead item. This request results from the desire to not only address a critical path item on the project schedule but to properly prepare the manufacturers of the strand for the timely delivery of the material that will be required to meet coil production schedules.

Cable and Joint Testing:

In contrast to strand testing, there has been insufficient and very limited testing on full-scale cable and joints.

Tests on subsize cables indicate that adequate performance can be expected, although there are large variations in AC losses that are not completely understood; however, effects of compaction on AC losses of full-size cable need to be established to verify the present manufacturing procedures.

We recommend that maximum effort be expanded to test a number of full-scale joints with sufficient instrumentation to measure AC losses and critical current. Several copies of each of the various full-scale joint configurations must be tested.

At the end of the EDA, sufficient data should be available (and digested) on the cable manufacture and much more data should be available on the tests of full-scale joints in addition to the joints of the Model Coils.

Construction Decisions:

We believe that a *Release for Construction* milestone should be instituted and based upon the availability (and dissemination) of the final results of the Model Coils and inserts testing programs. The placement of such a milestone on the top level schedule would remove the present ambiguity found in the coil design-procurement-fabrication schedule lines and will assist in the overall schedule credibility. It is our further belief that such a milestone should allow for the release of the coil fabrication -- proceeding up to the point of tooling production and trials -- so that minimum schedule impact will occur from such an action.

<u>Q-4</u> Will ITER meet its stated performance objectives with the present engineering design?

We believe that the material presented in the present DDD's indicate that the design will meet the engineering performance of the GRD. To insure the proper completion of this design, it requires the total efforts of the JCT and the Home Teams. Further *side studies* should be avoided during the balance of the EDA.

<u>Q-5</u> Does the engineering design properly address concerns of safety, health and environment?

We believe that within the DDD's the concerns of the GRD for safety , health and environment have, and are being addressed. Each magnet system component has been classified into one of four Safety Important Classes. This classification is based on the functional importance of the safety component. Systematic failure identification methods are being used to identify faults that could threaten components with safety functions. Detailed fault analysis is being performed where there is a potential for damage to confinement barriers. This analysis (FMEA) should be expanded to include *component* safety. Both areas should continue to receive attention during the remainder of the design effort so that concerns in this area will be at an absolute minimum when site specific design work begins.

CRYOSTAT

Introduction

The Cryostat provides the vacuum for the superconducting magnets and forms part of the radiological secondary containment. It also provides decay heat removal when all vacuum vessel and in-vessel cooling system have failed.

Answers to the Questions directed to the Panel:

<u>Q-1</u> Is the presented engineering design sound and does it meet the requirements stipulated for the basic machine parameters

Document DDD Cryostat (N24DDD 396-11-22W1.4) has extensive design descriptions. This document shows a considerable progress that has been made since the IDR. Design is well supported by a comprehensive analytical effort.

<u>Q-2</u> Is there a RAM plan and is it being followed?

Although there are numerous references to reliability there is no comprehensive RAM plan. A Quantize Reliability/Availability assessment is needed to evaluate and guide the design. There are numerous bellows and penetrations into the cryostat. Special attention will have to be paid to their reliability. FMEA need to be initiated to evaluate available options.

Q-3 **Is the R&D Program adequate and timely to support construction decisions? Are the results available at the end of the EDA?**

There are no significant R/D tasks identified in a cryostat area. Numerous design studies has been identified by the design team to be accomplished to assure successful completion of the final design.

Q-4 **Will ITER meet its stated performance objectives with the present engineering design?**

The requirements are well understood. A sufficient level of integration has been achieved. There are no technical or fabrication reasons that this design cannot be built and meet the specifications.

Q-5 **Does the engineering design properly address concerns of safety, health and environment?**

The DDD on Cryostat in general and specifically tables listing design linkage to safety requirements and postulated initiating events layout a sound basis for a comprehensive safety analysis.

When design and analysis will be completed it will fulfill ES&H objectives.

Cryogenic System:

The only document supplied to us for review was the IDR. It was stated that the DDD was not available because it was in a process of being revised. We feel it will not be appropriate to review a system that is going through a major revision.

The following observation will be still valid if capacity of the cryo-plant modules is substantially greater than 12KW.

The scale of this cryogenic system is much larger than an existing 4.5K system. The Fermilab and Brookhaven (CBA) systems are each about 25 kW total at 4.5 K. However, Fermilab*s Central Helium liquifier coldbox is equivalent only to roughly a 12 kW refrigerator. HERA at DESY in Hamburg, Germany, has three coldboxes at 6500 W each, for about 20 kW total installed capacity at 4.5 K. The ten cryogenic plants for the SSC were foreseen to be about 7 kW plus 45 g/s at 4 K each, or equivalent to about 12 kW each at 4 K. The eight cryogenic plants for LHC at CERN will each be approximately 2.5 kW at 1.9 K and 5 kW at 4.5 - 20 K roughly equivalent to 15 kW at 4.5 K. So 25 kW is at the upper end of the size of liquid helium plants.

Therefore, we agree with the opening statement in 4.3.3.I that *The cryo-plant modules up to a capacity of 25k W can be manufactured using the current technology base.* But the ITER designers should be more aware that the *standard* large helium refrigeration plant to date is about half that size, so there may be some risk in terms of uncertain reliability in going to the larger sizes of components.

What circulating pumps and cold compressors will be used in this system? Some developments and testing might be required before reliable cold compressors and circulating pumps of the size required here are obtained. It might be worthwhile to contract with industry for some prototype development and testing of cold compressors and/or circulating pumps of the size required for ITER.

THERMAL SHIELDS (W.B.S. 2.7)

Answers to the Questions directed to the Panel:

<u>Q-1</u> **Is the presented engineering design sound and does it meet the requirements stipulated for the basic machine parameters**

Thermal Shields (TS) function is to separate all cryogenically cooled elements of Tokomak Machine from warm surfaces and limit loads to 7 Kw. The thermal and structural aspects of the TS have been well analyzed. There are references to experience with JET and Tore Supra in the documents so the experience base from projects appears to have been used, which should greatly increase one’s confidence in this design. It is multielement, highly interactive specially and functionally complex system. A good foundation was laid for detailed design that will support machine parameters.

Q-2 **Is there a RAM plan and is it being followed?**

There is no formed RAM plan. The reliability of this system needs to be addressed in a more formal way. Although the inaccessible components of VVTS will have 100% redundancy and are classified as permanent parts of the machine (RH class 3), that Reliability/Availability must be rigorously quantified. A RAM plan has to be developed and detailed. FMEA should be initiated at earliest possible date.

Q-3 **Is the R&D Program adequate and timely to support construction decisions? Are the results available at the end of the EDA?**

The emissivity is the parameter with greatest uncertainty under the operational environment and a long lifetime. A literature search of experimental data will be required to finalize design.

Q-4 **Will ITER meet its stated performance objectives with the present engineering design?**

This design has a solid analytical base and will meet design specifications when an emissivity uncertainty has been eliminated.

Q-5 **Does the engineering design properly address concerns of safety, health and environment?**

The safety requirements are not fully documented. The ES & H issues are in the initial stages of being addressed.

COIL POWER SUPPLY AND DISTRIBUTION (W.B.S. 4.1)

Introduction

The Coil Power Supply and Distribution System (CPSDS) provides the supply and control of electric power for superconducting magnets and supplies AC power to the Additional Heating Power Supplies (W.B.S 4.2). The depth and quality of the design is very impressive and mature. The team should be highly commended.

Answers to the Questions directed to the Panel:

Q-1 **Is the presented engineering design sound and does it meet the requirements stipulated for the basic machine parameters?**

Most certainly. The DDR power supply design is, in general, quite mature and the expected performance of the proposed system design is consistent with the requirements imposed on it by the basic coil parameters and ITER operating scenarios.

Q-2 **Is there a RAM plan and is it being followed?**

No, a formal RAM plan is not in place as yet. It is acknowledged that throughout the EDA process, RAM issues pertaining to the coil power systems have been addressed in a qualitative, high level performance context, i.e., identification of major failure modes, redundancies, equipment ratings, etc. However, it is now time to begin a quantitative assessment as it may have surprisingly significant impact(s) on the design and implementation of the supplies. The argument that such an assessment may be premature because most of the ITER power supplies components are *one of a kind* or *lacking reliability data* is not compelling. While some of the components have not yet been prototype- or life-tested as part of the R/D program, much can be learned from an analysis based on engineering judgment and experience with similar components. The term *one of a kind* applies in a relative sense, i.e., ITER vis-a-vis other large experiments, but many of the power supply components must be produced in very large quantities in order to fully populate the system. Thus, much can be gained by advance knowledge of their anticipated or constrained failure rates. Comparison of a bottom-up calculation (even if partially based on assumptions) of reliability with the top down reliability allocations for the system is essential if major system, subsystem and/or component design perturbations or maintenance requirements are to be avoided in the future. A FMECA is a vital component of the reliability program and should not be delayed.

Q-3 **Is the R&D Program adequate and timely to support construction decisions? Are the results available at the end of the EDA?**

This needs to be developed: the Switching Network and Discharge Circuit components, including circuit-breaker for repetitive operation within PF power, system, bypass switches and circuit breakers, including-explosive-actuated switches, to be used in the Discharge Circuit for TF and PF coil protection.

C. Make switches for repetitive closing action

C. HV ... capability (after selection of ITER site)

Q-4 **Will ITER meet its stated performance objectives with the present engineering design?**

Yes. The coil power supply system, while immense in scope and taxing in its required input from the AC grid, does not pose severe challenges with regard to design, performance, manufacturability, or schedule relative to the rest of the ITER.

Q-5 **Does the engineering design properly address concerns of safety, health and environment?**

The safety requirements for the Coil Power Supply and Distribution System (CPSDS) are derived from the General Safety and Environmental Design Criteria (GSEDC), the General Design Requirements Document (GDRD) and the assigned safety functions. The identification of safety requirements is more than adequate of this phase of the design. There are minor issues at this point in the design process: Ecological constraints (as they pertain to generation of hazards) on the power system design are not addressed in the DDD. For example, are there constraints/limitations on toxic coolants, insulation additives, fire protection systems, etc.?

Table 1. Material Reviewed by the Sub Panel VII

ITER, Interim Design Report, dated July 12, 1995 (plus 10, 11x17 drawings)

1. DRAFT, A Technical Basis for the ITER Detail Design Report, Cost Review and safety Analysis (DDR)@, Chapter V, ITER Project Cost Estimate, dated November 12, 1996

2. ITER, A Detailed Design Report, Cost Review and Safety Analysis Dated December 1996, also marked AIC-11 AGENDA, Attachment 2.1

3. DRAFT, A Technical Basis for the ITER Detail Design Report, Cost Review and Safety Analysis (DDR)@, Chapter VI, ITER Construction Plan and Project Schedule, dated November 12, 1996

4. DRAFT, Chapter II, Section 4.2, Magnet Systems, dated November 12, 1996

5. DRAFT, Chapter II, Section 6.0, Tokamak Maintenance, dated November 12, 1996

6. Selected portions of TAC-2 to TAC-10 and TAC-11 minutes

7. A Conductor Design and Optimization for ITER@, N. Mitchell et. al, IEEE Transactions

8. TF DDD

9. TF DDD, App. A, A Conductor Design

10. TF DDD, App. A, Annex 1, A Selection of Jacket Materials for Nb_3Sn SC@

11. TF DDD, App. B, A Structural Design and Analysis

12. TF DDD, App. C, A Design Criteria

13. TF DDD, App. D, Electromagnetic Analysis

14. TF DDD, App. E, Assembly and Disassembly . . . Magnet Components

15. TF DDD, App. F, Manufacture of TF, PF & CS Coils & Mechanical Structure

16. TF DDD, App. G, A Superconducting Magnet System Safety Assessment

17. PF DDD

18. CS DDD

19. Structure DDD

20. List of Figures

21. Large 7 R&D Projects Document, Ver 0, January 1996

22. Package of 13 papers supplied by D. B. Montgomery

23. FESAC ITER Review Meeting, Jan. 21-24, 1997, notebook of handouts

24. A Segmentation of the ITER Central Solenoid, Design Study by the US Home Team, December, 1996 (received from J. Citrolo PPPL)

25. E-mail communications from D. B. Montgomery

26. ITER EDA Analysis Summary, Structural Analyses, P. Titus (USHT) TAC/JCT Informal Review 5/10-13/95

27. TF DDD, App. C, Annex 2, Conductor Data Base

28. Quarterly Progress Report, Executive Summary, US ITER Home Team , QPR-97-1, Oct. 1, >96 - Dec. 31, >96

29. DDD Cryostat Nov. 9, 96,WBS 2.4

30. DDD Thermal Shields, Nov. 7, 96, WBS 2.7

31. DDD Coil Power Supply and Distribution, Nov. 5, 96

32. Interim Report "Cryostat Thermal Shield Detailed Design", Aug. 96, Task 591 to 31 (D314)

33. ITER Cost and Manufacturing Feasibility Study by Lockheed Martin Corp.

Table 2. Prioritized Action Suggestions from the Coil Sub Panel to the U.S. Home Team

Most importantly, is the restoration of funding (to initially agreed to levels) for the balance of the EDA. This is crucial to the viability of the US position *at the table* where the future decisions will be made concerning the program.

1. Support of the CS Model Coil Program (to hold and/or <u>accelerate</u> the schedule.)

2. Support of an <u>expanded and accelerated</u> full scale conductor joint test program.

3. Metals R&D support (in small R&D items of material properties)

4. Insulation R&D support (in small R&D items of material properties.)

5. Support in the production of prototypical parts -- in particular the *breaks* (cryoline and feeders which should then be tested)

6. Support of the needed industrial costing studies of the final details

7. Support of developing the Criteria Document by reviewing , amplifying and backing up items that are in the present document.

8. Support of the administrative effort to draft a reasonable and viable procurement plan and the adoption of the needed procedures for material movement across our borders.

APPENDIX D.VII

SUB-PANEL VII: IN-VESSEL COMPONENTS

Dr. Samuel D. Harkness (Chairman) *
Westinghouse Science &
Technology Center

Dr. Tom J. McManamy
Oak Ridge National Laboratory

Dr. Tom Shannon
University of Tennessee

Dr. Dale Smith
Argonne National Laboratory

Prof. Don Steiner
Rensselaer Polytechnic Institute

* - FESAC member

The In-Vessel Component Subpanel focused on four major issues as discussed more fully in the body of our report. These are: 1) The Adequacy of the Nuclear Radiation Transport Calculations; 2) Potential In-Vessel Failure Modes and the Effects of Irradiation on Them; 3) The Dimensional Stability of the Design; and 4) The Approach Taken to Insure Remote Maintenance and Repair Capability.

While overall the In-Vessel systems have benefited from a great deal of innovative engineering, several areas of potential risk were identified that would benefit from a focused effort during the remainder of the Engineering Design Activity. The major points of concern identified by the Subpanel were:

1. Failure rate of blanket modules may be unacceptably high causing the machine to have low availability. An in-depth failure analysis that recognizes potential modes should be conducted. Attention should be paid to the rapid detection (including location) of leaks. Particular areas of concern include the copper to stainless steel bond on the first wall heat transfer surface, the many welded joints and the insulator integrity in the bolted module. The deleterious effects of irradiation on the mechanical properties of copper raise a significant concern for the extended performance phase.

2. The required dimensional tolerances for the location of in-vessel components are tight due to the need for magnetic field uniformity. The effects of temperature and magnetic forces have not been fully analyzed, especially in light of non-symmetrical port locations to determine if the required tolerances can be achieved.

3. The consequences of a failure in the remote maintenance system and the procedure for recovery should be analyzed to avoid a potential long delay once machine operations have begun.

4. The inclusion of ferritic-based blanket modules in the Extended Performance Phase is expected to be important to the qualification of a subsequent demonstration reactor design. The influence of two or three of these modules on the magnetic field uniformity should be addressed to insure that they will be able to be included. This should be done as part of an overall assessment that insures there is a logical qualification pathway between ITER and Demo that does not require the building of a separate major facility.

ADEQUACY OF THE NUCLEAR RADIATION TRANSPORT CALCULATIONS

(D. Steiner)

Accurate calculations of neutron and photon transport are essential for the success of ITER. Such calculations are required in order to determine the performance and life-time of the device components. Moreover, these calculations are also required to determine the operational and public safety aspects of ITER. The ITER Project has assembled a very knowledgeable group of individuals to carry out the required radiation transport evaluations. A Nuclear Analysis Group (NAG) has been formed and this group has been responsible for establishing the approach and methodology for the required radiation transport calculations. The NAG has developed a general purpose, three dimensional model to represent the machine, including for example, the toroidal and poloidal gaps between the blanket modules, the major ports and the manifolds running through the modules. The model also includes a very detailed representation of the divertor cassettes. The calculational codes which the group has been employing are state-of-the-art, including the Monte Carlo Code, MCNP, and the 3-D discrete ordinates code, TORT. Moreover, the group is using the most up-to-date nuclear data library, the FENDL-1 data library. The NAG provided me with several of its reports. In addition, I was referred to two papers presented at the recent Reno Meeting on Fusion Technology. One of these papers dealt with the three-dimensional neutronics and shielding analyses of the ITER divertor and the other dealt with the three-dimensional shielding analyses of the vertical and mid-plane ports in ITER. Both of these analyses were comprehensive, and demonstrated a thorough understanding of the very complex neutron streaming environment which exists in the machine. It is my assessment that the Nuclear Analysis Group consists of extremely competent people who are carrying out the necessary radiation transport calculations using the most up-to-date tools and nuclear data. The only issue regarding the "adequacy" of the calculations is that certain parts of the design are still evolving. For example, the shielding blanket design is still undergoing change and, thus, calculations performed on the current version of the shielding blanket may not adequately represent the characteristics of the shielding blanket design which emerges. Once the ITER design is frozen I am confident that the NAG can perform the radiation transport calculations, to the required level of accuracy.

POTENTIAL IN-VESSEL FAILURE MODES

(T. McManamy and D. Smith)

A key input to the analysis of potential in-vessel failure modes is the selection of materials and their associated properties in the ITER service environment.

Indeed, the materials requirements for in-vessel structural and shielding system for ITER are highly complex and quite severe. The ITER project has done a good job of identifying these requirements and in selecting reference materials for the in-vessel systems. In particular, the selection of solution annealed Type 316 austenitic steel as the primary shield/blanket structure and copper alloy as the primary heat sink material are considered appropriate choices. Two general concerns regarding the materials issues relate to (1) the continued changes in the design and materials selection in attempts to meet all design requirements and (2) the inadequacy of the materials data base to assure satisfactory performance. The project has identified the data base requirements and in most cases are developing the needed data base; however, sufficient resources have not been provided and in some cases the data will not be available by the end of the EDA. Several areas have been identified in which the effort should be substantially enhanced to provide the data needed in a more timely manner. Specific issues that should receive increased attention include the following:

- Degradation of the mechanical properties of the copper alloys.

  Copper alloys appear to be the correct choice for in-vessel heat sink applications, limiter and divertor. Preliminary data presented by the project indicate rapid loss of ductility of the candidate materials as a function of neutron fluence at projected operating temperatures. Increased effort is needed to evaluate the performance limitations under conditions of neutron damage including helium and hydrogen transmutation effects, cyclic heat loads, electromagnetically-induced mechanical loads during disruptions, and with high-velocity aqueous coolant.

- Reliability of copper/stainless steel/beryllium bonds.

  Very large surface areas must accommodate high cyclic surface heat loads during neutron irradiation. A data base on neutron irradiation effects on the performance of these bonds has not been presented by the project. Significant progress has been made in the manufacture and testing of small components; however, further effort is needed to evaluate the effects of irradiation on the integrity of the duplex structure under cyclic heat load conditions and in testing of larger components.

- Performance of weldments and rewelds after repair.

  Fabrication of the stainless steel first wall and shield will require extensive weld joints and the possibility for rewelding of manifolds after irradiation. Additional data are needed to assure satisfactory performance of these weld joints under thermal cyclic conditions after irradiation, and reweldments of irradiated materials under conditions projected for the shield-blanket manifolds. These effects include helium transmutation effects which could be significant since the joints may also be susceptible to aqueous stress corrosion effects and they are located in regions of significant neutron fluxes. Effects of bimetallic joints in the divertor manifolds also require additional attention.

- New materials incorporated.

  As the shield-blanket design has evolved, new materials, e.g., a high strength nickel alloy bolt and an electrical insulator to reduce electromagnetic loads during a disruption, have been incorporated into the design. The materials data base to support the integrity for these applications has not been provided by the project.

Beyond the need for an improved materials database, the Subpanel is concerned that the current Blanket Module attachment methods appears complex with many potential failure modes.

Blanket Module Design

The Blanket Module and attachment scheme should be evaluated from both a RAM (Reliability, Availability and Maintainability) and a FMEA (Failure Modes and Effects Analysis) perspective. The RAM analysis should address issues such as whether periodic inspection and tightening will be required, how thread damage would be corrected, and statistical evaluation of failure probabilities. The FMEA study should identify the major potential failure modes and evaluate the consequences. One question would be the consequence of a single insulation failure on a module bolt - would a current loop be formed which would result in excessive loads?

A final concern deals with the fact that a workable system for detecting and locating leaks has not been established. This is seen as an important component of an approach to insure machine availability.

Leak Detection System

It is recommended that leak detection methods and manifolding schemes should be developed to facilitate rapid detection and location of leaks.

THE DIMENSIONAL STABILITY OF THE DESIGN

(T. McManamy)

The tolerance requirements on plasma facing components specified in the IDR - General Design Requirements paragraph 5.5.1.3.2.1 and 5.5.1.3.2.2 as given below appear extremely difficult to meet.

“The limiters and baffles shall be installed so that they can be adjusted to be within +/- 3 mm of the corresponding magnetic surface, as defined in 2.2.4.5 (including ripples) at operating temperature. The primary wall shall be installed within +/-10 mm of the corresponding magnetic surface, as defined in 2.2.4.5 (including ripples) at operating temperature.”

“Edges of adjacent modules shall be aligned so to obtain a maximum radial step of +/- 2 mm.

One difficulty is that the current assembly plan as defined in DDD 2.2 completes assembly of the in-vessel components (blanket modules, the baffles and the limiters) prior to significant ex-vessel assembly and magnet cooldown and operation. The effects of dead weight loads, TF magnet cooldown, and magnetic loads will cause significant motion of the plasma facing surfaces, probably on the order of tens of millimeters. The draft DDD interface document between WBS 1.5 (the Vacuum Vessel) and WBS 1.4 the Magnet structures gives the room temperature location of the supports and loads but does not address motion due to cooldown or magnetic loads. While some of these can be predicted, there may also be some unexpected effects.

A second problem is the motion caused by heating and thermal growth of components during operation which could also result in motion on the order of 10 mm or more. A particular concern here would be any non-symmetric motion between the start and end of a discharge. Initial baking operations on the welded vessel could also cause dimensional changes.

A good assessment of fabrication tolerances for the divertor assembly was completed as reported in ITER/US/96/IV-DV-20 but it specifically excluded temperature effects.

Recommendations

A. Consideration should be given to a magnetic field mapping performed at cryogenic temperature prior to final installation of the plasma facing components. There may also be a need for some adjustment of the vacuum vessel support system to align the vessel axis with the magnetic vertical axis.

B. Consideration should be given to provide adjustment capability for the baffles and limiters.

C. The tolerance requirements should be reviewed to evaluate if some relaxation is possible to reduce cost and assembly difficulty.

D. Integrated structural models of the vacuum vessel, magnet and cryostat systems should be used to evaluate the displacements of the plasma facing surfaces due to operating conditions (temperatures and magnetic loads).

REMOTE HANDLING AND MAINTENANCE

(T. Shannon, M. Saltmarsh, T. McManamy, P. Spampinato,
M. Rennich and J. Haines)

Maintenance and repair of the International Thermonuclear Experimental Reactor (ITER) during its operating phase will require extensive use of remote handling and maintenance (RH/M) technologies. The project has recognized the importance of the remote handling requirements as a design driver, and has adopted appropriate strategies to deal with it. The ITER approach is to categorize components in one of four RH classes as shown in Table 1.

Table 1. RH Classifications

Class	Definition
1	Components that require scheduled remote maintenance or replacement
2	Components that do not require scheduled remote maintenance, but are likely to require unscheduled or very infrequent remote maintenance
3	Components not expected to require remote maintenance during the lifetime of ITER
4	Components that do not require remote maintenance or repair

For Class 1 and 2 systems the goal is to design all the required RH equipment during the EDA. Prototyping and mock-up of the two most critical tasks, divertor replacement and blanket maintenance, is being done in two of the seven large projects that comprise the bulk of the R&D during the EDA.

Elsewhere the objectives have been to design in hands-on capability whenever possible. Where appropriate, built in redundancy is used to extend expected component lifetimes. For in-vessel components in-situ repair is considered first, followed by exchange of a sub-component or complete component. Standardization and modularity have been used to reduce risk and cost.

The remote handling tasks and equipment fall into four general areas:

> The in-vessel components, including the divertor, blanket, heating/fueling, diagnostic systems, and associated RH equipment, particularly the in-vessel transporter and viewing/metrology systems;
>
> The ex-vessel components within the cryostat _ most notably the coils and structure, which will be designed for the full ITER lifetime;
>
> The RH transfer cask and transportation system which moves activated components from the vessel to the hot cell for repair or refurbishment, and back to the vessel for reinstallation; and
>
> The hot cell where repair and refurbishment of activated components is undertaken.

Most in-vessel components are RH class 1 and 2, and, as described above, the RH issues are being addressed in detail during the EDA.

The ex-vessel components within the cryostat are all in the RH 3 class. Access and repair will require warming up the system, a lengthy procedure. For these components the strategy is to design for the lifetime of the ITER, but to define procedures which would permit maintenance during the EDA to ensure that the possibility of repair exists.

A transportation system has been designed to move activated components in unshielded casks to the hot cell. Rail transporters and an elevator system are employed. This system has been designed to preserve personnel access to the pit and gallery, except during transfer of activated components when some areas must be evacuated.

The hot cell arrangements have not yet been designed in detail. They depend on the rate at which repair activities must proceed, which in turn depends on the lifetime and failure rates of the various components, and the balance between lengthy rework of activated components or increased volume of waste.

Comments and Recommendations

1. In-Vessel Maintenance

   The maintenance of in-vessel components has been an area of major uncertainty since the beginning of the ITER design activity. The need for totally remote handling within the toroidal vessel and the size and complexity of the blanket components has raised this to a high priority issue in the R&D program. Significant progress has been made both in design and in plans for development and testing. The component design and the maintenance equipment being developed is based on an extension of established concepts in use by the nuclear industry and research laboratories around the world. The ITER tokamak configuration has also evolved to provide a more practical access scheme with the use of the horizontal ports for maintenance of the blanket components. The design concept has been further improved by adopting a maintenance strategy based on a maximum use of in-situ repair and the use of modularity, standardization and segmentation.

   The design of the remote handling system has also evolved to what appears to be a more simple device using rail mounted transporters for the maintenance of blanket modules. These concepts will be demonstrated in the near term (97/98) in a large

R&D project. The combination of improved configuration for access, modular component design, the use of the horizontal rail system and finally the test program, provides a convincing argument that the maintainability goals will be well understood and hopefully met in the present design. It appears that the present design and R&D plan will result in in-vessel components that will be remotely handleable as required.

An issue for the rail mounted system that is not addressed in the documentation is the need to recover from a failure of the system itself. Analysis is required to determine the possible failure modes and a plan for recovery within an acceptable down-time. A recommendation for further analysis of this and other potential system failures is discussed in more detail below.

2. Overall RH/M Plan

The project has adopted an appropriate, staged strategy for addressing RH/M issues. The self-consistency of the strategy relies on the assumed lifetimes and maintenance frequencies for the various components. If these assumptions prove to be too optimistic, the ITER downtime could be unacceptably long. It is recommended that a reliability, availability and maintainability (RAM) analysis be performed to confirm that the assumed intervention frequencies are reasonable. This could further impact the design of many of the RH class 3 components, for which the repair procedures are likely to be extraordinarily time-consuming. In the case of the RH class 1 and 2 components the intervention frequency will impact the requirements on hot cell design, and perhaps the design of the port vacuum seals if significantly more than the assumed five openings should be required.

In the same spirit an analysis of failure modes and recovery procedures for RH/M operations should be done. Events such as the jamming of the in-vessel rail system during deployment or blocking of the single elevator by a disabled loaded transport cask may pose a difficult recovery problem. The combination of analysis to identify credible failure modes and development of appropriate recovery procedures should be given a high priority.

In addition to the two issues described above, a number of minor concerns surfaced during the review.

a) It is not clear if satisfactory solutions have been developed for alignment and structural attachment of the blanket module to the back-plate.

b) In the event of a water leak in a blanket module, is there a method to identify which blanket module should be replaced?

Finally, the importance of testing the RH/M procedures in various mock-ups before they are used in earnest is properly emphasized in the project documentation.

APPENDIX D.VIII

SUB-PANEL VIII: ITER COST AND SCHEDULE ASSESSMENT

Dr. Michael Knotek (Chairman) *
University of Washington State

Dr. Richard Callis
General Atomics

Dr. John Haines
Oak Ridge National Laboratory

Dr. Victor Karpenko
Retired, Lawrence Livermore National Laboratory

Dr. Michael Saltmarsh
Oak Ridge National Laboratory

Dr. John A. Schmidt
Plasma Physics Laboratory

Dr. L. Edward Temple
Argonne National Laboratory

* FESAC Member

The cost and schedule review responded to two specific questions:

1. Are the proposed cost estimates and schedules for the construction project and subsequent operations, exploitation and decommissioning credible, and are they consistent with the procurement methods and staffing methods recommended by the ITER Director? Focus on the methodology used to prepare the estimates.
2. Are there any cost effective opportunities for pursuing modest extensions of the current design features in order to enhance operational flexibility and increase scientific and technological productivity of ITER?

The bulk of the discussion focuses on the first question, and the second question was addressed through technical issues raised by the technical subpanels.

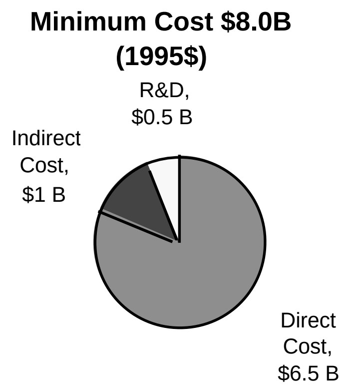
Summary Comments

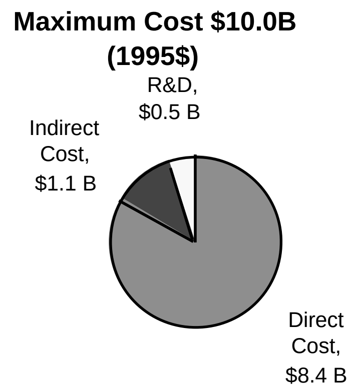
This review focused on the ITER Post-EDA and construction cost and schedule estimates in the DDR and supporting documents. Operations and decommissioning were not covered in any detail, and are not commented on. The cost and schedule development process used by the JCT was based on a detailed set of procedures and formats that facilitated a standardized and consistent cost and schedule estimate. It is the opinion of the review panel that the JCT has done a very disciplined and thorough job in gathering the complex data from diverse parties and then developing a self-consistent cost and schedule data base predicated on sound cost and schedule estimating methodologies. Estimates for components and systems are primarily based on industrial estimates from multiple parties, and have been extensively analyzed and processed to insure credibility, completeness and accuracy.

The DDR estimates exclude certain costs, including costs to be borne by the host (site, infrastructure, etc.) and the resources already spent on the CDA and EDA phases. To accommodate the practices of the various parties to the ITER, a contingency budget has not been included in the overall estimate, but provision for adequate funding, including contingency, will need to be properly accommodated in cost estimates for US participation.

This report includes potential cost impacts of issues and risks identified by the technical subpanels, and areas where the JCT should focus during the FDR preparation and during the post-EDA period. Generally, this is a success oriented plan, in that there is little or no budget or time allotted to accommodate problems. Additionally, the discipline which has been imparted to the project by the now departing administrative officer must be continued to guarantee further progress. Finally, an efficient management structure and a procurement system which takes maximum advantage of industrial competition is required to meet the aggressive cost and schedule goals of the project.

The cost for the ITER construction phase are estimated to range between $8.0 B and $10.0 B in 1995 $. The distribution of the costs are shown below.





Direct construction costs include all components, systems structures, buildings, materials, and construction labor to construct the complete ITER facility that would operate during the basic performance phase. Indirect costs include project management, procurement, engineering, support of construction, and preoperational testing / startup. R&D includes the cost of R&D scheduled, but not performed during the EDA (~116M 1995$) and R&D forecast as being needed during construction.

To meet this aggressive cost and schedule, a very disciplined and efficient management structure, which takes advantage of industrial competition will be needed.

ITER JCT METHODOLOGY

The methodology employed by the JCT to construct the estimate relies as much as possible on industrial estimates provided by the Parties. For many components, and for virtually all of the tokamak components, industrial estimates have been obtained from multiple Parties (herein to be understood as industries of those Parties) in preparation for the Interim Design Report (IDR). For some components, estimates were obtained from a single Party, and for buildings, diagnostics, and machine tooling they were internally generated by the JCT. The IDR Cost Estimate represented a bottoms-up estimate of almost every element of ITER. For the vast majority of elements, estimates were developed based on sufficient design details. For those minority of elements where sufficient design details didn't exist, a rough estimate either was prepared by the JCT or was included in the Allowance for Indeterminates (AFI). As the design progresses, these items will be estimated in detail, and the amount in AFI will progressively reduce to zero.

The JCT estimate includes the elements required for the basic performance of ITER. The IDR includes the estimates for construction, R&D and prototypes during construction, design after the end of the EDA, construction management, construction inspection and oversight, acceptance testing, preoperational checkout, and commissioning. It also includes estimates for the shared cost of operation and decommissioning. However, the estimate at this time explicitly excludes experimental testing during operation. For the Detailed Design Report, industrial estimates were not directly solicited, but adjustments in the costs of individual components and/or activities to reflect design development were made utilizing unit rates developed from new information since the IDR cost estimate.

To accommodate the different currencies, practices, and industrial indices of the ITER parties, the JCT developed a reasonable normalization procedure to arrive at the cost of each project element in 1989 dollars. The JCT then generally chose the lowest of the credible estimates as the cost of each item. In practice, an aggressive procurement process which takes full advantage of industrial competition must be employed to realize these costs, and make this a valid estimating process. This cost of each item in the construction cost estimate was reported by the JCT within a range of uncertainty. The uncertainty can be either positive or negative, and generally reflects the uncertainty in the cost due to the maturity of the design. The total positive uncertainty is 11% and the negative cost uncertainty is 15%. As the design progresses, it is anticipated that the uncertainty will reduce. The JCT does not provide a contingency to its estimate, because such a concept is generally not consistent with the estimating practices of the parties. In the US a contingency is normally applied to an estimate. Contingency is based on project requirements relative to the current state of the art, and on project uncertainties that could affect specific cost elements including potential technical, cost, and schedule changes. Provision for adequate funding, including contingency, will need to be accommodated in cost estimates for the elements the US will provide as its responsibility in participation during construction.

The items to be provided by the host of the ITER site, either from an existing infrastructure or by commitment of further resources, are not in the estimate. Items to be provided by the parties themselves, such as test blankets and their services are also excluded, although space is provided for the blanket modules. CDA and EDA costs have also been excluded from the estimate. Finally, the transportation costs in the DDR estimate only includes the export packing preparatory to export and shipping and transportation to the nearest point of export of the providing party. The Final Design Report should specifically address the transportation costs from the port of export to the ITER site.

Recognizing the exclusions enumerated below, the JCT cost and schedule estimates are quite complete. The JCT has indicated that a new bottoms-up comprehensive industry estimate of the ITER cost is beginning in support of preparation for the Final Design Report.

ASSUMPTIONS

The following assumptions have been made in creating the ITER DDR Cost Estimate:

That the four parties will share approximately equally in the costs for ITER. At present, it appears that the US and RF will participate at only much lower funding levels.

That the lowest credible cost estimate for a system from one of the parties is a reasonable basis for the ITER estimate, assuming that a competitive procurement process is used. This was a reasonable assumption, but now with projected lower funding from the US, the base ITER estimate may need to be adjusted upward to account for reduced USA participation.

That an effective and efficient management organization will be put in place for the construction of ITER. See below (in General Issues and Risks section) for more detail.

That international commercial competition will drive many system actual costs below the present estimates. This may not be fully achieved with much lower US and RF participation.

That parties will provide the requested funding profile on schedule and that the parties are committed to maintaining the proposed ITER construction schedule. The JA and EU parties seem to have a good record here. The US record, due to overall budget exigencies, is not as good.

That the post EDA R&D will be completed and successful prior to contracting for component manufacture. The R&D program is currently lagging due to shortfalls in funding.

That the management team will be staffed as rapidly as proposed, and that adequate local craft labor is available.

STRENGTHS AND WEAKNESSES

Some of the strengths of the estimate are:

That it proceeds from a well developed documented design basis.

That a detailed working schedule has been prepared.

That a detailed estimate has been assembled in a disciplined manner based on input from all parties.

That an extensive R&D program is being executed to address many system design uncertainties and establish component manufacturability before component fabrication begins.

The ITER JCT, through the Administrative Officer, adopted a disciplined approach to cost estimating, scheduling, and cost control. The discipline must continue with his departure.

Some of the weaknesses of the estimate are:

The DDR Estimate is not a "bottoms up" estimate. The DDR has been only incrementally adjusted relative to the IDR in areas where there are known changes. The next "bottoms up" estimate will be in the Final Design Report (FDR) due about one year from now.

Host site costs are not included anywhere in the DDR estimate. These costs cannot be estimated until the readiness of the accepted site is known. Thus the total cost of ITER has not been defined.

GENERAL ISSUES AND RISKS

The two most important elements leading to cost indeterminants are: (1) the management organization that is established by the parties for implementation of construction, and (2) the actual schedule achieved for construction. Both of these elements impact the efficiency of implementation which has a profound impact on costs. The implications are briefly described below.

Management Organization

The optimum organization for ITER construction is one which has single point leadership, decision making, and control of work; including engineering, procurement, quality, schedule priorities, and most importantly, resource allocation. There might be a general contractor reporting to the ITER Director, which would add this quality to the Directors management and procurement model. The extent to which this is achieved will have a profound impact on ITER's schedule and costs. During completion of the design in the post EDA period, a single central management must be able to optimize design choices to minimize the project cost. Similarly, if centralized ITER management can compete the hardware procurements for the participating countries, this will lower costs by taking advantage of industrial competitiveness. It is also important to provide continuing central oversight to deal with technical fabrication issues and to allocate resources to assure technical success and timely performance. The Director needs to have the ability to place some fraction of procurements as he sees fit, even if the currency source and the procurement source are different parties.

Schedule Performance

A careful analysis has been performed of the schedule for construction of ITER and a credible schedule has been developed which will enable ITER first plasma to be achieved by 2008. This schedule is critically dependent upon certain key actions: (1) selection of the ITER site, (2) establishment of an organization or legal entity to begin ordering long lead procurements by the end of the EDA, and (3) the implementation of an efficient organization to implement the work. Because ITER is planned as an international collaboration, there are many agreements that must be reached. Delays are possible in this process and it is therefore appropriate to assess the potential impact of such delays.

The present plan is to continue both design and R&D for a 2-3 year post EDA period. It was always planned that some of this work would continue into the construction period, but the amount of this work has increased because of resource shortfalls in 2 of the ITER parties during the EDA. The implication is that the present team is required to complete post EDA activities. The project completion will be delayed in direct proportion to any delays in site selection or initiation of long lead procurement, with a commensurate increase in cost, due to inflation, maintenance, and stretchout of the project team.

SPECIFIC ISSUES AND RISKS

The present cost estimate is based on the assumption that the design, fabrication, and assembly and installation of certain components important to safety, namely the vacuum vessel and the vacuum vessel pressure suppression system, the cryostat, and the primary heat transport system which is Section VIII of the ASME Code or equivalent, will be accepted by the Regulatory Authorities. While this position is well founded, there exists a small possibility that the regulatory authority would require that it be done to Section III of the ASME Code or equivalent. Should this occur, the cost of these components could significantly increase.

Uncertainties and risks attendant to the manufacturing of the tokamak components (first-of-a-kind) are being or will be removed by the R&D program which will be completed prior to procurement of the components. Delays in the R&D program will result in schedule slippage in placing procurements, or components will be procured with a higher level of risk.

There is one commodity delivery that is not under the control of the Construction phase project management team (but is under the control of the Parties). This one exception is the Nb3Sn conductor for the TF, CS, and two of the PF coils. The total quantity of strand needed for these magnets is 1200 tonnes. For this quantity to be delivered in the time required by the schedule, the capacity of the world producers of the Nb3Sn strand will have to more than triple. Strand producers have indicated that the increased production is achievable. Since this will involve all of the producers of Nb3Sn strand in the Parties, there is a question as to whether the other Parties can increase their production further if the US is limited in production by their limited contribution to ITER. While the additional increased production appears to be well within the means of the other Parties, this is an area which merits early focused attention. It is also possible that the Large Hadron Collider Project, and various superconducting RF accelerator projects will place an additional significant demand on the Nb production capacity.

Another issue is the Incolloy jacket material. The US is presently the sole provider of Incolloy. There will likely be a need for a second supplier; and it is anticipated that this will be accomplished by a licensing arrangement.

Due to initial conservative costing, there is an opportunity to experience some reduction in the costs of the buildings as the design progresses. The JCT has indicated that the FDR estimate will reflect the more mature design.

ISSUES IDENTIFIED BY THE TECHNICAL SUBPANELS

The FESAC technical subpanels provided a set of issues and risks which may have an impact on the cost and schedule of ITER. These have been grouped into two subcategories, physics and hardware.

Physics

Uncertainties in the physics performance of ITER could result in performance shortfalls that would require changes to the tokamak or its subsystems. Flexibility has been built into the design to accommodate such changes. Several such possibilities have been identified by the Physics sub-panel and the ITER project team. Examples identified by the sub-panel are:

(i) Deeper fueling penetration to allow operation above the Greenwald density limit. This may be achieved either by modifications to the pellet injector specification (high field launch or higher velocities) or by compact toroid injection (DDR Ch III, 3.2.3). Although cost impacts have not been estimated it is likely that only CT injection would result in significant changes.

(ii) Increased plasma heating power beyond the currently planned 100MW (DDR III 2.2.3) could be required in order to access the high confinement regime. The design can accommodate up to 100MW additional power for certain heating systems, which would increase costs by about 500M$ (95$)

(iii) An additional 50MW of lower energy neutral beams (80keV) has been suggested by the US in this review, to better control the plasma rotation. Such an option has not been considered by ITER. Additional costs for the subsystem itself might be in the range of 400M$, and some modification to the tokamak access ports may be required. If this system were needed in addition to the existing beams, quite extensive redesign would be required.

(iv) Reversed Central Shear (RCS) plasmas may require off-axis RF current drive. Analysis shows that 100 MW of ECH will be required to support RCS plasmas. This as a minimum will require an additional 50 MW of ECH power at an anticipated cost of $200M. If ECH is not one of the original implemented heating systems then the cost will double. This off-axis current drive may also be supported using Lower Hybrid Current Drive, which should have a higher current drive efficiency.

However, LHCD is not included in the present ITER program, and the cost to implement such a system can not be evaluated at this time.

(v) If the severity of plasma disruptions, runaway electrons, etc. is greater than the base line assumptions, more frequent intervention for repair and maintenance, and higher throughput for rework in the hot cells may be required. The cost impact is likely to be minor, but there may be a significant impact on device operations.

Many of the costs for increased power (heating) could be appropriately accommodated within the provisions for “capital improvement” in the proposed operating budgets for the basic performance phase, and would not impact construction costs.

Hardware

Operability

There are several operation related issues that if not resolved could impact cost or schedule.

(i) The design of diagnostics required for machine protection and plasma control appear to be lagging, with the concern that the appropriate diagnostics will not be available at startup. It is possible that as the design matures new R&D efforts may be identified. The cost impact in this area can not be defined at this time.

(ii) The modified backplate design with higher resistivity first wall increases the vertical instability growth rate and will negatively affect the plasma controllability. The dynamic control analysis of the new configuration must be carefully evaluated before the new backplate design is adopted. If this analysis shows the modified backplate is not acceptable, a redesign will be required, which could have a schedule and cost impact.

Divertor

The need to provide more flexibility to accommodate alternate divertor configurations is an issue that could potentially impact the project cost. Although a large amount of space has been allocated for the divertor, the flexibility of the reference PF coil system needs to be investigated to ensure that other reasonable configurations are possible. This flexibility could result in a cost increase for the PF coil system, but the schedule impact can be minimized by examining this issue before the end of the EDA.

Magnets

In order to achieve the proposed magnet production schedule in the light of the delays already encountered from funding shortfalls, it will be necessary to have TF manufacturing and cold testing done at two facilities. This will result in increased cost of $130M to cover extra tooling, cryogenic equipment, and additional duplicate manufacturing facilities as required.

Overall, the TF magnet manufacturing schedule calls for three shift, five day per week operation for a long period of time (3 years). This plan has an increased risk, since very little leeway exists to make up (weekends) for problems arising during manufacturing.

CONCLUSIONS

Generally the JCT has done an excellent job in pulling together a complex DDR, in a complicated multiparty management environment. The plan is aggressive and success oriented, and there are some technical issues that need to be considered in the FDR. Success will require continuation of the discipline shown in this process, and a management and procurement system capable of meeting these aggressive performance, cost and schedule goals.

APPENDIX D.IX

SUB-PANEL IX: FACILITIES

Mr. John Davis (Chairman) *
McDonnell Douglas Aerospace

Dr. Michael Saltmarsh
Oak Ridge National Laboratory

Dr. G. Hutch Neilson, Jr.
Oak Ridge National Laboratory

* - FESAC Member

1.0 Introduction and Scope

The construction of ITER will present more engineering challenges than encountered in any previous fusion experiment, primarily because from a facility and maintenance standpoint, it will begin to address the requirements necessary for the successful operation of any fusion reactor. This section briefly reviews the work done to date by the ITER project with regard to the facility layout, strategy for assembling the tokamak, and finally the movement and repair of the radioactive components extracted from within the vacuum vessel. In conducting this review, the panelists attempted to answer the following questions posed by FESAC:

- Is the relative position of buildings in regard to power supply, cryogenics, cooling, tritium handling, and waste treatment logical with regard to the Tokamak Services Building and do the buildings meet the specifications in the General Design Requirements Document (GDRD)?
- Is the tokamak assembly procedure realistic and is the approach presented capable of achieving the dimensional tolerances specified?
- Are the remote handling procedures, equipment, and facility requirements logical and capable of meeting the maintenance requirements specified in the GDRD?

2.0 Facilities Overview

Detailed information on the general layout of the buildings and rationale for the building placement are described in Section 6.2 of the ITER Detailed Design Document (DDD). In general the facilities layout consists of five contiguous buildings with the Tokamak Hall & Pit located in the center, the Tritium Building on the east side, the Electrical Termination Building to the west, the Assembly Hall to the south and the Laydown Hall to the north. On top of the Tritium Building is the Plant Gaseous Effluent Stack. This welded stack is at an elevation high enough to ensure that the gaseous release will be above the wake of the highest nearby structures (~120 m above grade). On the east and west sides of the Assembly Hall are the Tokamak Services Buildings, and at the east side of the Laydown Hall are Hot Cell and Radwaste Buildings. In general, the various buildings have a logical placement and appear to meet the requirements laid out in the ITER GDRD.

An interesting feature of the buildings is the decision to locate part of the Tokamak below grade to reduce shielding costs. To accomplish this, they are proposing constructing a pit 50 m deep with a diameter varying between 66 m and 88 m, into which the 36.48-m-diameter cryostat is placed. Surrounding the cryostat are the diagnostics, hardware for plasma heating, plasma fueling equipment, primary heat transfer components, superconducting magnet terminal boxes, remote maintenance equipment, and a large number of service penetrations. In looking at the size of this pit, the obvious question is the feasibility of constructing something so technically sophisticated on this large a scale. Discussions with ITER personnel and others familiar with large construction projects indicate that structures of similar diameter or depth have been fabricated, but not in this combination. Accelerators for high-energy physics are examples of high-technology systems on a comparably large scale.

The Tokamak Service Buildings do not have a general tritium confinement function; however, the Heat Transport System vaults and the vault-to-vault connecting ducts within the Tokamak Pit serve as a confinement barrier. The primary coolant systems, which contain tritiated water, have confinement systems built around them. The supporting equipment, which can contain tritium, has a separate (higher velocity) ventilation system at negative pressure relative to the rest of the building to prevent leakage. This approach should prevent any small leaks from the confinement systems or chronic releases during maintenance when confinement is breached. The tritium in the Tritium Building is isolated from workers and the public by two strong confinement barriers, the equipment itself, and secondary confinement boxes. The building also acts as a confinement barrier for the water detritiation system, isotope separation system, and vacuum pumping areas. For areas which have glove box operations, the buildings provide rooms surrounding the secondary confinement boxes which can be isolated from the rest of the building. When isolation is actuated, the air in the room is exhausted to the Plant Gaseous Effluent Stack (PGES) via an air detritiation system. Portions of the building function as a ventilation path to the PGES. The basic tritium handling system and release scenarios have been reviewed by the U.S. ITER Home Team. In presentations at the January 1997 FESAC meeting, they indicated that the tritium system and clean up system are acceptable from their view point. Based on the experts' opinions, it does not appear that there are any outstanding issues regarding the tritium facility.

With all facilities handling radioactive material, support buildings need to be provided to handle both the solid and liquid forms of radioactive waste. In addition, non-radioactive waste that may be toxic to the environment, such as machine oils, will need a facility to handle this class of waste. A description of these facilities can be found in Section 6.2 of the DDD, and a detail of the plan to isolate and separate the waste from the various buildings within the ITER complex can be found in Section 6.3 of the DDD. In general, the waste handling facility not only has space for the storage of components, but also provides space for waste processing, supporting laboratories, component maintenance, and equipment storage. The building is designed to bear the dead weight and vibration loads caused by components and the processing of materials in addition to the loads caused by installing and transporting components. These buildings are capable of withstanding seismic, wind, snow, tornado, and wind-generated missile loads. The three buildings provided for waste handling are the radwaste processing building, radwaste storage building, and the support personnel building. The radwaste processing building is close to the hot cell building. The support personnel building is located between the hot cell building and tritium building. Again in looking at the layout as well as reviewing information provided by the U.S. Home Team members, the facilities and approach appear to be well thought out, without any obvious faults. A key issue is to tailor the facility requirements to the host countries requirements, which cannot be done until a site is selected. However as a generic approach to facility layout, the documentation provided by ITER is adequate.

3.0 Machine Assembly

The ITER project has developed a strategy for assembling the machine. This is an important step that serves several purposes as far as the design process is concerned:

- To determine the requirements on assembly tooling.
- To determine the cost and schedule for the assembly phase.
- To demonstrate that the machine can be assembled to meet the demanding tolerance requirements.

The project's main strategic objective in developing its assembly procedure has been to satisfy the requirements for a tight tolerance on the first wall position relative to the magnet axis (the vertical axis of symmetry of the tokamak), which itself must be established during assembly. These tolerance requirements are summarized as follows:

- The limiters and baffles are aligned within ±3 mm of the toroidal field lines (including ripple effects) at operating temperatures.
- Elsewhere, the first wall is aligned within ±10 mm of the toroidal field lines at operating temperatures.
- Edges of adjacent blanket modules are aligned to obtain a maximum radial step of ±2 mm.

In addition, adjacent divertor modules must be aligned to within ±2 mm of each other. This will rely on the accuracy of the divertor support rails and the divertor cassette support pads, which is to be achieved in part by machining customized sections of the support rails during the assembly process.

Consistent with this objective, an assembly philosophy has been adopted which features the application of dimensional control and correction at each stage. Specific objectives arising from this philosophy are to:

- Minimize the accumulation of deviations.
- Control, predict, and anticipate the distortion due to welding operations.
- Detect and partially correct deformations which do occur during assembly.
- Mitigate the effect of manufacturing tolerances, which are large compared to the tolerances on the final position of components.
- Permit the re-adjustment of the position of the components relative to a common reference axis after completion of their assembly.
- Permit the re-adjustment of the position of the first wall once the position of the machine's magnet axis is known.

Specific measures taken to realize these objectives are:

- Use of an optical metrology system to provide accurate and repeatable position measurements as the principal tool for achieving accurate positioning accuracy.
- Use of special welding procedures, including surveys and adjustments, to control weld distortion.
- Use of surveys and adjustments to align all components and supports relative to a single reference.
- Use of customized shims between component and support interfaces to avoid build-up of deviations.

- Preassembly and welding of the backplate independent of the vacuum vessel, and provision for adjusting its shape and position during installation.
- Provision for moving the entire vacuum vessel assembly relative to the toroidal field if necessary.

The main phases of the assembly process, which takes a little over four years, are:

- Installation of the toroidal-field magnets, vacuum vessel, and backplate. This includes the assembly of the lower cryostat and lower poloidal-field coils, assembly of the prefabricated toroidal field coil-vacuum vessel sector modules on the machine base, subassembly of backplate sectors with some of their attached first-wall modules, and assembly of these sectors in the tokamak.
- In-vessel assembly. This includes the final positioning and closure welding of the vacuum vessel and backplate, installation of the backplate supports, divertor, remaining first-wall modules, divertor cryopumps, cooling lines, and ports.
- Ex-vessel assembly. This includes assembly of the upper cryostat, upper poloidal-field coils, and central solenoid; activation of the toroidal-field out-of-plane supports and central solenoid vertical preload; and installation of magnet leads and cooling pipes.
- Preparation for commissioning. This includes comprehensive test programs and possibly some remaining assembly procedures to be completed prior to ITER commissioning.

4.0 Remote Handling and Maintenance

Maintenance and repair of ITER during its operating phase will require extensive use of remote handling and maintenance technologies. The project has recognized the importance of the remote handling requirements as a design driver and has adopted appropriate strategies to deal with it. The ITER approach is to categorize components in one of four remote handling classes as shown in Table 1.

Table 1. Remote Handling Classifications

Class	Definition
1	Components that require scheduled remote maintenance or replacement
2	Components that do not require scheduled remote maintenance, but are likely to require unscheduled or very infrequent remote maintenance
3	Components not expected to require remote maintenance during the lifetime of ITER
4	Components that do not require remote maintenance or repair

For Class 1 and 2 systems the goal is to design all the required remote handling equipment during the EDA. Prototyping and mock-up of the two most critical tasks, divertor replacement and blanket maintenance, are being done in two of the seven large projects that comprise the bulk of the R&D during the EDA.

Elsewhere, the objectives have been to design for hands-on capability whenever possible. Where appropriate, built-in redundancy is used to extend expected component lifetimes. For in-vessel components in-situ repair is considered first, followed by exchange of a sub-component or complete component. Standardization and modularity have been used to reduce risk and cost.

The remote handling tasks and equipment fall into four general areas:

- The in-vessel components, including the divertor, blanket, heating/fueling, diagnostic systems, and associated remote handling equipment, particularly the in-vessel transporter and viewing/metrology systems.
- The ex-vessel components within the cryostat, most notably the coils and structure, which will be designed for the full ITER lifetime.
- The remote handling transfer cask and transportation system, which moves activated components from the vessel to the hot cell for repair or refurbishment and back to the vessel for reinstallation.
- The hot cell, where repair and refurbishment of activated components is undertaken.

Most in-vessel components are remote handling Class 1 and 2, and as described above, the remote handling issues are being addressed in detail during the EDA.

The ex-vessel components within the cryostat are all in the remote handling Class 3. Access and repair will require warming up the system, which is a lengthy procedure. For these components the strategy is to design for the lifetime of ITER, but also to define procedures which would permit maintenance during the EDA to ensure that the possibility of repair exists.

A transportation system has been designed to move activated components in unshielded casks to the hot cell. Rail transporters and an elevator system are employed. This system has been designed to preserve personnel access to the pit and gallery, except during transfer of activated components when some areas must be evacuated.

The hot cell arrangements have not yet been designed in detail. They depend on the rate at which repair activities must proceed, which in turn depends on the lifetime and failure rates of the various components and the balance between lengthy rework of activated components or increased volume of waste.

5.0 Conclusion, Observations, and Recommendations

An objective in laying out the facilities of ITER was to try to avoid, wherever possible, the crossing of different services such as electrical power, cooling water, and waste handling. To achieve this objective, the ITER project placed the tokamak in the center, with the various support buildings radially located to the north, south, east, and west. This strategy is logical and appears to have achieved the objective. A lot of time and thought went into designing the tritium handling and waste treatment facilities. In general, they appear to have built upon the experience gained from fusion experiments that handle tritium as well as fission reactors, which must deal with radioactive waste streams. Within the time constraints allocated to this review, the ITER project appears to have done an excellent job in both of these areas.

The decision to locate the tokamak in a pit is an interesting approach. This approach has been proposed in some early inertial and magnetic conceptual reactor designs but not with the level of detail accomplished by the ITER project. In design of the pit, consideration needs to be given to both seismic conditions and ground water levels. The seismic requirements are essentially site specific but the ITER project did look at generic requirements and have proposed approaches for load isolation which appear reasonable.

Observation: There is no discussion on handling ground water or protection against leakage.

ITER's assembly tolerance requirements are extremely demanding. It is not obvious that they can be realized in practice, but the approaches being taken will probably do about as well as can be done based on the current state of knowledge. Their realization will depend on having a capable optical metrology system, tooling to position heavy components accurately, and adequate support structures to keep components from shifting after assembly. These are all receiving appropriate consideration but the capabilities will need to be demonstrated in advance to be convincing. As noted by the project, R&D will also be needed to improve estimates of weld distortion of large vacuum vessel and backplate subassemblies. These measures will not only enhance confidence in being able to meet the tolerance requirements, but are also expected to benefit the assembly cost and schedule. Moreover, much remains to be done in the area of tool development. We consider these R&D activities to be crucial for meeting the assembly tolerance objectives.

Recommendation: Fundamentally, the tolerance requirements are derived from requirements for uniform distribution of power and particle fluxes on plasma-facing surfaces under operating conditions. Appropriate controls (i.e., in design, surveys, and adjustment procedures) must be established to ensure that the machine assembly accuracy is maintained in the transition from assembly conditions to operating conditions.

Observation: While the project has given due consideration to first-wall alignment requirements, another class of tolerance requirements—field errors—has been overlooked in the assembly planning. These are specified in Requirements 2.2.3.6 and 5.2.6.1 of the GDRD. The GDRD would be expected to drive requirements for detection of field errors, measurement and adjustment of coil positions, and procedures for field-error compensation during assembly, but such requirements are not identified. Installation of the field error compensation coils is neglected in the assembly procedures. Some operations might occur in the Preparation for Commissioning phase, which is as yet undeveloped. Even so, the pertinent requirements should be identified now.

Recommendation: Requirements pertaining to the realization of field-error tolerances should be identified in the Assembly plan. The impact on assembly tooling, procedures, cost, and schedule should be assessed.

The project has adopted an appropriate, staged strategy for addressing remote handling and maintenance issues. The self-consistency of the strategy relies on the assumed lifetimes and maintenance frequencies for the various components. If these assumptions prove to be too optimistic, the ITER downtime could be unacceptably long.

Recommendation: It is recommended that a reliability and maintainability analysis be performed to confirm that the assumed intervention frequencies are reasonable. This could further impact the design of many of the remote handling Class 3 components, for which the repair procedures are likely to be extraordinarily time-consuming. In the case of remote handling Class 1 and 2 components, the intervention frequency will impact the requirements on hot cell design and perhaps the design of the port vacuum seals if significantly more than the assumed five openings should be required.

Recommendation: In the same spirit, an analysis of failure recovery procedures for remote handling and maintenance operations should be done. Events such as the jamming of the in-vessel rail system during deployment or blocking of the single elevator by a disabled, loaded transport cask may pose a difficult recovery problem. The combination of analysis to identify credible failure modes and development of appropriate recovery procedures should be given a high priority.

Finally, the importance of testing the remote handling and maintenance procedures in various mock-ups before they are used in earnest is properly emphasized in the project documentation.