GEORGIA INSTITUTE OF TECHNOLOGY — OFFICE OF CONTRACT ADMINISTRATION

PROJECT ADMINISTRATION DATA SHEET

E25-621

[X] ORIGINAL [ ] REVISION NO. ____

Project No. ~~E-26-601~~ (Follow-on to E-26-612) GTRI/GIT DATE 3 / 14/ 84

Project Director: W. M. Stacey School/~~Lab~~ Nuclear Engr.

Sponsor: Dept. of Energy, Oak Ridge Operations, Oak Ridge, TN

Type Agreement: Grant No. DE-FG05-84ER52107

Award Period: From 1/1/84 To 12/31/85 (Performance) 3/31/86 (Reports)

Sponsor Amount:	This Change	Total to Date
Estimated:	$ 240,000	$ 240,000
Funded:	$ 115,000	$ 115,000 (Est. through 12/31/84

Cost Sharing Amount: $ None Cost Sharing No: N/A

Title: Support of International Workshop on the Next Major Tokamak Experiment

ADMINISTRATIVE DATA

OCA Contact William F. Brown X4820

1) Sponsor Technical Contact:

Dr. A. L. Opdenaker
Office of Fusion Energy
Office of Energy Research
Dept. of Energy
Washington, D.C. 20545

2) Sponsor Admin/Contractual Matters:

Ms. Joyce Carringer, Contract Mgmt. Br.
Procurement & Contracts Div.
Dept. of Energy
P. O. Box E
Oak Ridge, TN 37831
(615) 576-7564

Defense Priority Rating: N/A

Military Security Classification: N/A

(or) Company/Industrial Proprietary: N/A

RESTRICTIONS

See Attached --- Supplemental Information Sheet for Additional Requirements.

Travel: Foreign travel must have prior approval — Contact OCA in each case. Domestic travel requires sponsor approval where total will exceed greater of $500 or 125% of approved proposal budget category.

Equipment: Title vests with None proposed or anticipated

COMMENTS:

This is a new grant to continue the effort previously authorized by a contract (E-26-612) which ended 12/31/83.

RECEIVED MAR 1984 Office of Contract Reports

Sponsor I.D. #02.141.002.84.002

COPIES TO:

Project Director (Stacey)
Research Administrative Network
Research Property Management
Accounting
Procurement/EES Supply Services
Research Security Services
Reports Coordinator (OCA)
Research Communications (2)
GTRI
Library
Project File
Other

12/88

GEORGIA INSTITUTE OF TECHNOLOGY
OFFICE OF CONTRACT ADMINISTRATION

NOTICE OF PROJECT CLOSEOUT

Date 6/23/89

Project No. E-25-621 Center No. Q5296-1A1

Project Director W. M. Stacey, Jr. School/Lab ME

Sponsor U. S. Dept. of Energy, Oak Ridge, TN

Contract/Grant No. DE-FG05-84ER52107 GTRC ___ GIT XX

Prime Contract No. ___

Title International Tokamak Reactor (INTOR) Workshop

Effective Completion Date 12/31/88 (Performance) 3/31/89 (Reports)

Closeout Actions Required:

___ None
___ Final Invoice or Copy of Last Invoice
X Final Report of Inventions and/or Subcontracts
X Government Property Inventory & Related Certificate
___ Classified Material Certificate
X Release and Assignment
___ Other ___

Includes Subproject No(s). E-26-601/Stacey/ME

Subproject Under Main Project No. ___

Continues Project No. ___ Continued by Project No. ___

Distribution:

_ Project Director
_ Administrative Network
_ Accounting
_ Procurement/GTRI Supply Services
_ Research Property Management
_ Research Security Services

X Reports Coordinator (OCA)
___ GTRC
X Project File
2 Contract Support Division (OCA)
___ Other ___

Georgia Institute of Technology
A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
SCHOOL OF NUCLEAR ENGINEERING AND HEALTH PHYSICS
ATLANTA, GEORGIA 30332

February 1, 1984

Report on Foreign Travel

Grant No. DE-FG05-84ER-52107

Traveler - Weston M. Stacey, Jr. WMS

Project No. E-26- 601

Site Visited - INTOR Workshop, Vienna
January 16-27, 1984

Persons Contacted - INTOR Workshop participants from USSR, EC and Japan

The traveler attended Session VIII of the Phase 2A, Part 2 INTOR Workshop in Vienna January 16-27, 1984. Each country (US, Japan, EC, USSR) presented detailed contributions (the Japanese contributions were particularly notable in their thoroughness) on several critical technical issues under consideration by the Workshop. These contributions were discussed at length and provided the basis for defining future home tasks to be performed for the next Workshop session May 21-June 1. Detailed plans for a worldwide data base assessment were formulated. A summary of the status of the INTOR Workshop work was prepared and is included as Attachment 1 to this report. The organization and participants in this session are indicated in Attachment 1.

It is the general opinion of the traveler that this Workshop session was successful in forwarding the objectives of INTOR to focus worldwide effort on the most vital technical problems facing the next generation of tokamak experiments.

Enclosures: Attachments 1-3

Attachment 1

TASKS, ORGANIZATION, SCHEDULE AND PROGRESS

INTOR WORKSHOP

PHASE 2A, PART 2

JULY 1983 - JUNE 1985

Presented to IFRC by INTOR Steering Committee

January 1984

PHASE 2A, PART 2 - INTOR WORKSHOP ORGANIZATION

		Chairman	EC	Japan	USA	USSR
CRITICAL ISSUE	A. Impurity Control and First Wall	Post	Harrison Schiller - VC Leger	Fujisawa Seki Kobayashi	Post Mattas	Pistunovich Serebrennikov
	B. RF Heating and Current Drive	Engelmann	Engelmann	Miyamoto	Colestock	Shatalov Pistunovich
	C. Transient Electromagnetics	Kostenko	Salpietro	Iida Ueda	Tarrh Peng	Kostenko Sadakov
	D. Maintainability	Tomabechi	Farfaletti-Casali Salpietro Chazalon	Tomabechi Iida Honda	Spampinato Shannon Baker	Shatalov Kolbasov Sadakov
	E. Technical Benefit	Knobloch	Knobloch	Tomabechi Kobayashi	Shannon	Kolbasov
DISCIPLINARY	F. Physics	Pistunovich	Engelmann Harrison	Fujisawa Miyamoto Kobayashi	Post Peng Colestock	Pistunovich
	G. Engineering	Shannon	Farfaletti Casali Salpietro Knobloch	Seki Honda	Shannon Spampinato Tarrh	Kostenko Sadakov
	H. Nuclear	Tomabechi	Chazalon Schiller Leger	Iida Ueda Tomabechi	Baker Mattas	Shatalov Serebrennikov Kolbasov
Steering Committee		Mori	Grieger	Mori	Stacey	Kadomtsev

1/23/84

B. RF HEATING AND CURRENT DRIVE

TASKS: Analysis of physics and engineering aspects of rf heating, current drive and plasma initiation. Collection of experimental data; comparison of calculational models; development of launcher design concepts; etc.

PROGRESS:

1. ICRF (reference bulk heating method 85 MHz)
   a. Experiments continue to show good heating results ($\sim$3.3 MW PLT); some concern with impurity production.
   b. Good progress in code development. Benchmark calculations in progress.
   c. Launcher designs in progress on several concepts.

2. LHR
   a. Current ramp-up to $\sim$250 kA in PLT and 20 kA in JIPPT-IIU.
   b. Current maintenance at 200 kA, $n = 10^{14}/cm^3$, w = 4 GHz in ALCATOR. ($\eta \simeq .1$ A/W)
   c. Plasma breakdown and current ramp-up to 100 kA at $n = 10^{12}/cm^3$ in PLT.

3. ECRH (reference start-up assist method - 150 GHz)
   a. New results from T-10 (500 kW, w = 86 GHz, $\Delta T_e^{\circ} = 1$ KeV) and D-III (250 kW, $\eta \simeq 4$ ev/kW)
   b. Work in progress to improve modeling.

D. MAINTAINABILITY

TASKS: Re-evaluation of design requirements to enable personnel access 24 hours after shutdown. Development of a totally remotely maintained design concept. Comparison of "limited personnel access" (present INTOR) and "totally remote" design concepts - costs and engineering design impact.

PROGRESS:

1. Requirements for tritium containment, cleanup and atmospheric recovery systems identified, and definition of concepts initiated.

2. An initial all-remote design concept has been identified.

3. Time-dependent calculations indicate lower rates of tritium permeation into coolant.

II. DATA BASE ASSESSMENT

A. TASKS

1. Collect and compare the scientific and technical data relevant to INTOR which are being produced worldwide.
2. Identify the capability that exists in present and future facilities to extend that data base over the next 4 - 5 years.
3. Identify additional R & D that would be required to provide an adequate data base for the detailed design and construction of an INTOR-like machine.
4. Based on 1) - 3), make a judgement on the technical feasibility of initiating construction of an INTOR-like machine 4-5 years from now.

   (Note: This activity is essentially an updating of the Phase - Zero data base assessment.)

B. PROGRESS

1. Detailed outlines have been developed to guide the assessment activity.
2. Lists have been developed to collect information on existing data and on facility capability.

III. OTHER ITEMS

A. SPECIALISTS MEETINGS

1. A 2-day INTOR specialists' meeting will be held on MHD β-limits July 4 - 5, 1984, following the IAEA-sponsored International Conference on Plasma Physics in Lausanne.

2. The report on the IAEA technical committee meeting on Fusion Safety and Environment will be reviewed, when available, to determine if further action is needed in this area for INTOR.

3. The Nagoya Plasma-Surface Interactions Conference (May 1984) will be monitored by INTOR Workshop members.

4. The ANS Topical Meeting on Robotics and Remote Handling (April 1984) will be monitored by INTOR Workshop members.

B. The INTOR Workshop participants are prepared to present an INTOR session at the IAEA Plasma Physics and Controlled Nuclear Fusion meeting in London.

PART A—To be completed by traveler's administrative officer

Budget and Reporting Classification to be charged: _____
(See Appendix 1101, Part II)

(The category of a trip, either I, II, or III, is determined by the definitions set forth in Appendix 1501, Part VIII) TRIP CATEGORY: _____

PART B—To be completed by traveler

1a. NAME OF TRAVELER Weston M. Stacey, Jr. b. CITIZENSHIP USA	c. DATE AND PLACE OF BIRTH July 23, 1937; Birmingham, AL USA d. PASSPORT NUMBER (if available)
2a. HOME ADDRESS X-4 Royal Coach Apts. Atlanta, GA 30309	b. BUSINESS ADDRESS Georgia Institute of Technology School of Nuclear Engineering, Atlanta, GA 30
3a. EMPLOYER Georgia Institute of Technology Atlanta, GA 30332 USA	c. CONTRACT NUMBER DE-AS05-79ET-52049
b. ORGANIZATIONAL UNIT School of Nuclear Engineering & Health Physics	d. POSITION TITLE (including profession) Callaway Professor of Nuclear Engineering

4. PURPOSE OF TRAVEL—Include all pertinent background information leading to travel and attach copies of invitations and correspondence regarding travel to present papers, give speeches, or to attend conferences or symposia. Also identify by name and organization other DOE and contractor personnel who, to the traveler's knowledge, are going to the same destination at the same time as the traveler. In addition, specify nature and classification of information to be disclosed including titles of papers to be presented; nature of information to be obtained at each of the places to be visited and conferences to be attended and its relation to traveler's work. Travelers are responsible for obtaining clearances for papers or speeches when necessary. If more space is required, attach a separate sheet. NOTE: IF THIS INFORMATION IS CLASSIFIED BE SURE TO CLASSIFY THIS FORM APPROPRIATELY.

To attend session I, Phase 2A, part 2 (January 16-27, 1984) of the IAEA INTOR Workshop as US Steering Committee member.

rivacy Act Information Statement. Collection of the information is authorized by the Department of Energy Organization Act of 1977, P.L. 95-91. isclosure of the personal information requested is mandatory to support authorization for official travel to foreign countries, and to obtain a assport

he information furnished will be used by DOE to authorize travel and payment of travel expenses; by the Department of State to issue a passport; nd by the General Accounting Office to audit and verify the accuracy and legality of disbursements.

PART E—To be completed at DOE Field Organization and/or Laboratory

11. APPROVAL AND/OR REMARKS BY HEAD OF DOE FIELD ORGANIZATION OR DIRECTOR OF LABORATORY:

(Signature) *(Date)*

PART F—To be completed at Headquarters

12. CONCURRENCE AND/OR REMARKS OFFICE OF SAFEGUARDS AND SECURITY
(Note compliance with Chapter 2502)

(Signature and Title) *(Date)*

13. CONCURRENCE AND/OR REMARKS OFFICE OF INTERNATIONAL SECURITY AFFAIRS

(Signature and Title) *(Date)*

14. APPROVAL OR CONCURRENCE/REMARKS BY ASSISTANT SECRETARY FOR INTERNATIONAL AFFAIRS

(Signature and Title) *(Date)*

15. CONCURRENCE AND/OR REMARKS—OTHERS

(Signature and Title) *(Date)*

16. CONCURRENCE/REMARKS OF DIRECTOR OF DIVISION OR OFFICE

(Signature) *(Date)*

17. APPROVAL/REMARKS BY ASSISTANT SECRETARY/DIRECTOR

(Signature) *(Date)*

REPORT OF FOREIGN TRAVEL COST

1. Name of Traveler: Weston M. Stacey, Jr.

2. Name of Contractor: Georgia Institute of Technology

3. Grant ~~Contract~~ Number: DE-FG05-84ER-52107

4. Inclusive Dates of Travel: January 13-29, 1984

5. Estimated Cost of Trip Shown on Form ~~AEC-445~~ DOE-F-1512-1: $3514

6. Actual Cost of Trip Charged to AEC Funds: $3264.20

U.S. DEPARTMENT OF ENERGY

NOTICE OF ENERGY RD&D PROJECT

FORM DOE 538
(1/78)

APPROVED FOR USE BY
SMITHSONIAN SCIENCE INFORMATION EXCHANGE

FORM APPROVED
OMB NO. 38 R-0190

1. DESCRIPTIVE TITLE OF WORK: Perform Work in Support of the International Workshop on the Next Major Tokamak Experiment

2. PERFORMING ORGANIZATION CONTROL NUMBER: E-26-601

3. CONTRACT, GRANT OR PURCHASE ORDER NUMBER: Grant No. DE-AS05-79ET-52049

4. CONTRACTOR'S PRINCIPAL INVESTIGATOR/PROJECT MANAGER AND ADDRESS WHERE WORK IS PERFORMED

a. NAME (Last, First, MI): Stacey, Weston M., Jr. PHONE (404) 894-3714

b. BUSINESS ADDRESS: STREET School of NE&HP, Georgia Institute of Technology

CITY Atlanta STATE GA ZIP 30332

5. a. NAME OF PERFORMING ORGANIZATION

Georgia Institute of Technology (Organization) Nuclear Engineering & Health Physics (Department)

b. MAILING ADDRESS (If Different From 4B)

c. TYPE OF ORGANIZATION PERFORMING THE WORK (Enter applicable code from instructions). C | H | | |

6. SUPPORTING ORGANIZATION

a. DOE PROGRAM DIVISION OR OFFICE (Full Name): Office of Fusion Energy

b. TECHNICAL MONITOR (Last, First, MI): Stone, P. M.

c. ADDRESS (If Different from DOE Hqs)

d. ADMINISTRATIVE MONITOR (Last, First, MI): Carringer, Joyce C., Oak Ridge, TN

7. PROJECT SCHEDULE (a) START DATE January, 1984 (Month) (Year) (b) EXPECTED COMPLETION DATE December, 1985 (Month) (Year)

8. a. FUNDING OPERATING AND CAPITAL EQUIPMENT OBLIGATION (In Thousands of Dollars)

FUNDING ORGANIZATION(S)		APPROXIMATE CUMULATIVE PRIOR FISCAL YEARS	CURRENT FY		
1. DOE		$552,478	$115,000		
2.					
3.					

b. DOE BUDGETING AND REPORTING CLASSIFICATION CODE

9. DIRECT SCIENTIFIC AND TECHNICAL MANPOWER

	PROFESSIONAL	GRAD. STUDENTS	OTHER	TOTAL
NUMBER	1		1	2
EQUIVALENT PERSONYEARS	1/2		1/2	1

DETACH HERE BEFORE SUBMITTING

E-25-621

COVER SHEET

FOR TRIP REPORTS SUBMITTED TO THE

OFFICE OF ENERGY RESEARCH

Destination(s) and Dates for
Which Trip Report Being Submitted: London, England 9/11/84 - 9/21/84

Name of Traveler: Weston M. Stacey, Jr.

Joint Trip Report [ ] Yes

[x] No

If so, Name of Other Traveler(s): ---

Georgia Institute of Technology

A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA

SCHOOL OF MECHANICAL ENGINEERING

October 31, 1984

Please reply to:

NUCLEAR ENGINEERING AND
HEALTH PHYSICS PROGRAM
CHERRY EMERSON BUILDING
GEORGIA INST. OF TECH.
ATLANTA, GEORGIA 30332 U.S.A.

Report on Foreign Travel

Grant No. DE-FG05-84ER-52107

Traveler - Weston M. Stacey, Jr.

Project No. E-26-621 E26-601

A. The traveler attended the 10th International Conference on Plasma Physics and Controlled Nuclear Fusion Research, September 12 - 19, 1984, London England. He presented a paper, attended the technical sessions and engaged in conversations. The principal impressions he received were:

1. The experimental and theoretical results on β-limits in present tokamak experiments are converging to a significant degree. Extrapolation to future tokamak reactors is not favorable, implying a need to focus attention on ways to achieve higher values of β.

2. Energy confinement scaling in auxilliary heated tokamaks is far from understood. The data in present machines can be more or less fit by a wide range of empirical or semi-empirical relations. The favorable H-mode scaling has been obtained rather regularly in divertor experiments. Extrapolation to future tokamak reactors is satisfactory, if H-mode confinement obtains.

3. Impurity contamination is receiving increased attention. The experimental and theoretical evidence supporting a high-recycling poloidal divertor as an impurity control scheme is rather convincing.

B. The traveler attended the International Fusion Research Council (IFRC) meeting on September 20, 1984 to report on the INTOR Workshop. The IFRC was more-or-less unanimous in their opinion of the value of the Workshop and their wish to see it continued. There was a widespread feeling among the IFRC members that the β-limit issue should receive more attention in the next phase. The French representative stated his opinion that the relation of INTOR to the following DEMO should be defined more clearly. The US representative stated his opinion that the Workshop should find a new focus which involved more far-reaching innovations. These three points were referred to the INTOR Steering Committee for consideration in preparing a revised proposal for activities during the next phase.

DOE-N-1500.3

(Previous Editions are Obsolete)

REQUEST FOR APPROVAL OF OFFICIAL FOREIGN TRAVEL

PART A—To be completed by traveler's administrative officer

Budget and Reporting Classification to be charged: E-25-621
(See Appendix 1101, Part II)

The category of a trip, either I, II, or III, is determined by the definitions set forth in Appendix 1501, Part VIII) TRIP CATEGORY: _____

PART B—To be completed by traveler

a. NAME OF TRAVELER Weston M. Stacey, Jr. CITIZENSHIP	c. DATE AND PLACE OF BIRTH July 23, 1937, Birmingham, AL USA d. PASSPORT NUMBER (if available)
b. HOME ADDRESS X-4 Royal Coach Apts. Atlanta, Ga. 30309 USA	b. BUSINESS ADDRESS A Georgia Institute of Technology Atlanta, GA 30332
c. EMPLOYER Georgia Institute of Technology Atlanta, Ga. 30332 USA	c. CONTRACT NUMBER E-25-621
ORGANIZATIONAL UNIT School of Mechanical Engineering/Nuclear Engineering & Health Physics	d. POSITION TITLE (including profession) Callaway Professor of Nuclear Engineering

PURPOSE OF TRAVEL—Include all pertinent background information leading to travel and attach copies of invitations and correspondence regarding travel to present papers, give speeches, or to attend conferences or symposia. Also identify by name and organization other DOE and contractor personnel who, to the traveler's knowledge, are going to the same destination at the same time as the traveler. In addition, specify nature and classification of information to be disclosed including titles of papers to be presented; nature of information to be obtained at each of the places to be visited and conferences to be attended and its relation to traveler's work. Travelers are responsible for obtaining clearances for papers or speeches when necessary. If more space is required, attach a separate sheet. NOTE: IF THIS INFORMATION IS CLASSIFIED BE SURE TO CLASSIFY THIS FORM APPROPRIATELY.

Attend IAEA Conference on Plasma Physics and Controlled Fusion and to present a paper. Attend IFRC meeting to report on INTOR.

acy Act Information Statement. Collection of the information is authorized by the Department of Energy Organization Act of 1977, P.L. 95-91. losure of the personal information requested is mandatory to support authorization for official travel to foreign countries, and to obtain a sport

Information furnished will be used by DOE to authorize travel and payment of travel expenses; by the Department of State to issue a passport; by the General Accounting Office to audit and verify the accuracy and legality of disbursements.

5. PROPOSED ITINERARY (Account for all time from beginning and ending dates of travel. NOTE: IF INFORMATION IS CLASSIFIED, CLASSIFY THIS FORM APPROPRIATELY.)

DATES	LOCATION (Installation, city, country)	INDIVIDUALS TO BE CONTACTED	SUBJECT OF DISCUSSION	(Check one) CLASSIFIED	(Check one) UNCLASSIFIED
9/11 - 21/84	London		IAEA Conference		X

6a. TRAVELER'S "Q" OR TOP SECRET ACCESS AUTHORIZATION NUMBER AND DATE:

QX ERDA #SC-27393-TN Sept. 20, 1977

b. HAS DOE FORM DP-290 PURSUANT TO CHAPTER 2502 BEEN SUBMITTED: (Required only for travel to a Soviet-bloc country by an individual who holds a DOE "Q" or Top Secret Access Authorization)

☐ YES ☒ NO

c. IF NOT, WHY NOT?

N/A

7. ARTICLE OF AGREEMENT FOR COOPERATION UNDER WHICH DISCUSSIONS WILL FALL *(Information on the article may be obtained from local DOE Organization):*

8. SIGNATURE OF TRAVELER—By signing, the traveler acknowledges the obligation to file a trip report within 15 days of return to duty stati within 30 days for team trips.

W. M. Stacey, Jr. *(Signature)*

August 2, 1984 *(Date)*

PART C—To be completed by official responsible for travel funds

9a. ESTIMATED COST OF TRAVEL TO DOE

Transportation	$ 1,863.00
Per Diem and Miscellaneous	$ 900.00
Total	$ 2,763.00

b. IF PART OF COST OF TRAVEL IS TO BE PAID OR HAS BEEN REQUESTED FROM SOURCES OTHER THAN DOE, INDICATE SOURCE AND AMOUNT.

c. TRAVEL FUNDS ARE NOW AVAILABLE FOR THIS TRIP.

(Signature and Title) *(Date)*

PART D—To be completed by Traveler's supervisor

10. ENDORSEMENT AND REMARKS:

(Signature and title of Supervisor) *(Date)*

ENERGY RESEARCH FOREIGN TRAVEL REQUEST COVER SHEET

1. TRIP NO.: ------
2. HQ APPROVAL REQUIRED: Y
3. TRAVELER'S NAME: Stacey, Weston M., Jr.
4. EMPLOYER: Georgia Institute of Technology
5. EMPLOYER DESIGNATION: U
6. DEPARTURE DATE: 09/11/84
7. RETURN DATE: 09/21/84
8. NO. DAYS PERSONAL: 0
9. FISCAL YEAR: 84
10. DESTINATION(S): 11. TRIP CATEGORY:

--	__/__/__/__
--	__/__/__/__
--	__/__/__/__
--	__/__/__/__
--	__/__/__/__

12. CONFERENCE NO.: ----
13. B&R CLASSIFICATION: -- ----
14. ESTIMATED COST TO DOE: $2,663.00
15. ESTIMATED COST TO OTHERS: $---,---
16. JUSTIFICATION/PURPOSE: Attend IAEA Conference on Plasma Physics and Controlled Fusion and to present a paper

---

E-25-621

GEORGIA TECH 1885-1985

DESIGNING TOMORROW TODAY

Georgia Institute of Technology
Office of Contract Administration
Centennial Research Building
Atlanta, Georgia 30332-0420
(404) 894- 4817

TELEX: 542507 GTRC OCA ATL
FAX: (404) 894-3120

Refer to: JG/02.141.000.87.003

16 October 1986

E-25-621 / STACEY

U.S. Department of Energy
Oak Ridge Operations
P.O. Box E
Oak Ridge, TN 37831

Attention: Ms. Marlena Clark
Contracts Management Branch
Procurement and Contracts Division

Subject: Research Proposal for Continued Support of the INTOR Workshop.

Dear Ms. Clark:

Georgia Institute of Technology is pleased to submit the enclosed proposal prepared by Dr. W.M. Stacey. This proposal is for continuation of the work currently in progress on Grant Number DE-FG05-84ER52107. The following items are enclosed for your review:

1. Continuation proposal including SF424, Assurance of Compliance, and ER F 4620.1.

2. Progress Report for the period 1 January - 31 December 1986.

We appreciate the opportunity to submit this proposal. If you need any additional information, please call me at 404-894-4817 or Dr. Stacey at 404-894-3714/3758. We look forward to hearing form you soon.

Sincerely,

Jerry Goldbaugh
Contracting Officer

JG/rda

Addressee: 6 copies
Enclosures: 6 copies

cc: Al Opdenaker
Office of Fusion
Office of Energy Research
U.S. Dept. of Energy
Mail Stop G-256
Washington, D.C. 20545

DOE Grant DE-FG05-84ER52107

ACCOMPLISHMENTS DURING YEAR 1 (1/1/86 - 12/31/86)

The US work in support of Phase 1 Part 3 of the INTOR Workshop was organized and technically directed. The US work in support of the IAEA Meeting on Tokamak Concept Innovations was organized. The PI represented the US as Steering Committee member at sessions of the INTOR Workshop and as a member of the organizing committee for the above-mentioned IAEA meeting.

PROPOSED WORK DURING YEAR 2 (1/1/87 - 12/31/87)

The work proposed for the second year of the two-year grant is as follows: organization and technical direction of the US work in support of the INTOR Workshop including publishing and distribution of the US INTOR report; representing the US as Steering Committee member at sessions of the INTOR Workshop; performing certain technical analyses in support of the US INTOR work; and participating in DOE activities related to ETR, in particular ETOC.

OMB Approval No. 0348-0043

FEDERAL ASSISTANCE

1. TYPE OF SUBMISSION (Mark appropriate box)	☐ NOTICE OF INTENT (OPTIONAL) ☐ PREAPPLICATION ☐ APPLICATION	2. APPLICANT'S APPLICATION IDENTIFIER	a. NUMBER: JG/02.141.000.87.003
			b. DATE (Year month day): 86 10 16
		3. STATE APPLICATION IDENTIFIER (NOTE TO BE ASSIGNED BY STATE)	a. NUMBER
			b. DATE ASSIGNED (Year month day): 19

Leave Blank

4. LEGAL APPLICANT/RECIPIENT	
a. Applicant Name	Georgia Tech Research Corporation
b. Organization Unit	Centennial Research Building, Room 246
c. Street/P.O. Box	Georgia Institute of Technology
d. City	Atlanta — e. County: Fulton
f. State	Georgia — g. ZIP Code: 30332-0420
h. Contact Person (Name & Telephone No.)	Jerry Goldbaugh — 404/894-4817

5. EMPLOYER IDENTIFICATION NUMBER (EIN): 58-060 3146

6. PROGRAM (From CFDA): a. NUMBER: 81.049 — MULTIPLE ☐ — b. TITLE: Basic Energy Sciences

7. TITLE OF APPLICANT'S PROJECT (Use section IV of this form to provide a summary description of the project.)

Continued Support of the INTOR Workshop

8. TYPE OF APPLICANT/RECIPIENT (A–State, B–Special Purpose District, C–Administrative Agency, D–County, E–City, F–Other Specify): Enter appropriate letter: I

9. AREA OF PROJECT IMPACT (Names of cities, counties, states, etc.)

10. ESTIMATED NUMBER OF PERSONS BENEFITING

11. TYPE OF ASSISTANCE (A–Grant, B–Cooperative Agreement, C–Loan, D–Other): Enter appropriate letter: A

12. PROPOSED FUNDING

FEDERAL	$131,421 .00
APPLICANT	0 .00
STATE	0 .00
LOCAL	0 .00
OTHER	0 .00
Total	$131,421 .00

13. CONGRESSIONAL DISTRICTS OF:

a. APPLICANT	b. PROJECT
Fifth	Fifth

14. TYPE OF APPLICATION (A–New, B–Revision, C–Continuation, D–Augmentation): Enter appropriate letter: B

15. PROJECT START DATE (Year month day): 87 01 01

16. PROJECT DURATION: 12 Months

17. TYPE OF CHANGE (For 4a or 4b) (A–Increase Dollars, B–Decrease Dollars, C–Increase Duration, D–Decrease Duration, E–Cancellation, F–Other Specify): Enter appropriate letter:

18. DATE DUE TO FEDERAL AGENCY (Year month day): 19

19. FEDERAL AGENCY TO RECEIVE REQUEST	U.S. Department of Energy
20. EXISTING FEDERAL GRANT IDENTIFICATION NUMBER	DE-FG05-84ER52107
ORGANIZATIONAL UNIT (IF APPROPRIATE)	Oak Ridge Operations
b. ADMINISTRATIVE CONTACT (IF KNOWN)	Ms. Marlena Clark
ADDRESS	Contracts Management Branch, Procurement and Contracts Division, P.O. Box E, Oak Ridge, TN 37831
21. REMARKS ADDED	☐ Yes ☐ No

APPLICANT CERTIFIES THAT: To the best of my knowledge and belief, data in this preapplication/application are true and correct, the document has been duly authorized by the governing body of the applicant and the applicant will comply with the attached assurances if the assistance is approved.

a. YES. THIS NOTICE OF INTENT/PREAPPLICATION/APPLICATION WAS MADE AVAILABLE TO THE STATE EXECUTIVE ORDER 12372 PROCESS FOR REVIEW ON: DATE _____

b. NO. PROGRAM IS NOT COVERED BY E.O. 12372 ☑

OR PROGRAM HAS NOT BEEN SELECTED BY STATE FOR REVIEW ☐

TYPED NAME AND TITLE OF AUTHORIZED REPRESENTATIVE	Jerry Goldbaugh, Contracting Officer
b. SIGNATURE	
APPLICATION RECEIVED (Year month day) 19	23. FEDERAL APPLICATION IDENTIFICATION NUMBER
24. FEDERAL GRANT IDENTIFICATION	

ACTION TAKEN: AWARDED / REJECTED / RETURNED FOR AMENDMENT / RETURNED FOR E.O. 12372 SUBMISSION BY APPLICANT TO STATE / DEFERRED / WITHDRAWN

25. FUNDING

a. FEDERAL	$ .00
b. APPLICANT	.00
c. STATE	.00
d. LOCAL	.00
e. OTHER	.00
f. TOTAL	$ .00

26. ACTION DATE (Year month day): 19

27. CONTACT FOR ADDITIONAL INFORMATION (Name and telephone number)

28. STARTING DATE (Year month day): 19

29. ENDING DATE (Year month day): 19

30. REMARKS ADDED: ☐ Yes ☐ No

OMB No. 1910-0400
Expires 4/30/86

U.S. Department of Energy

Assurance of Compliance

Nondiscrimination in Federally Assisted Programs

GEORGIA TECH RESEARCH CORPORATION (Hereinafter called the "Applicant") HEREBY AGREES to omply with Title VI of the Civil Rights Act of 1964 (Pub. L. 88-352), Section 16 of the Federal Energy Administration Act f 1974 (Pub. L. 93-275), Section 401 of the Energy Reorganization Act of 1974 (Pub. L. 93-438), Title IX of the Education mendments of 1972, as amended, (Pub. L. 92-318, Pub. L. 93-568, and Pub. L. 94-482), Section 504 of the Rehabilitation ct of 1973 (Pub. L. 93-112), the Age Discrimination Act of 1975 (Pub. L. 94-135), Title VIII of the Civil Rights Act of 968 (Pub. L. 90-284), the Department of Energy Organization Act of 1977 (Pub. L. 95-91), and the Energy Conservation nd Production Act of 1976, as amended, (Pub. L. 94-385). In accordance with the above laws and regulations issued pur- uant thereto, the Applicant agrees to assure that no person in the United States shall, on the ground of race, color, national rigin, sex, age, or handicap, be excluded from participation in, be denied the benefits of, or be otherwise subjected to iscrimination under any program or activity in which the Applicant receives Federal assistance from the Department of nergy.

Applicability and Period of Obligation

the case of any service, financial aid, covered employment, equipment, property, or structure provided, leased, or im- oved with Federal assistance extended to the Applicant by the Department of Energy, this assurance obligates the Appli- nt for the period during which Federal assistance is extended. In the case of any transfer of such service, financial aid, uipment, property, or structure, this assurance obligates the transferee for the period during which Federal assistance extended. If any personal property is so provided, this assurance obligates the Applicant for the period during which it ains ownership or possession of the property. In all other cases, this assurance obligates the Applicant for the period ring which the Federal assistance is extended to the Applicant by the Department of Energy.

Employment Practices

ere a primary objective of the Federal assistance is to provide employment or where the Applicant's employment practices ect the delivery of services in programs or activities resulting from Federal assistance extended by the Department, the plicant agrees not to discriminate on the ground of race, color, national origin, sex, age, or handicap, in its employment ctices. Such employment practices may include, but are not limited to, recruitment, recruitment advertising, hiring, layoff termination, promotion, demotion, transfer, rates of pay, training and participation in upward mobility programs, or er forms of compensation and use of facilities.

Subrecipient Assurance

Applicant shall require any individual, organization, or other entity with whom it subcontracts, subgrants, or subleases the purpose of providing any service, financial aid, equipment, property, or structure to comply with laws cited above. To end, the subrecipient shall be required to sign a written assurance form, however, the obligation of both recipient and recipient to ensure compliance is not relieved by the collection or submission of written assurance forms.

Data Collection and Access to Records

Applicant agrees to compile and maintain information pertaining to programs or activities developed as a result of the licant's receipt of Federal assistance from the Department of Energy. Such information shall include, but is not limited

E-25-21

COVER SHEET

FOR TRIP REPORTS SUBMITTED TO THE

OFFICE OF ENERGY RESEARCH

Destination(s) and Dates for
Which Trip Report Being Submitted: May 10 - June 2, 1984

Name of Traveler: Weston M. Stacey, Jr.

Joint Trip Report [ ] Yes

[x] No

If so, Name of Other Traveler(s): ---

Georgia Institute of Technology
A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
SCHOOL OF NUCLEAR ENGINEERING AND HEALTH PHYSICS
ATLANTA, GEORGIA 30332
(404) 894-3720

June 8, 1984

Report on Foreign Travel — Grant No. DE-FG05-84ER-52107

Traveler - Weston M. Stacey, Jr. — Project No. E-26-601

A. Visit to Kernforschungzentrum Karlsruhe, FRG - KfK (5/15-6/84)

Persons Contacted: W. Heinz (Director) and P. Komarek, Institute for Technical Physics, G. Kessler (Director), R. Frohlich and E. Kiefhaber, Institute for Neutron Physics and Reactor Technology, H. Hinnies (Director of KfK), and H. Vetter (Director of the Fusion Program)

1. Technical discussions were held with W. Heinz (Director) and P. Komarek of the Institute for Technical Physics. The European coil for the Large Coil Project has recently been successfully tested at KfK, and is now being stored at KfK awaiting successful completion of the US coils. KfK has established extensive facilities for superconducting development and intends to use these facilities for fusion development. KfK will soon undertake the development of a set of superconducting PF coils for Tore-Supra.

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Georgia Institute of Technology
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ATLANTA, GEORGIA 30332
(404) 894-3720

June 8, 1984

Report on Foreign Travel (Continued) Grant No. DE-FG05-84ER-52107

Traveler - Weston M. Stacey, Jr. Project NO. E-26-601

B. Visit to Max-Planck Institute for Plasma Physics, Garching, FRG (5/17-18/84)

Persons Contacted: G. von Gierke, M. Kaufman, G. Grieger, R. Tosci (Director NET), F. Engelmann and H. Raeder

1. Technical discussions were held with G. von Gierke and M. Kaufman on ASDEX and ASDEX - upgrade. The current ASDEX experimental program is testing the "Troyon beta-limit". Discharges with varying values of q and with varying impurity content have been obtained. To date, it has not been possible to exceed the Troyon limit. It has also been found that with some extended discharges (several hundred milliseconds) the H-mode confinement deteriorates to a level intermediate between H- and L-mode. ASDEX-upgrade, which will become operational in 1986 - 87, will be capable of testing the poloidal diverter and pumped limiter in configurations that are similar to INTOR/NET/TFCX in geometry and power flux.

2. Technical discussions were held with G. Grieger on the W7 Stellarator program. W7-A will end nine years of operation in December, 1984, and W7-AS will begin operation in 1986. Detailed testing of W7-AS components will take place over the next year. A segment of the vacuum vessel has already been fabricated and tested successfully. The winding pack design for the W7-AS coils consists almost entirely of conductor, in contrast to the US ATF design, which features a large structural "tee".

3. Technical discussions were held with R. Tosci (Director NET), F. Engelmann and H. Raeder on NET. The NET objectives are similar to those of INTOR (ignition and long burn physics, reactor relevant technology, and engineering testing capability). The present NET concept is very similar to INTOR, except slightly smaller because of somewhat less demanding performance requirements (~3 $MW.y/m^2$ and 25% availability for NET as compared to ~7$MW.y/m^2$ and 40 - 50% availability for INTOR). The schedule of the NET project is: 1) to develop a set of reference parameters for the preliminary design by the end of 1984, 2) to initiate detailed design in 1988; and 3) to start construction in 1992. The NET team consists of 20 professionals located at Garching (to increase to 65 by 1987) plus smaller detached groups at other locations, with a total of about 40 - 50 professionals. It is felt by leaders of the NET team that US commitment to TFCX could delay the NET schedule unless the JET results are outstanding.

C. Attendance at the INTOR Workshop in Vienna - (5/21-6/1/84)

The traveler attended Session IX of the Phase 2A, Part 2 INTOR Workshop as leader of the US delegation. Other attendees are shown in the attachment. The primary business of this workshop session was to discuss the work on critical issues and data base assessment. Tentative conclusions are listed below:

1. Impurity Control

   a. Experiments confirm physics basis for high recycling (high n, low T) divertor. There is relatively good agreement between experiments and calculations.

   b. Calculations indicate that it may be possible to reduce the height of the divertor chamber from ~1m ⟶ ~.5m. This may substantially reduce the space occupied by the divertor.

   c. Low-z limiters work, but extrapolation of the pumped-limiter to INTOR conditions indicates potential problems with erosion, pumping and impurity control.

   d. It appears credible to design divertor plates (W on SS) that will have long ($\gtrsim$ 2 y) lifetimes under INTOR conditions, although tritium permeation and radiation damage are a concern. New physics calculations indicate the peak heat flux may be ~15MW/m^2.

2. RF Heating and Current Drive

   a. ICRH remains the preferred option for bulk heating to ignition, although worrisome problems (e.g. impurity production) remain.

   b. The data base for LH current ramp-up and transformer recharge is not yet adequate to justify reliance thereon to achieve basic design objectives, although provision should be made to incorporate LH and to take advantage of it in the event it works as hoped.

   c. Use of LH current initiation could reduce the required inductive voltage ~35⟶~5V, with significant engineering impact. However, the data base is not yet adequate to justify reliance thereon.

3. Transient Electromagnetics

   a. Active control coils located outside the torus shield but inside the TF coils are required to stabilize vertical displacements.

b. Passive stabilization elements close to the plasma may not be required if the normal torus structure has adequate EM properties. Toroidal continuity does not appear to be required for FWBS.

4. Maintainability

   a. There does not appear to be any significant advantage to be realized by replacing the present "personnel access" maintenance concept by a "totally remote" concept.

5. Technical Benefit

   a. Partitioning the detailed design and fabrication of high technology components among the four partners so that all can benefit technically would increase the total cost by ~ 70%, increase the required staff by ~15% and lengthen the construction and design schedule by about 2 years.

REPORT OF FOREIGN TRAVEL COST

1. Name of Traveler: Weston M. Stacey, Jr.

2. Name of Contractor: Department of Energy

3. Contract Number: Grant DE-FG05-84-52107

4. Inclusive Dates of Travel: May 10, 1984

5. Estimated Cost of Trip Shown on Form AEC-445: $3,800.00

6. Actual Cost of Trip Charged to AEC Funds: $4,106.86

ENERGY RESEARCH FOREIGN TRAVEL REQUEST COVER SHEET

1. TRIP NO.: ------
2. HQ APPROVAL REQUIRED: Y
3. TRAVELER'S NAME: Stacey, Weston M., Jr.
4. EMPLOYER: Georgia Institute of Technology
5. EMPLOYER DESIGNATION: U
6. DEPARTURE DATE: 5/10/84
7. RETURN DATE: 6/2/84
8. NO. DAYS PERSONAL: 4
9. FISCAL YEAR: 84
10. DESTINATION(S):

	11. TRIP CATEGORY:
W.GE	LR/__/__/__
AU	IC/__/__/__
--	__/__/__/__
--	__/__/__/__
--	__/__/__/__

12. CONFERENCE NO.: ----
13. B&R CLASSIFICATION: -- ----
14. ESTIMATED COST TO DOE: $ 3800
15. ESTIMATED COST TO OTHERS: $ ---
16. JUSTIFICATION/PURPOSE: Attend INTOR Workshop; visit 2 German Laboratories in route.

DOE-F-1500.3
(Previous Editions are Obsolete)

REQUEST FOR APPROVAL OF OFFICIAL FOREIGN TRAVEL

PART A—To be completed by traveler's administrative officer

Budget and Reporting Classification to be charged: _____
(See Appendix 1101, Part II)

(The category of a trip, either I, II, or III, is determined by the definitions set forth in Appendix 1501, Part VIII) TRIP CATEGORY: _____

PART B—To be completed by traveler

a. NAME OF TRAVELER Weston M. Stacey, Jr. CITIZENSHIP USA	c. DATE AND PLACE OF BIRTH July 23, 1937; Birmingham, AL USA d. PASSPORT NUMBER (if available)
a. HOME ADDRESS X-4 Royal Coach Apts. Atlanta, Ga. 30309 USA	b. BUSINESS ADDRESS Georgia Institute of Technology Atlanta, GA 30332
a. EMPLOYER Georgia Institute of Technology Atlanta, GA 30332 USA	c. CONTRACT NUMBER Grant No. DE-FG05-84-52107
ORGANIZATIONAL UNIT School of Mechanical Engineering/Nuclear Engineering & Health Physics	d. POSITION TITLE (including profession) Callaway Professor of Nuclear Engineering

PURPOSE OF TRAVEL—Include all pertinent background information leading to travel and attach copies of invitations and correspondence regarding travel to present papers, give speeches, or to attend conferences or symposia. Also identify by name and organization other DOE and contractor personnel who, to the traveler's knowledge, are going to the same destination at the same time as the traveler. In addition, specify nature and classification of information to be disclosed including titles of papers to be presented; nature of information to be obtained at each of the places to be visited and conferences to be attended and its relation to traveler's work. Travelers are responsible for obtaining clearances for papers or speeches when necessary. If more space is required, attach a separate sheet. NOTE: IF THIS INFORMATION IS CLASSIFIED BE SURE TO CLASSIFY THIS FORM APPROPRIATELY.

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The information furnished will be used by DOE to authorize travel and payment of travel expenses; by the Department of State to issue a passport; by the General Accounting Office to audit and verify the accuracy and legality of disbursements.

PART E—To be completed at DOE Field Organization and/or Laboratory

11. APPROVAL AND/OR REMARKS BY HEAD OF DOE FIELD ORGANIZATION OR DIRECTOR OF LABORATORY:

(Signature) *(Date)*

PART F—To be completed at Headquarters

12. CONCURRENCE AND/OR REMARKS OFFICE OF SAFEGUARDS AND SECURITY
(Note compliance with Chapter 2502)

(Signature and Title) *(Date)*

13. CONCURRENCE AND/OR REMARKS OFFICE OF INTERNATIONAL SECURITY AFFAIRS

(Signature and Title) *(Date)*

14. APPROVAL OR CONCURRENCE/REMARKS BY ASSISTANT SECRETARY FOR INTERNATIONAL AFFAIRS

(Signature and Title) *(Date)*

15. CONCURRENCE AND/OR REMARKS—OTHERS

(Signature and Title) *(Date)*

16. CONCURRENCE/REMARKS OF DIRECTOR OF DIVISION OR OFFICE

(Signature) *(Date)*

17. APPROVAL/REMARKS BY ASSISTANT SECRETARY/DIRECTOR

(Signature) *(Date)*

COVER SHEET

FOR TRIP REPORTS SUBMITTED TO THE

OFFICE OF ENERGY RESEARCH

Destination(s) and Dates for
Which Trip Report Being Submitted: May 10 - June 2, 1984

Name of Traveler: Weston M. Stacey, Jr.

Joint Trip Report [ ] Yes

[x] No

If so, Name of Other Traveler(s): ---

Georgia Institute of Technology
A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
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ATLANTA, GEORGIA 30332
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June 8, 1984

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(404) 894-3720

June 8, 1984

Report on Foreign Travel (Continued) Grant No. DE-FG05-84ER-52107

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REPORT OF FOREIGN TRAVEL COST

1. Name of Traveler: Weston M. Stacey, Jr.

2. Name of Contractor: Department of Energy

3. Contract Number: Grant DE-FG05-84-52107

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5. Estimated Cost of Trip Shown on Form AEC-445: $3,800.00

6. Actual Cost of Trip Charged to AEC Funds: $4,106.86

ENERGY RESEARCH FOREIGN TRAVEL REQUEST COVER SHEET

1. TRIP NO.: ------
2. HQ APPROVAL REQUIRED: Y
3. TRAVELER'S NAME: Stacey, Weston M., Jr.
4. EMPLOYER: Georgia Institute of Technology
5. EMPLOYER DESIGNATION: U
6. DEPARTURE DATE: 5/10/84
7. RETURN DATE: 6/2/84
8. NO. DAYS PERSONAL: 4
9. FISCAL YEAR: 84
10. DESTINATION(S):

W.GE	
AU	
--	
--	
--	

11. TRIP CATEGORY:

LR/__/__/__	
IC/__/__/__	
__/__/__/__	
__/__/__/__	
__/__/__/__	

12. CONFERENCE NO.: ----
13. B&R CLASSIFICATION: -- ----
14. ESTIMATED COST TO DOE: $ 3800
15. ESTIMATED COST TO OTHERS: $ ---
16. JUSTIFICATION/PURPOSE: Attend INTOR Workshop; visit 2 German Laboratories in route.

DOE-M-1500.3
(Previous Editions are Obsolete)

REQUEST FOR APPROVAL OF OFFICIAL FOREIGN TRAVEL

PART A—To be completed by traveler's administrative officer

Budget and Reporting Classification to be charged: _____
(See Appendix 1101, Part II)

(The category of a trip, either I, II, or III, is determined by the definitions set forth in Appendix 1501, Part VIII) TRIP CATEGORY: _____

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ORGANIZATIONAL UNIT School of Mechanical Engineering/Nuclear Engineering & Health Physics	d. POSITION TITLE (including profession) Callaway Professor of Nuclear Engineering

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The information furnished will be used by DOE to authorize travel and payment of travel expenses; by the Department of State to issue a passport; by the General Accounting Office to audit and verify the accuracy and legality of disbursements.

PART E—To be completed at DOE Field Organization and/or Laboratory

11. APPROVAL AND/OR REMARKS BY HEAD OF DOE FIELD ORGANIZATION OR DIRECTOR OF LABORATORY:

(Signature) *(Date)*

PART F—To be completed at Headquarters

12. CONCURRENCE AND/OR REMARKS OFFICE OF SAFEGUARDS AND SECURITY
(Note compliance with Chapter 2502)

(Signature and Title) *(Date)*

13. CONCURRENCE AND/OR REMARKS OFFICE OF INTERNATIONAL SECURITY AFFAIRS

(Signature and Title) *(Date)*

14. APPROVAL OR CONCURRENCE/REMARKS BY ASSISTANT SECRETARY FOR INTERNATIONAL AFFAIRS

(Signature and Title) *(Date)*

15. CONCURRENCE AND/OR REMARKS—OTHERS

(Signature and Title) *(Date)*

16. CONCURRENCE/REMARKS OF DIRECTOR OF DIVISION OR OFFICE

(Signature) *(Date)*

17. APPROVAL/REMARKS BY ASSISTANT SECRETARY/DIRECTOR

(Signature) *(Date)*

Project No. E-26-601
NE&HP
Lab/School

GEORGIA INSTITUTE OF TECHNOLOGY

RESEARCH REPORT APPROVAL SHEET

(Please see Instructions on back)

I. GENERAL

A. Report Title INTOR/Foreign Travel Report

B. Author(s) and/or Proj. Dir. Weston M. Stacey, Jr.

C. Period Covered May 10 - June 2, 1984

II. REPORT IDENTIFICATION

A. Deliverable Line No. _____ (from Deliverables Schedule)

B. Type Report:

- ☐ Monthly ☐ Quarterly ☐ Semi-Annual
- ☐ Annual ☐ Progress ☐ Cost & Performance
- ☐ Contract Mgt. ☐ Final ☒ Other Foreign (Specify)

C. Draft Copy for Sponsor Approval ☐

D. Approved Copy for Distribution:

1. ☐ Send Reproducible Copy to Sponsor
2. ☐ Return Reproducible Master to Project Director

III. SECURITY CLASSIFICATION

- ☒ Unclassified ☐ Secret ☐ In Confidence (Specified Industrial Contracts Only)
- ☐ Confidential ☐ Top Secret ☐ Other _____ (Specify)

IV. DATA RESTRICTIONS ☐ Yes ☐ No

Export License; Trade Secret; Copyright or Publishing Restrictions; Patentable Technology; Proprietary Data (Confidential Disclosure).

NOTE: If "Yes", contact OCA LEGAL SERVICES, Ext. 4812.

V. PRINTING/DISTRIBUTION INSTRUCTIONS

A. No. Copies to be Printed for:

Sponsor _____; OCA 2; School/Lab _____; Other _____; Total _____

B. Printing Desired: ☐ One-side only ☐ Two sides

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A. Proj. Director _____ _____

B. Div. Chief/Branch Head _____ _____

C. Lab/School Director _____ _____
(Required on Finals)

D. Coordinator Log Entry: _____ _____

Report Distributed by: _____ Date: June 8, 1984

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This form supercedes Forms RA64(5-70) & EES Form 722(R270)

OCA Form 1-481

ATTACHMENT

Department of Research and Isotopes
Division of Research and Laboratories

Issue No.2
22 May 1984

NOTIFICATION OF AN AGENCY-SPONSORED MEETING

Title of Meeting: INTOR Workshop Session IX

Opening Meeting: 9:30 a.m.

Dates Inclusive: 21 May-1 June 1984

Responsible Officers: M. Leiser
A-2375

Place: B07

A. Shurygin
A-2305

PARTICIPANTS AND DESIGNATING MEMBER STATES AND ORGANIZATIONS	ADDRESSES ABROAD	IN VIENNA	FOR THE PERIOD
EURATOM			
G. Grieger	Max-Planck-Institute for Plasma Physics D-8046 Garching FRG		21 May-1 June 1984
M. Chazalon	Max-Planck-Institute for Plasma Phyics D-8046 Garching FRG		21 May-1 June 1984
F. Engelmann	FOM-Institute for Plasma Physics Postbus 7 Jutphaas, Nieuwegein NETHERLANDS		21 May-1 June 1984
F. Farfaletti-Casali	Joint Research Centre Ispra Establishment 1-21010 ISPRA (VARESE)		21 May-1 June 1984

M. Harrison	Culham Laboratory Abingdon, Oxfordshire OX14 3DB UK		21 May-1 June 1984
A. Knobloch	Max-Planck-Institute for Plasma Physics D-8046 Garching FRG		21 May-1 June 1984
D. Leger	CEN-Saclay, DESICP 91191 Gif-sur-Yvette Cedex FRANCE		21 May-1 June 1984
E. Salpietro	Max-Planck-Institute for Plasma Physics D-8046 Garching FRG		21 May-1 June 1984
P. Schiller	Joint Research Centre Ispra Establishment 1-21010 ISPRA (VARESE) ITALY		21 May-1 June 1984
R. Verbeek	CEC, SDMI/126 200, rue de la Loi B-1049 Brussels BELGIUM		21 May-1 June 1984
JAPAN			
S. Mori	JAERI Fukoku- Seimei Bldg 2-2-2, Uchisaiwai-cho Chiyoda-ku, Tokyo 100	Hotel Europa 1010, Neuer Markt 3 52 15 94	21 May-1 June 1984
N. Fujisawa	Fusion Research Center JAERI Naka-machi, Naka-gun	Hotel Royal 1010, Singerstr. 3 52 46 31	21 May-1 June 1984

T. Honda	Toshiba Corporation 1-1-6, Uchisaiwai-cho,Chiyoda-ku Tokyo, 100	Hotel Royal 1010. Singerstr. 3 52 46 31	21 May-1 June 1984
S. Itoh	Hitachi Ltd. Saiwai-cho, Hitachi-shi Ibaraki-ken	- " -	21 May-1 June 1984
H. Kimura	Fusion Research Center JAERI Naka-machi, Naka-gun Ibaraki-ken 319-11	- " -	21 May-1 June 1984
T. Tone	Fusion Research Center JAERI Naka-machi, Naka-gun Ibaraki-ken	- " -	21 May-1 June 1984
K. Tomabechi	Fusion Research Center JAERI Naka-machi, Naka-gun Ibaraki-ken 319-11	- " -	21 May-1 June 1984
K. Ueda	Mitsubishi Fusion Center 2-3, Marunouchi, 2 -Chome Chiyoda-ku, Tokyo		21 May-1 June 1984
USSR			
B. Kadomtsev	I.V. Kurchatov Institute of Atomic Energy 46 Ulitsa Kurchatova D-182 Moscow		21 May-1 June 1984
B. Kolbasov	I.V. Kurchatov Institute of Atomic Energy 46 Ulitsa Kurchatova D-182 Moscow		21 May-1 June 1984

A. Kostenko	D.V. Efremov Scientific Institute P.O. Box 42 Leningrad		21 May-1 June 1984
V. Pistunovich	I.V. Kurchatov Institute of Atomic Energy 46 Ulitsa Kurchatova D-182 Moscow		21 May-1 June 1984
V. Kukushkin	I.V. Kurchatov Institute of Atomic Energy 46 Ulitsa Kurchatova D-182 Moscow		21 May-1 June 1984
D. Serebrennikov	D.V. Efremov Scientific Institute P.O. Box 42 Leningrad		21 May-1 June 1984
G. Shatalov	I.V. Kurchatov Institute of Atomic Energy 46 Ulitsa Kurchatova D-182 Moscow		21 May-1 June 1984
USA			
W.M. Stacey, Jr.	School of Nuclear Engineering Georgia Institute of Technology Atlanta, Georgia		21 May-1 June 1984
C. C. Baker	Argonne National Laboratory 9700 South Cass Avenue Argonne, Illinois	Pension Wiener 1010, Seilerg.16 52 48 16	21 May-1 June 1984

P.L. Colestock	Princeton Plasma Physics Laboratory P.O. Box 451 Princeton, New Jersey	Pension Wiener 1010, Seilerg.16 52 48 16	21 May-1 June 1984
C.A. Flanagan	Oak Ridge National Laboratory P.O. Box Y Oak Ridge, Tennessee 37830	- " -	21 May-1 June 1984
F. Mattas	Argonne National Laboratory 9700 South cass Avenue Argonne, Illinois	- " -	21 May-1 June 1984
M.K. Peng	Oak Ridge National Laboratory Oak Ridge, Tennessee 37830	- " -	21 May-1 June 1984
D. Post	Princeton Plasma Physics Laboratory P.O. Box 451 Princeton, New Jersey	- " -	21 May-1 June 1984
P. Spampinato	Oak Ridge National Laboratory Oak Ridge, Tennessee 37830	- " -	21 May-1 June 1984
R.J. Thome	Massachusetts Institute of Technology Cambridge, Massachusetts	- " -	21 May-1 June 1984

E-25-62

COVER SHEET

FOR TRIP REPORTS SUBMITTED TO THE

OFFICE OF ENERGY RESEARCH

Destination(s) and Dates for
Which Trip Report Being Submitted: Vienna, Austria October 12-28, 1984

Name of Traveler: Weston M. Stacey, Jr.

Joint Trip Report [ ] Yes

[X] No

If so, Name of Other Traveler(s): ---

ENERGY RESEARCH FOREIGN TRAVEL REQUEST COVER SHEET

1. TRIP NO.: ------
2. HQ APPROVAL REQUIRED: Y
3. TRAVELER'S NAME: Stacey, Weston M., Jr
4. EMPLOYER: Georgia Institute of Technology
5. EMPLOYER DESIGNATION: U
6. DEPARTURE DATE: 10/15/84
7. RETURN DATE: 10/26/84
8. NO. DAYS PERSONAL: --
9. FISCAL YEAR: 84
10. DESTINATION(S): ALL
11. TRIP CATEGORY: WS/__/__/__
12. CONFERENCE NO.: ----
13. B&R CLASSIFICATION: -- ----
14. ESTIMATED COST TO DOE: $3,522.00
15. ESTIMATED COST TO OTHERS: $---,---
16. JUSTIFICATION/PURPOSE: attended Session X of INTOR Workshop at IAEA headquarters in Vienna, Austria

Georgia Institute of Technology
A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
SCHOOL OF MECHANICAL ENGINEERING

October 31, 1984

Please reply to:
NUCLEAR ENGINEERING AND
HEALTH PHYSICS PROGRAM
CHERRY EMERSON BUILDING
GEORGIA INST. OF TECH.
ATLANTA, GEORGIA 30332 U.S.A.

Report on Foreign Travel

Grant No. DE-FG05-84ER-52107

Traveler - Weston M. Stacey, Jr.

Project No. E-26-621

The traveler attended session X of the INTOR Workshop at IAEA headquarters in Vienna, Austria over the period October 15-26, 1984. An agenda and a combined attendee list/workshop organization are attached (attachments 1 and 2).

The principal objectives of this session were:

a. to draw conclusions and prepare a detailed documentation outline for the critical issue studies and data base assessment studies which have been going on for somewhat more than a year; and

b. to review the results of studies of possible modifications of the INTOR design concept and to tentatively decide on changes in the design concept, to be checked for consistency before the next session.

These objectives were accomplished.

The preliminary conclusions of the critical issues studies and data base assessment, which are summarized in attachment 3, were by and large confirmed and amplified. Detailed outlines were prepared and writing assignments were made.

A set of recommended design modifications, which would improve the INTOR design concept without sacrificing capability, were identified (attachment 4). Homework tasks were developed to check the consistency of these modifications to a first level of detail. The intent is to reach a final consensus of the modified INTOR design concept at the next Workshop session. This modified INTOR concept will represent an updated international consensus of the appropriate next-step tokamak experiment to be undertaken on an international basis, and as such will provide a useful reference in ongoing negotiations on international collaboration in fusion.

A revised proposal (attachment 5) from the INTOR Steering Committee to the International Fusion Research Council (IFRC) of the IAEA was developed for the next phase of the INTOR Workshop. The principal objectives of this next phase, as proposed, would be to:

a. stimulate and evaluate innovations that could lead to an improved tokamak reactor concept; and

b. focus international effort on critical technical issues for next-step tokamak experiments.

Attachment 1

1 June 1984

Revised 10/19/84

TENTATIVE AGENDA

SESSION X

INTOR WORKSHOP PHASE IIA, PART 2

OCTOBER 15 - 26, 1984

Monday, October 15		
A.M.	Plenary Session	9:30 - 12:30
P.M.	Exchange Reports	13:30 - 14:30
	Read Reports	
	Coordinating Committee	17:00
Tuesday, October 16		
A.M.	Group Discussions (A-E)	
P.M.	Group Discussions (A-E)	
Wednesday, October 17		
A.M.	Group Discussions (F-H)	
P.M.	Group discussions (A-E)	
	Coordinating Committee	17:00
Thursday, October 18		
A.M.	Group Discussions (F-H)	
P.M.	Group Discussions (A-E)	
Friday, October 19		
A.M.	Group Discussion (F-H)	
P.M.	Plenary Session	14:00 - 17:00
	Coordinating Committee	17:00
Monday, October 22		
A.M. and PM	Working Time (Drafting of critical issues and data base detailed outlines, report content, etc.)	
	Critical Issue and Data Base final reports due to typists.	

Tuesday, October 23

A.M.	Group Discussions (F-H)	8:30 - 10.30
	Group Discussions (A-E)	11:00 - 12:30
P.M.	Intergroup Discussion (led by G)	
	Coordinating Committee	17:00

Wednesday, October 24

A.M.	Intergroup Discussion (led by G)	
P.M.	Group Discussions (A-E)	14:00 - 16:00
	Group Discussions (F-G)	16:30 - 18:00

Thursday, October 25

A.M.	Plenary Session	9:00 - 10:30
	Group Discussions (F-H)	11:00 - 12:00
P.M.	Group Discussions (A-E)	14:00 - 15:30
	Group Discussions (F-H)	15:45 - 17:00
	Coordinating Committee	17:00
	Design modification final reports due to typists.	

Friday, October 26

A.M.	Working Time	
P.M.	Plenary Session	14:00 - 16:00

1. Discussions in the groups in the first week should put emphasis on critical issues and data base assessment, so as to reach agreement at the end of the week on content and conclusion of the report and to develop a detailed outline and writing assignments for the critical issues and data base chapters.

2. The agreement is then drafted and due to typists by Monday, October 22.

3. The second week is devoted to task of design modification.

4. Tentative agreement on design modifications should be reached by Wednesday, October 24, and reported at Plenary Session in p.m. of Wednesday. After comments from Plenary Session and Coordinating Committee, final agreement on design modifi...

PHASE 2A, PART 2, SESSION X - INTOR WORKSHOP ORGANIZATION (15th October 1984)

	Chairman	EC	Japan	USA	USSR
A. Impurity Control and First Wall	Post	Harrison Schiller - VC Leger	Fujisawa Seki	Post Mattas	Pistunovich Serebrennikov
B. RF Heating and Current Drive	Engelmann	Engelmann	Sugihara (Engr.: Iida)	Colestock	Kukushkin
C. Transient Electromagnetics	Kostenko	Salpietro	Kasai	Peng Thome	Kostenko Litunovskij
D. Maintainability	Tomabechi	Farfaletti-Casali Vieider	Tomabechi Honda	Spampinato Flanagan	Kolbasov Shatalov
E. Technical Benefit	Knobloch	Knobloch	Iida (Tomabechi)	Flanagan	Kolbasov Litunovskij
F. Physics	Pistunovich	Engelmann Harrison	Fujisawa Sugihara	Post Peng Colestock	Pistunovich Kukushkin
G. Engineering	Flanagan	Farfaletti-Casali Salpietro Knobloch	Iida Seki Kasai	Flanagan Spampinato Thome	Kostenko Serebrennikov Litunovskij
H. Nuclear	Tomabechi	Vieider Schiller Leger	Tomabechi Honda (Iida)	Mattas	Shatalov Kolbasov
eering Committee	Mori	Grieger	Mori	Stacey	Kolbasov

INTERNATIONAL ATOMIC ENERGY AGENCY

TENTH INTERNATIONAL CONFERENCE ON PLASMA PHYSICS AND CONTROLLED NUCLEAR FUSION RESEARCH

London, UK, 12–19 September 1984

IAEA-CN-44/G-I-2

OVERVIEW OF THE INTOR WORKSHOP

INTOR GROUP*
Prepared by W. M. Stacey, Jr.**

IAEA
Vienna, Austria

September, 1984

* INTOR Group is identified in paper G-I-1.

** Georgia Institute of Technology
Atlanta, Georgia USA

IAEA-CN-44/G-I-2

OVERVIEW OF THE INTOR WORKSHOP

ABSTRACT

This paper summarizes the present status of activities being carried out in the INTOR Workshop. The present INTOR design concept is described. Preliminary results from the ongoing critical issues studies and data base assessment are discussed.

1. INTRODUCTION

During the present Phase 2A, Part 2 the INTOR Workshop is examining several critical technical issues which affect the feasibility or cost of a next generation tokamak experiment, is reassessing the physics and technology data bases which support such an experiment, is evaluating the technical and financial benefit that would result from an international project, and is evolving the INTOR design concept on the basis of the results of the critical issues studies and the data base assessment. The present phase of the Workshop began in mid-1983 and runs through mid-1985; so this paper represents an interim report on the status of work in progress.

2. INTOR DESIGN CONCEPT

INTOR has the objectives of being the maximum reasonable step beyond the present generation of large tokamaks, of operating in a reactor relevant mode, of incorporating reactor-relevant technologies, and of providing a facility for engineering testing. The reference INTOR design concept was developed during Phase One [1] and modified during Phase Two A, Part 1 [2]. This concept is characterized by the parameters given in Table 1 and by the cross section view given in Fig. 1. The concept will be further evolved during the present phase of the INTOR Workshop.

3. IMPURITY CONTROL

The INTOR impurity and particle control system must be able to exhaust both the alpha particle heating power (~120MW) and the alpha particles themselves (2×10^{20} He atoms/sec). This must be done in such a way as to keep the plasma reasonably free from impurities and provide a reasonable component lifetime. The helium "ash" must be removed with reasonably sized pumping systems. The major candidate systems which have been studied are

poloidal divertors and pumped limiters. These studies have consisted of an assessment of the experimental and engineering data base for impurity control systems based on current experiments, the use of sophisticated computational models to extrapolate to the operating parameters and performance of INTOR, and the development of an engineering concept for the design of such a system for INTOR.

The major impurity control problem is likely to be sputtering of the collector plate materials by energetic plasma ions and charge exchange neutrals. This problem can be reduced or even eliminated if the energy of the ions and neutrals can be reduced to a value below the sputtering threshold of potential collector plate materials. Both modelling calculations and experiments on ASDEX, D III and PDX indicate that this can be accomplished by the use of a suitably designed poloidal divertor. A cool, dense plasma ($n_e \gtrsim 10^{14}$ m^{-3}, $T_e \lesssim 30$eV) can be produced near the collector plate by intense, localized recycling of the plasma and neutrals. The experiments and models also indicate that this "high-recycling" type of divertor can be produced in an open geometry ("expanded boundary") compatible with the placement of the poloidal field coils outside the toroidal field coils. The low temperature of the diverted plasma minimizes the erosion, and the high density of the plasma provides a high neutral density which eases the helium pumping speed requirements.

An impurity control system based on a pumped limiter would be cheaper and simpler than a divertor in that it would require less space inside the vacuum vessel and a smaller and lower current poloidal field system. However, it was concluded, based on current experiments and modelling, that the production of a cool plasma edge with a limiter was not the most likely operating regime for INTOR. The most probable edge temperature with a limiter is expected to be 100-200 eV, and the net sputtering rates, even taking into account redeposition of the sputtered material back onto the limiter, would be too large to be acceptable from either the standpoint of erosion lifetime or of impurity contamination of the main plasma. It is possible that the edge temperature could be reduced, either by intense recycling or by some flow reversal mechanism which caused the impurities to accumulate in the plasma edge region and radiatively cool it, but the available evidence does not yet support such a possibility.

The engineering studies have primarily concentrated on the design of limiter and divertor collector plates. The materials for these plates must withstand high particle and heat fluxes (10^{20} particles/per cm^2 and 3-5 MW/m^2) as well as a large flux of 14 MeV neutrons. They must be resistant to erosion losses and radiation damage, and not be a source of contamination to the main plasma. No one material satisfies both the surface sputtering and plasma contamination requirements, and the resistance to thermal stresses, etc., necessary for high rates of heat removal.

Thus, the designs have incorporated a plasma side material with good sputtering and impurity contamination properties attached to a structural material selected to meet the strength and radiation damage requirements. For the limiter design, Be is the preferred plasma side material due to its low Z, low self-sputtering rates and high thermal conductivity. The preferred plasma side material for the divertor is W, since the sputtering rate for a 20-30eV plasma would be negligible. The preferred structural material is a copper alloy. These alloys are readily available, easily fabricated, and compatible with water cooling. As long as the peak heat flux is on the order of 4 MW/m^2 or less, both the limiter and divertor systems can have adequate heat removal. The major remaining materials question is the long term degradation of the material properties of the system components due to radiation damage.

Based on these considerations, the INTOR reference impurity control system is a poloidal divertor. The poloidal divertor is preferred because of the low sputtering rate of the collector plate produced by the cool, dense diverted plasma compared to the higher sputtering rates expected for the higher temperature plasma incident on the limiter. The experiments on D-III, ASDEX, and PDX confirm the existence of the cool diverted plasma, and the models used for INTOR show good agreement when used to analyze the experiments. Thus the poloidal divertor provides a highly credible impurity control system for INTOR. While the pumped limiter is less promising, it is retained as the back-up option, and experience on JET, TFTR, and other machines will contribute to the ultimate choice between the divertor and limiter. With the short compact design of the divertor, it takes only ~10% more space than a limiter.

4. RF HEATING AND CURRENT DRIVE

For INTOR, RF waves are considered for bulk heating to ignition, non-inductive current drive and start-up assist (plasma formation and preheating, current initiation, profile control).

4.1 Bulk Heating

For bulk heating to ignition, fast magnetosonic wave heating in the ion cyclotron range of frequencies was chosen as the reference method. This choice is based on potential technical and economic advantages of an RF system with respect to a neutral beam system. Moreover, among the various possible RF heating methods (Alfven wave, fast magnetosonic wave, ion Bernstein wave, lower hybrid wave and electron cyclotron wave) the fast wave heating approach was chosen since it is at present the most developed method. The most attractive heating mode is heating of deuterium at the second harmonic frequency (85 MHz)

which permits single pass absorption in the parameter range encountered during the heating phase except during the cool (a few keV) and rarified around (5×10^{19} m^{-3}) plasma close to the ohmic regime; in this case, single pass absorption can still occur, if important for efficient heating, by using a small concentration of protons in the deuterium plasma in a mixed proton minority/second harmonic deuterium heating scheme. Up to 3.2 MW of second harmonic heating power has been applied in PLT, resulting in heating efficiencies of 3eV/kW/10^{19} m^{-3} and achieving central ion temperatures of 3.2 keV.

In principle, overall power requirements could be further reduced by the production of sufficient thermonuclear power from second-harmonic-produced deuterium tails during the heating phase; however, the competition between proton minority and second harmonic deuterium heating will determine the effectiveness of the deuterium tail generation. The heating power required (in the absence of D^+ tail formation) is estimated to be 60 MW, including 10 MW for redundancy.

Plug-in launcher concepts were developed, both for antenna array and for waveguide systems. The critical issues of the scheme are, from a physics point of view, the impurity production during heating encountered in present-day tokamaks as well as the impact of heating on confinement and, as far as launcher technology is concerned, the design and lifetime of the Faraday shield and the ceramic vacuum barriers.

4.2 Current Drive

Non-inductive current drive would allow reduction of fatigue problems by going to longer pulse length, but is anticipated to require large power (> 100 MW) if applied at reactor-grade plasma density. Therefore, rather current ramp-up and recharging the transformer while the plasma current is sustained non-inductively, at low plasma density (~10^{19} m^{-3}) and temperature (~1keV), is under consideration for INTOR. Among all possible candidates for non-inductive current drive, lower hybrid current drive has by now the most solid experimental data base and thus is the preferred scheme. For current ramp-up and transformer recharging, a power of 10 MW to be applied for about 10-100 s is anticipated to be needed; the required wave frequency is around 2 GHz. For launching lower hybrid waves, waveguide grill arrays integrated into a plug-in module were conceived. Avoiding break-down in the wave guides by applying an appropriate surface treatment is an important issue here.

The present position with respect to using non-inductive current drive on INTOR is that INTOR should be prepared to operate with lower hybrid current ramp-up and transformer recharging with a burn pulse length of about 1000 s, but that it

should not rely on it, because of the considerably increased physics risks: 1) a burn pulse length of $\gtrsim$ 1000 s corresponds to times equal to or larger than the global classical skin time; therefore current profile control, presently not available, might be indispensable; and 2) the experimental and theoretical data base for the scheme is still insufficient for a quantification of all operational constraints.

4.3 Start-up Assist

For start-up assist, the reference choice for INTOR has been the use of electron cyclotron waves, because of their proven effectiveness for plasma formation and pre-heating as well as profile control during the current rise phase. A frequency of 140 GHz and a power of 10 MW is required. Concepts for plug-in launchers for electron cyclotron waves using wave guide or quasi-optical systems for power transmission were developed. The most critical technical problem is developing a reliable ceramic window for the transmission line, apart from the necessity to provide power sources of reasonable efficiency for the frequency range in question. A weak point of this solution for start-up assist is that there is presently no sufficient data base for non-inductive current initiation by electron cyclotron waves. The suitability of lower hybrid waves for this function (as well as for plasma formation and pre-heating) has recently been demonstrated and could prove feasible for INTOR.

As the data base develops, it will be desirable to reduce the number of auxiliary devices on INTOR that are applied to fulfill the different functions to a minimum, possibly one.

5. TRANSIENT ELECTROMAGNETICS

Transient electromagnetic effects impact tokamak design requirements for start-up, plasma stabilization, and the ability of the torus components to withstand the forces and voltages induced during plasma disruption. The most favorable design characteristics in each of these areas is often in conflict with desirable features in the other areas or with constraints imposed by other subsystems.

5.1 Start-up

Penetration delay times have been shown to differ by as much as an order of magnitude for the equilibrium field and electric field, depending on geometry, rates of change of fields and distribution of resistance in the torus.

The resistance of sector walls around blanket/shield sections relative to the resistance between sectors (e.g.-bellows) has a strong influence on the penetration of the equilibrium field because the EF is inductively well-coupled to both toroidally continuous eddy current paths and to saddle type induced current

paths which do not require toroidal continuity. On the other hand, the OH flux, which is the source of the electric field required for the plasma, is well coupled to toroidally continuous paths, but not to the saddle paths, so it is not strongly influenced by the resistance distribution azimuthally but only by the "average" level of the resistance. As a result, the study of start-up and radial position control field requirements remains an area of intense interest and requires better definition of the specifications to be imposed on the torus configuration, PF coil and plasma current scenario.

5.2 Vertical Stabilization

Recent operating scenarios for INTOR have considered the use of separate control coils to provide active vertical stabilization of the plasma. A rapid vertical plasma displacement would be initially restrained by fields due to the eddy currents induced in passive elements, then the set of active coils would be excited to provide the required field. This would limit the response time needed for the power supply and effect the power required for the control coils.

Codes are being used which allow modeling of complex passive elements and selected active coil locations with specified feedback control laws and "rigid" plasma models which can move vertically. Effort has also been started using deformable plasma models with distributed currents. In general, it has been shown that: 1) location of the active control coils outside the bulk shield results in an acceptable power requirement for the fast feedback circuit, provided that; 2) a passive element is located in close proximity to the plasma; 3) the torus structure may be capable of providing the necessary passive stabilization without the requirement for additional conducting elements; and 4) toroidal continuity of the active or passive components is not a necessary requirement, so that sectored or saddle-shaped geometries may be used effectively.

Analyses have also been carried out based on a single general stabilization circuit with gain, leading and lagging control time constants and a rigid filamentary plasma. Although oversimplified, the model leads to a concise statement of stability requirements in terms of dimensionless parameters which show: 1) active feedback is required and must exceed a threshold level; 2) the passive characteristics must exceed a threshold level which becomes more stringent as lagging control time constants increase.

For the special case where the generalized single circuit consists of two series opposing circular loops symmetrically located above and below the $z = 0$ plane, the stabilization criteria can be reduced to determine locations for which loops are effective.

Future effort will involve improved modeling of power supply characteristics, integration of the active coils into the design, and further code development to treat the combined radial and vertical control problem with a distributed current, deformable plasma.

5.3 Disruption

The rapid decay of magnetic flux associated with a plasma disruption induces currents in, and voltages between, conducting bodies which are nearby. Recent design concepts utilize toroidal shells, shell segments or limiters near the plasma which are divided into sectors for assembly and maintenance purposes or to reduce eddy current em loads. The sectoring, however, results in gaps which must withstand the voltages during disruption without arcing.

The currents, loads and voltages associated with disruption have been estimated with models using a stationary, distributed current plasma as a current source. Toroidal and poloidal induced currents interact with the applied fields and their self fields to produce overturning moments and components of force in various directions. Results indicate that the distribution of these loads is strongly dependent on toroidal continuity and sector geometry and that the loads are non-trival, but manageable.

The minimum voltages across gaps to initiate discharge (25-50 V) and to sustain arcing (10-20 V) have been estimated on the basis of a preliminary survey of experimental data. Estimates for the voltage generated across sector gaps during disruption can also be in this range, depending on sector geometry, resistance distribution, and plasma current decay model. The development of a specification for the voltage across gaps is necessary and will require model validation and experiments with realistic plasma conditions near a gap.

6. PHYSICS DATA BASE

Since the studies of INTOR Phase One, a large amount of experimental and theoretical work has contributed to clarify the main plasma physics issues. Nevertheless, additional efforts are needed to complete the INTOR data base.

6.1 Stability

Several tokamak experiments have shown that the beta limit is soft, with saturation of beta when the heating power increases. The highest beta value (4.6%) was attained in Doublet-III. The highest poloidal beta, achieved in ASDEX, is $\beta_p = 0.65$ A. These experimental values of beta do not exceed the theoretical ideal $n = \infty$ ballooning stability limit, but possibly are slightly above the low-n kink instability limit. The reference value of beta

for INTOR (5.6%) thus is not yet supported by existing data. Low q_o operation, as achieved in some small size tokamaks, increasing the plasma current or optimizing the geometrical configuration may be necessary to achieve the INTOR reference beta.

The density limit in tokamaks is frequently expressed in terms of the Murakami scaling $n_C = CB_T/R$. C values of 10^{20} $m^{-2}T^{-1}$ were reported for PDX, ISX-B, D-III at low q. This is close to what is needed for INTOR.

If the plasma operates outside the stability region, disruptions will occur. Major plasma disruptions are described theoretically as a nonlinear mixture of helical modes which build up and cool the plasma because of enhanced thermal conductivity. Accurate control of the plasma current and density profiles can help to diminish the disruption probability.

6.2 Confinement

The scaling of the energy confinement in ohmically heated plasmas is rather clear now. The ion heat transport is very close to neoclassical. However, the more important electron heat transport is much higher than neoclassical and empirically is described fairly well by neo-Alcator and by T-11 scalings. These scalings predict 5 sec for the energy confinement time of an ohmically heated INTOR discharge.

As for auxiliary heated plasmas, the experimental data are not converging so well. As compared with ohmic heating, a deterioration of confinement in the so called L-regime is evident; the confinement time decreases with the input power, increases with the current and is weakly dependent on the density. The H-regime has a factor of two better confinement, and in some cases the favorable density dependence of ohmically heated plasmas is restored. The energy confinement time is beta-dependent near the beta limit. Assuming a temperature dependence $\tau_E \sim T^{-1/2}$ as seen in the T-10 and T-11 experiments and starting from the T-11 scaling, the confinement time in INTOR can reach the reference value of 1.4 sec. Even assuming the rather unfavorable empirical correlation of Goldston and Kaye, with a multiplicative factor of two to account for H-mode, the confinement time in INTOR can reach the reference value.

The particle bulk confinement time is usually 3 to 5 times larger than the energy confinement time; the experimental values of the momentum confinement time are of the order of the energy confinement time.

6.3 Neutral Beam Heating and Current Drive

Neutral beam injection continues to be a promising back-up option for plasma heating in future large tokamaks. For current drive by neutral beam in INTOR-like devices, about 100 MW at 1 MeV would be required during the burn phase.

6.4 Equilibrium Control

Plasma position and shape control does not seem to be subject to major plasma physics uncertainties. To diminish the vertical and horizontal displacements, fast active control coils seem to be needed.

6.5 Burn Control

There are several mechanisms for controlling the plasma burn temperature, namely the toroidal field ripple, beta limits, compression-decompression, high Q-operation, and fueling. However, the optimal way of burn control is not clear yet.

7. ENGINEERING AND NUCLEAR

7.1 Maintenance Philosophy

Maintenance considerations for INTOR were established at the outset of the INTOR design studies as a fundamental consideration in the development of the design concept. A philosophy was adopted based on remote maintenance when the torus must be opened and upon "hands-on" maintenance for operations external to the torus, with personnel access possible to the outside of the torus 24 hours after machine shutdown. Implementation of this philosophy has led to a modularized design concept, and designing to achieve the required access has had a significant impact on the design of the tokamak systems.

7.2 Mechanical Configuration and Maintenance

The original mechanical configuration concept was based on the use of oversized TF coils in order to accomodate a relatively straight-forward assembly/disassembly procedure in which the number of torus sectors was equal to the number of TF coils. A significant improvement in reducing the size of the TF coils has been made in the recent INTOR design study. The present TF coils have a bore size of 6.6 m x 9.3 m. Twelve blanket sectors are assembled with straightline horizontal motion through windows between TF coils. Semipermanent inboard, upper and lower shields form the primary vacuum boundary of the inner surface. The final closure of the vacuum boundary is made on the outer interface of each removable torus sector and semipermanent shield. All superconducting coils are placed in a common cryostat except for the lower outboard PF coil.

It has been recognized that the emphasis upon maintainability is one of the major factors in determining the mechnical configuration. The requirement of a relatively simple torus and divertor assembly/disassembly procedure led to somewhat larger TF coils than was required to meet the ripple criteria. The requirement of maintainability led also to a choice of an all-external PF coil system. The requirement of personnel access for maintenance led to considerably more outboard shielding than would be necessary for component protection. An additional shield of approximately 50 cm is required.

During the present phase of the INTOR Workshop, two design concepts are being compared; one where personnel access is allowed for maintenance 24 hours after shutdown, and the other where personnel access is forbidden, demanding fully remote operations for all maintenance activities. A preliminary conclusion drawn from this study indicates that a minimum outboard shield design does not necessarily improve the all-remote design. A reduction in the outboard shield thickness to achieve the minimum reactor size for the all-remote design appears to result in a significant increase in the reactor building wall thickness, reactor hall activation and nuclear heating to the TF coils. There does not appear to be any significant advantage to be realized by replacing the present "limited personnel access" maintenance concept by a "totally remote" concept.

7.3 First Wall and Blanket

Extensive analysis supports the design of the first wall, blanket and shield. A water-cooled stainless steel first wall is predicted to last the full lifetime of the device, provided that localized melt layers which may result during a plasma disruption are stable.

A tritium producing blanket having solid breeding material covers the outboard and upper surface of the plasma chamber and would produce more than 60% of the tritium to be consumed in INTOR. Li_2O has been adopted as the breeding blanket material. The recommended temperature range for tritium recovery is $400^{\circ}C$ to $650^{\circ}C$. An estimated tritium inventory in the blanket is in the range of 0.3 kg to 1 kg. Extensive studies are also being made on other candidate materials for the blanket, such as Li_2SiO_3 and $Li_{17}Pb_{83}$.

7.4 Tritium Permeation

Investigation of tritium permeation and inventory in the first wall, limiter and divertor indicated large uncertainties in a number of areas. The present best estimate for the steady-state tritium permeation rate to the coolant is in the range of 10^2 - 10^4 Ci/day. However, several methods for separating tritium from

the coolant are available. Since the capital and operating costs of such a separation system are strongly dependent on the process flow rate, which is proportional to the permeation rate and varies inversely with the allowable tritium concentration in the coolant, a reassessment of the date base for predicting the permeation rate and inventory is being made.

8. TECHNICAL BENEFIT

All the participants in a collaborative international project of the INTOR type would undoubtedly want to receive the technical benefit that would result from developing and manufacturing the technologically advanced components, as well as the financial benefit that would result from sharing the costs. The Workshop is evaluating the technical feasibility of partitioning the detailed design, fabrication and construction tasks in such a way that all partners would participate in all technologically advanced aspects.

The modular character of most advanced technology components in INTOR - such as superconducting coils, torus sectors, RF heating launchers and others - would suggest exploiting this modularity for developing that experience. International partitioning in design/fabrication/construction of INTOR implies some obvious advantages deriving from only one international INTOR (reduced overall cost) versus four independent similar national experiments, and also some disadvantages (more demanding management co-ordination and control requirements, some delay in schedule).

Relative to a single national project carried out by domestic industry, an international project in which the detailed design and fabrication of all high technology components was partitioned among four partners would cost about 70% more and would require about two years longer to design and construct, because of the more complex project management requirements. On the other hand, each of the four countries would realize the technical benefits of constructing and operating an INTOR type experiment for only about 40% of the cost of building and operating such an experiment unilaterally.

REFERENCES

[1] INTOR GROUP, "International Tokamak Reactor Phase One" IAEA report STI/PUB/619, Vienna (1982); also Nucl. Fusion, 22, 135 (1982).

[2] INTOR GROUP, "International Tokamak Reactor Phase Two A, Part 1", IAEA report STI/PUB/638, Vienna (1983); also Nucl. Fusion, 23, 1513 (1983).

TABLE 1. INTOR DESIGN PARAMETERS

GEOMETRY	
Chamber major radius, R	5.2 m
Plasma radius, a	1.2 m
Plasma elongation, K	1.6
Plasma aspect ratio, A	4.4
PLASMA	
Average beta,	5.6%
Poloidal beta,	2.6
Average ion temperature, $\langle T_i \rangle$	10 KeV
Average ion density, $\langle n_i \rangle$	$1.4 \times 10^{20}\ m^{-3}$
Energy confinement time, τ_E	1.4s
Plasma current, I_p	6.4 MA
Field on chamber axis, B_T	5.5 T
Safety factor (separatrix), q_I	2.1
Peak thermonuclear power, P_{th}	620 MW(th)
Neutron wall load, P_n	$1.3\ MW.m^{-2}$
OPERATION	
Burn time, Stage I/Stages II and III	100/200 s
Duty cycle, Stage I/Stages II and III	70/80%
Number of pulses (lifetime)	7×10^5
Neutron fluence	$\sim 3 MW.y/m^2$
HEATING: ICRF	
Power at start-up	50 MW
Frequency	85 MHz
FUELLING	
Method	Pellet injection and gas puffing
IMPURITY CONTROL	
Method	single-null poloidal divertor
Power to divertor	80 MW

TABLE 1. (cont.)

FIRST WALL	
Power to first wall (excluding neutrons)	44 MW
Material	Water-cooled SS316
BREEDING BLANKET	
Material	D_2O or H_2O, SS316 Li_2O
SHIELDING	
Inboard (non-breeding blanket and shield)	1.1 m
Outboard (breeding blanket and shield)	1.5 m
TRITIUM INVENTORY	
Breeding blanket	0.5-1.0 kg
Storage	2.3 kg
First wall/divertor	0.1-1.0 kg
Tritium handling systems	1.4 kg
TOROIDAL FIELD COILS	
Number	12
Bore	6.6 m X 9.3 m
Conductor	Nb_3Sn and/or NbTi
Stabilizer	Cu
Maximum field	~11T
POLOIDAL FIELD COILS	
Total flux	110 V s
Breakdown voltage	35V for 0.3 s
Location	external to TF coils
Conductor	NbTi
Maximum allowable field	8 T
POWER SUPPLIES	
Stationary loads	200 MW
Pulsed energy storage	14GJ

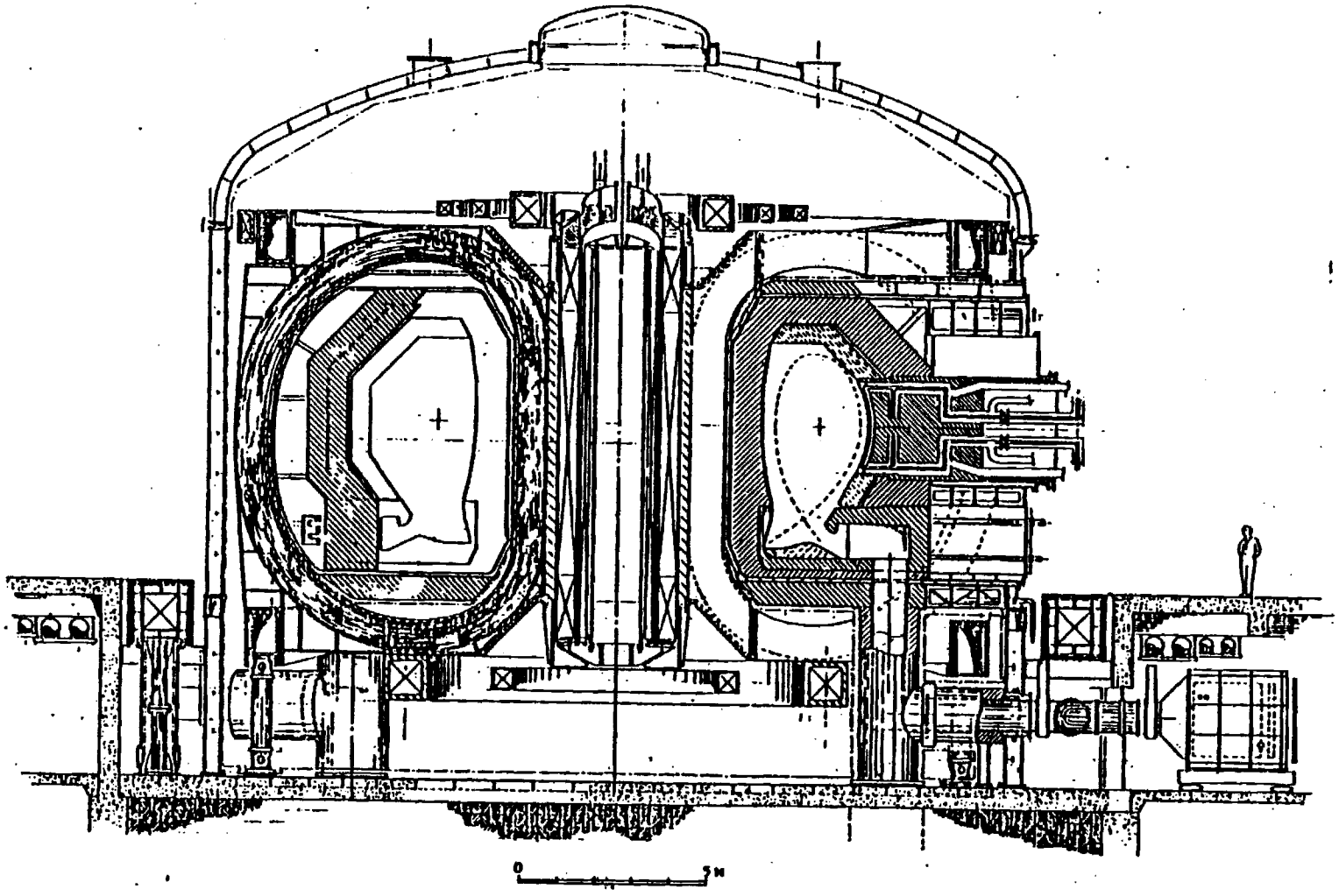

Figure 1. INTOR Phase II Design

INTERNATIONAL ATOMIC ENERGY AGENCY Attachment 4

INTEROFFICE MEMORANDUM

TO: INTOR Participants

DATE 25 October 1984

OUR REF.:

FROM: Steering Committee

YOUR REF.:

SUBJECT: Recommended Design Concept Modifications

1. Based upon conclusions from the critical issues studies and the data base assessment, a set of recommended design concept modifications has been formulated (attachment 1).

2. The indicated groups in attachment 1 are requested to define homework tasks to develop a modified INTOR design concept based on the recommended changes. The effort on each task should be at least at the level of scoping studies to determine consistency at the first level of detail, although more detailed work is appropriate in some cases.

3. The basic philosophy of the main changes is:
   a) to reduce the size and cost where possible, without sacrificing capabilities;
   b) to incorporate LH current ramp-up and transformer recharge capability, while retaining the full volt-second capability;
   c) to utilize the LH current ramp-up, if it proves out, to increase the plasma current;
   d) to take advantage in the future of the increased plasma current, if realized, to improve the confinement potential; and
      i) to improve the potential for achieving the minimum required β, if it is not possible to improve upon the present β - limitations; or
      ii) to realize higher β and be able to reduce the toroidal magnetic field.

4. The major tasks of the homework until the next session should be the design modification tasks and writing the critical issue and data base chapters.

RECOMMENDED DESIGN MODIFICATIONS

1. Non-inductive current ramp-up and transformer recharge with LH should be incorporated. The full present volt-second capability should be retained. (A,B,C,G)

2. RF startup assist should be incorporated into the design. Maintain ~35V loop voltage requirements. (B,C,G)

3. The plasma current should be increased to 8 MA. (C,F,G)

4. The inboard shield thickness should be decreased by 10-15 cm. (G,H)

5. The major radius should be reduced to 5.0 m. Consider reducing inboard shield thickness, increasing current density in TFC and increasing field in OHC. Do not reduce transformer flux capability. (G,C,F,H)

6. The 2 operating scenarios (inductive 6.4 MA & non-inductive 8.0 MA) should be optimized to minimize PFC power supply requirements. (B,C,F,A)

   a. extend heating time 6→15-20s.

   b. extend current ramp-up time.

   c. extend cool-down time 6→15-20s.

   d. extend current ramp-down time.

7. The modified poloidal divertor (shortened) should be incorporated. (A,G)

8. Active control coils should be incorporated. (C,G)

9. A tritium-producing blanket covering ~60% of the plasma chamber surface area on the top and outboard should be retained. (H)

10. The TFC size should be reduced to the ripple limit of $\pm$ 1.2%. (G) Maintain 12 TFC

11. The fluence should be 3 $MW.y/m^2$. (H)

12. A design concept and performance analysis should be prepared for the impurity control system. (A)

13. A design concept and performance analysis should be prepared for the rf heating, current-drive and startup assist systems. (B)

14. Alternate config. maint. & assy. schemes. Investigate (scoping) this phase, with aim of possibly changing in next phase. (G)

DRAFT

RECOMMENDATION ON CONTINUATION OF INTOR WORKSHOP

TO IFRC FROM INTOR STEERING COMMITTEE

The activities of the INTOR Workshop since its inception have been in three principal areas:

1) the development of a concept for a next-step tokamak experiment;
2) assessment of the worldwide scientific and technological data base that would support such an experiment; and
3) concentrated analysis of certain critical technical issues that affect the feasibility of this next-step tokamak experiment.

The evolving INTOR design concept has served to define the requirements for a next-step tokamak experiment and has served to advance the design concepts for a number of components and systems; these requirements and concepts have guided a number of national design efforts as well. The results of the present phase will again define an international consensus on the characteristics of a next-step tokamak experiment which could be initiated within the next few years. The data base assessment and definition of required R & D has stimulated and guided R & D programmes worldwide. The critical issues studies have focused international effort on the most important technical issues faced by a next-step tokamak.

We propose a continuation in the above three principal areas for the next phase, but with a modification in effort and a major change in emphasis and guiding principles in the first area. Up through the present phase, INTOR has been conceived as the "maximum reasonable step" beyond the present generation of large tokamaks and has been based upon science and technology that could be reasonably extrapolated from the present data base. As such, INTOR represents the next logical step based upon a projection of present line of thought in tokamak physics and the associated technologies. This type of design concept will be retained as the reference INTOR concept, but the emphasis in the concept development area will change to address other issues, as discussed in the following paragraphs.

Progress in fusion research has now reached the threshold of demonstrating physical feasibility, which has stimulated anew the discussion of whether a fusion reactor based upon projections of the present thinking in tokamak physics and the associated technologies represents the maximum ultimate potential of fusion. This discussion, in turn, raises the question of whether or not the present INTOR concept is the proper step along the path to the most attractive end product of fusion development. It is possible to conceive of more attractive physics parameters and technological solutions for tokamak reactors which are not yet reasonably extrapolatable from the existing data base.

Furthermore, it now appears likely that a major next-step tokamak experiment will only be initiated, either unilaterally or internationally,

after the essential information is obtained from the present generation of large tokamaks, which is expected late in this decade.

These circumstance make it appropriate to recommend that a significant part of emphasis in the INTOR workshop be changed, temporarily, to conceiving more innovati concepts that would lead to a more attractive end product (e.g. eventual tokamak reactor), even if such concepts are not supported by reasonable extrapolation from the present data base. In this regard, essential functions of the INTOR workshop during the next phase would be to:

1) seek out and stimulate innovations that would fundamentally improve the tokamak as a reactor concept;
2) evaluate such innovations and others within an overall systems context by assessing their impact on the (reference) INTOR design concept;
3) define and stimulate the R & D required to determine if the innovations with a large favourable impact are technically feasible;
4) develop an "innovative" design concept for comparison with the INTOR design concept for a next generation tokamak experiment; and
5) evaluate how the INTOR design concept could be improved by those innovations which are supported by a reasonable extrapolation from the then-existing data base or how INTOR could be used, as a test bed for those favourable innovations which are not yet so supported, thus making the INTOR design a more suitable step towards the most attractive end product.
6) provide a somewhat more detailed and updated definition of a DEMO and evaluate whether INTOR will still be able to fulfill its function as the single step between the present generation of large tokamaks and DEMO.

We further recommend that the INTOR workshop continue the studies on the most critical technical issues, with some modifivations in scope of effort. These activities are important for the advancement of any next-step tokamak experiment, be it "innovative" or "standard". Through these activities the INTOR workshop provides an important focus for technical studies in critical areas. The workshop will keep under continuous review new developments in the scientific and technological data base.

The length of the next phase should be two years, from July 1985 through June 1987. Four to five working sessions of two to three weeks duration are envisaged, with the possibility that an additional short session may be necessary after this period for proofreading the report. The present workshop mode of operation should be continued, and about the same number of participants (eight per partner) should be involved. The home base effort should be at the level of previous phases.

The results of the next phase will be documented in a report to be published at the end of the phase.

The recommended tasks of the workshop are discussed in more detail in the attachment.

TASKS PROPOSED FOR NEXT PHASE OF INTOR WORKSHOP

1. Tokamak Concept Improvements

(a) The Workshop will undertake to stimulate the origination of innovations that will significantly improve the prospects of tokamak development leading to an attractive end product – a viable tokamak reactor. For the most part, such innovations will not be supported by the existing data base. Examples may be the following: first-wall design and replacement procedures that allow rapid replacement; highly shaped plasma cross sections to access the second stability regime; mechanisms for steady-state current drive; mechanisms for elimination of disruptions; advanced superconducting magnets that can operate at higher temperatures; non-organic magnet insulators that can tolerate higher radiation doses; higher strength structural materials that can tolerate higher heat fluxes and neutron fluences; low activation materials; etc.

(b) The innovations will be evaluated by assessing their impact on the INTOR design concept and by projecting their benefit to a future tokamak reactor concept. The net benefit of each innovation will be evaluated, taking into account impact on other components.

(c) For those innovations with a large favourable impact, the R & D that is required to determine the feasibility of the innovation will be identified.

(d) Towards the end of the next phase, those innovations that would significantly improve a tokamak reactor will be collected, an evaluation will be made of which of these innovations could conceivably be supported by results from an R & D program by

the early part of the next decade. These latter inovations will then be incorporated into an "innovative", alternative design concept.

(e) An evaluation will be made of how the INTOR design concept:

(i) would be improved by those favourable innovations which are supported by a reasonable extrapolation from the then-existing data base; and

(ii) could be used as a test bed for the other favourable innovations which are not so supported.

(f) A somewhat more detailed and updated definition of DEMO will be provided and an evaluation will be made of whether INTOR will still be able to fulfill its function as the single step between the present generation of large tokamaks and DEMO.

2. Critical Issues

Continuing its present work, the INTOR workshop will focus international resources on six critical technical issues, concentrating upon the most essential aspects of the following:

(a) Beta - Analysis and optimization studies focussed on defining practical ways by which beta values larger than that which would be inferred from the present data base can be achieved in INTOR. Analysis of engineering impact upon INTOR design concept of achieving such configurations.

(b) Impurity Control - Analysis of poloidal divertor, pumped limiter and innovative schemes, Review of experimental data (plasma, atomic and surfact physics) and comparison with calculations. Development and comparison of advanced computer models for divertors, pumped limiter, scrape-off region, impurity transport interaction of the impurity control system with the bulk plasma, etc.

(c) RF Heating and Current Drive - Analysis of RF bulk heating, current drive and start-up assist. Review of experimental data and comparison with calculations. Development and comparison of advanced computer models for rf penetration, heat deposition.

current drive, current initiation, profile control, etc. Examination of innovative current-drive mechanisms. Applications to next-generation tokamak plasmas. Development of launcher concepts. Extention of work initiated in current phase of INTOR workshop.

(d) Transient Electromagnetics - Analyis of transient electromagnetic effects (involving the coil systems, torus structure and plasma) associated with start-up, stabilization and disruptions. Development and comparison of advanced computer models. Extension of computer model to include distributed current, deformable plasma models. Extension of studies to include disruption control and start-up as well as vertical stabilization and shape control. Development of criteria for torus EM requirements.

(e) Configuration and Maintenance - Examine innovative concepts for configuration, assembly and maintenance that would fundamentally improve the tokamak concept. Examine schemes for rapid replacement of limiter/divertor and first-wall.

(f) Blanket and First Wall - Evaluation of a broad range of blanket concepts for INTOR, eventually selecting one preferred concept. Continued development of concepts and lifetime analysis methods for first wall (including limiter and divertor) components. Comparison of calculational models, nuclear data, etc.

3. Data Base Assessment

New worldwide experimental and theoretical results in tokamak plasma physics and related technologies will be kept under continuous review.

PROPOSED ORGANIZATION OF INTOR WORKSHOP DURING THE NEXT PHASE

Tentatively, the INTOR Workshop would be organized during the next phase into six critical issue groups and two disciplinary groups, with tasks as follows. Each member of the Workshop would belong to one or more critical issue group and one disciplinary group. Intergroup meetings will be an essential feature of the organization. The organization will be finalized after detailed planning.

CRITICAL ISSUES GROUPS

A. Impurity Control

1. Impurity Control Critical issue b)
2. Innovative impurity control schemes
3. Review and R&D results

B. Beta Enhancement and Confinement

1. Beta critical issue a)
2. Confinement
3. Review new R&D results

C. RF Heating and Current Drive

1. RF critical issue c)
2. Innovative Current drive schemes
3. Review new R&D results.

D. Electromagnetics

1. Transient electromagnetics critical issue d)
2. Innovative magnet concepts
3. Review new R&D results

E. Configuration and Maintenance

1. Configuration and maintenance critical issue e)
2. Systems integration
3. Review new R&D results

F. Blanket & First-Wall

1. Blanket and first-wall critical issue f)
2. Innovative materials and nuclear systems
3. Review new R&D results

DISCIPLINARY GROUPS

G. Physics

1. Interaction among groups A-C.
2. Physics innovations not covered by groups A-C.
3. Review new physics R & D results not covered by groups A-C.

H. Engineering

1. Interaction among groups D-F.
2. Engineering innovations not covered by groups D-F.
3. Review new engineering R & D results not covered by groups D-F.
4. RF launcher design concepts.

E-25-621

COVER SHEET

FOR TRIP REPORTS SUBMITTED TO THE

OFFICE OF ENERGY RESEARCH

Destination(s) and Dates for
Which Trip Report Being Submitted: December 1-12, 1986

Name of Traveler: Weston M. Stacey, Jr.

Joint Trip Report [ ] Yes

[X] No

If so, Name of Other Traveler(s): _____

Georgia Institute of Technology
A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
NUCLEAR ENGINEERING AND HEALTH PHYSICS PROGRAMS
SCHOOL OF MECHANICAL ENGINEERING

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CHERRY EMERSON BUILDING
GEORGIA INST. OF TECH.
ATLANTA, GEORGIA 30332 U.S.A.

January 12, 1987

REPORT OF FOREIGN TRAVEL

TRAVELER: Weston M. Stacey, Jr. WMS
Georgia Institute of Technology
Atlanta, GA 30332

DESTINATION: INTOR WORKSHOP
VIENNA, AUSTRIA

DATES: December 1-12, 1986

SUMMARY

The traveler attended session XIV of the Phase 2 INTOR Workshop as the US Steering Committee member. The list of attendees and agenda are attached.

This session had two principal objectives: 1) to arrive at preliminary conclusions on the critical issues and innovations studies which have been ongoing for the past year; and 2) to plan in detail how to carry out a new task of conducting critical analyses of existing INTOR-like designs. These two objectives were accomplished. The preliminary conclusions from the ongoing studies were documented in the internal working documents of the Workshop, and work required to arrive at final conclusions was defined. A new work plan for 1987 was developed (attached). A Specialists' Meeting was planned to gather information as a first step in the analysis of INTOR-like designs (attached).

WMS/sgj
Attachment

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1430P
Department: RI
Division: RIRL

Issue No. 2
1986-12-01
F1-TC-433.35

NOTIFICATION OF A MEETING HELD AT HEADQUARTERS

Title of Meeting: XIVth INTOR Workshop

Opening Meeting: 9.30 a.m., Room V,
Responsible Officers: M. Leiser
Ext 1756, A2375
A. Shurygin
Ext 1702, A2305
Secretary: N. Pawel, A2374, Ext. 1

Dates Inclusive: 1-12 December 1986
Place: B-08, B-09

PARTICIPATING AND DESIGNATING MEMBER STATES AND ORGANIZATIONS	ADDRESSES ABROAD	IN VIENNA	FOR THE PERIOD
EURATOM			
G. Grieger	Max Planck Institut für Plasmaphysik D-8046 Garching FRG	Rothensteiner Apartments Neustiftgasse 66 1070 Vienna 93 96 43-0	1-12 December 1986
E. Coccorese	NET TEAM Max Planck Institut für Plasmaphysik D-8046 Garching FRG	"	1-5 December 1986
F. Engelmann	NET Team Max Planck Institut für Plasmaphysik D-8046 Garching FRG	"	1-12 December 1986
F. Farfaletti-Casali	Joint Research Centre Ispra Establishment I-21010 Ispra (Varese) Italy	"	1-12 December 1986

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E. Salpietro	NET TEAM Max Planck Institut für Plasmaphysik D-8046 Garching FRG	"	1-12 December 1986
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C. Vieider	NET TEAM Max Planck Institut für Plasmaphysik D-8046 Garching FRG	"	1-12 December 1986
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J. Miller	Lawrence Livermore National Laboratory P.O.B. 808 Livermore, California 94550	1-12 December 1986
M. Peng	Oak Ridge National Laboratory P.O. Box Y Oak Ridge, Tennessee 37831	1-12 December 1986
N. Pomphrey	Princeton Plasma Physics Laboratory P.O. Box 451 Princeton, NJ 08544	1-12 December 1986
D. Post	Princeton Plasma Physics Laboratory P.O. Box 451 Princeton, New Jersey 08544	1-12 December 1986
D. Smith	Argonne National Laboratory 9600 South Cass Avenue Argonne, Illinois 60439	1-12 December 1986
P. Spampinato	Fusion Engineering Design Center Oak Ridge National Laboratory P.O. Box Y Oak Ridge, TN 37830	1-12 December 1986

5 December 1986

AGENDA

Session XIV

INTOR Workshop Phase IIA/Part 3

December 1 - 12, 1986

Monday, December 1

09:00 - 09:30	Steering Committee
09:30 - 10:30	Plenary Session (National Summary)
11:00 - 11:30	Exchange of National Reports
11:30 - 17:00	Reading Reports
15:00 - 15:30	Coordinating Committee

Tuesday, December 2

AM & PM	Group Discussion on Innovations and Critical Issues

Wednesday, December 3

AM & PM	Group Discussion
17:00 -	Coordinating Committee

Thursday, December 4

AM & PM	Group Discussion

Friday, December 5

AM	Group Discussion
14:00 - 17:00	Plenary Session (Preliminary Conclusions/Decisions on Critical Issues and Innovations, and DEMO-requirements)
17:00 -	Coordinating Committee

Monday, December 8

AM & PM	Group Discussion

Tuesday, December 9

AM & PM	Group Discussion (Detailed Task Definition)
17:00 -	Coordinating Committee

Wednesday, December 10

9:00 - 10:30	Plenary Session (Critical Analyses)
11:00 - 17:00	Group Discussion and Report Writing

Thursday, December 11

AM & PM	Group Discussion and Report Writing
17:00 -	Coordinating Committee

Friday, December 12

AM	Typing and Copying of Summary
14:00 - 15:30	Plenary Session

REMARKS

1. The objects of this session are to reach
   (a) Preliminary Conclusions/Decisions on Critical Issues and Innovations;
   (b) Detailed Task Definition on Critical Analysis of INTOR-like Designs, and
   (c) Final Conclusions on DEMO requirements.
2. Preliminary conclusions/decisions should be reached in the first week on (a).
3. In the second week, Critical Analysis of INTOR-like Designs discussion should have priority in scheduling the meetings.

INTERNATIONAL ATOMIC ENERGY AGENCY

INTEROFFICE MEMORANDUM

TO: INTOR WOrkshop Participants

DATE 12 December 1986

OUR REF.:

FROM: INTOR Steering Committee

YOUR REF.:

SUBJECT: INTOR Work Plan for 1987 - Overview

The IFRC, at their November 1986 meeting in Kyoto, recommended a change of objectives for the INTOR Workshop during 1987. These recommendations are attached. This change in objectives has resulted in a replanning of the INTOR work and schedule for 1987 and the convening of an INTOR-related IAEA Specialists' Meeting. This schedule is also attached.

The first objective recommended by the IFRC - to continue the data base assessment, critical issue analysis, and innovation assessment - is part of the ongoing INTOR work. Homework tasks have been planned at this session XIV to enable final conclusions and recommendations to be reached during session XV in July. Drafts of the INTOR report chapters on this work will be prepared at home between sessions XV (July) and XVI (November).

The second objective recommended by the IFRC - to conduct critical analyses of existing INTOR-like designs (i.e. FER, INTOR, NET, OTR, TIBER) with the aim of preparing a useful information base for future design work for the ETR - has resulted in a revised work plan as follows. An INTOR-related IAEA Specialists' Meeting will be convened in March to bring together some key members of the FER, INTOR, NET, OTR, TIBER teams for the purpose of compiling in a common format and discussing information about these designs. Homework tasks have been prepared at this session XIV to guide the compilation of information as input to this Specialists' Meeting.

Subsequently, the INTOR Workshop will conduct critical analyses of these INTOR-like designs, based upon input from the Specialists' Meeting and INTOR homework tasks which have been defined at this session for session XV (July), Preliminary conclusions about how different objectives, different choices of main physics and engineering "driving" features, and different assumptions about physics and engineering design constraints impact designs should be reached at session XV (July). Homework tasks should then be defined at session XV to enable final conclusions on these matters to be reached at session XVI (November). Documentation of this material for the INTOR report should be done at session XVI, allowing for the possibility of same drafting between sessions XV and XVI.

A final draft of the Phase 2A, Part 3 INTOR report should be prepared and turned over to the editors at session XVI (November).

IFRC RECOMMENDATIONS TO THE DIRECTOR GENERAL

OBJECTIVES FOR 1987 WORK

The IFRC adopted in April, 1985 a programme of work until the end of 1987. In view of the proposals from the USSR and the US for a new design and supporting research initiative, and the preliminary positive reactions from the EC and Japan, the IFRC recommends a change of the 1987 objectives from those previously adopted.

The new objectives should be:

- To continue the data base assessment, critical issue analysis, and innovation assessment,
- To discontinue the task of incorporating innovations into and updating the design concept.

and, instead

- To conduct critical analyses of the existing INTOR like designs with the aim of preparing a useful information base for future design work for the ETR.

The final report should be completed by the end of 1987 using the resources originally agreed upon.

Attachement 2

MEETING DATES 1987

March 23-27	INTOR-related IAEA Specialists' Meeting on Information on Engineering Test Reactor Deisgn Concepts
July 13-24	INTOR Session XV
November 9-20	INTOR Session XVI

INTERNATIONAL ATOMIC ENERGY AGENCY

INTEROFFICE MEMORANDUM

TO: IAEA Scientific Secretariat for INTOR Workshop

DATE 10/12/1986

OUR REF.:

FROM: INTOR Steering Committee

YOUR REF.:

SUBJECT: INTOR-Related Specialists'Meeting

The INTOR workshop has been asked by the IFRC to take on the new task during 1987 "to conduct critical analyses of the existing INTOR-like designs with the aim of preparing a useful information base for future design work for the ETR". In planning for this new task, the INTOR Steering Committee has determined that it is essential that information on these designs be compiled on a common basis and discussed by persons knowledgeable of the designs, at the earliest possible date.

We request that the IAEA convene an INTOR-related Specialists'Meeting on Information on Engineering Test Reactor Design Concepts to be held March 23-27,1987. This meeting would bring together the principal people from the four national design teams (FER-Japan, NET-EC,OTR-USSR, TIBER-US) with their counterparts in the INTOR workshop. Certain information should be prepared as input to this Meeting. Detailed plans for the meeting are attached. These should be distributed with the invitation to the Specialists'Meeting. The INTOR Steering Committee will act as the organizing committee for the Meeting.

We anticipate that each of the four "countries" (EC, Japan, USA, USSR) will send at least five participants representing their national design team and 2-3 additional people representing INTOR, for a total of $\sim$ 8-9 participants per country. Thus, we request that you provide office space for about 35 people. Four meeting rooms, one of which that can accommodate 35 people, will be needed. Commissary privileges should be arranged for the specialists.

Attachment 1

INTOR-Related IAEA Specialists' Meeting on

Information on Engineering Test Reactor Design Concepts

GENERAL PURPOSE: Provide information to enable INTOR Workshop to conduct critical analysis of existing INTOR-like (FER, INTOR, NET, OTR, TIBER) designs with the aim of preparing a useful information base for future design work for ETR.

SPECIFIC PURPOSES:

1. Document and compare objectives of different design concepts. Identify how objectives affect design concepts and why they were chosen.
2. Document and compare the present design specifications (i.e. major parameters, heating and impurity control methods, materials) for the different design concepts.
3. Identify the main features which "drive" the different design concepts (e.g. non-inductive current drive, assembly and maintenance philosophy).
4. Document and compare the physics and engineering design constraints (e.g. beta limits, radiation damage limits, stress limits) used in the different design concepts and discuss why they were chosen.

DATE: March 23-27, 1987

ORGANIZING COMMITTEE: S. Mori- Chairman, G. Grieger, B. Kadomtsev, W. Stacey

ORGANIZATION: The Specialists' Meeting will be organized into four groups,

Group 1 Physics
(beta, confinement, current drive, heating, impurity control, etc.)

Group 2 Electromagnetic, Heating & Current Drive Technologies
(TF coil, PF coil, rf system, NB system, transient EM, etc.)

Group 3 Mechanical and Configuration
(Mech. design, assy. & maintenance, configuration, etc.)

Group 4 Nuclear
(FW, Blkt, shld, divertor, tritium, etc. technologies)

PARTICIPANTS: Each country is requested to send the people from their national design team with major responsibilities in each of the four areas into which the meeting will be organized plus a person with overall responsibility for the national design concept, plus 2-3 members of the INTOR workshop to be suggested by the INTOR steering committee member for that country. A total of 8-9 participants per country is desired.

PREPARATION: Each design team (FER,INTOR,NET, OTR, TIBER) should prepare the following documents as input to the Specialists' Meeting.

1. A list of technical and programmatic objectives and a brief($\sim$1-2 pages) written discussion on why these objectives were chosen and how they have affected the design concept. (The INTOR objectives are included as appendix 1 as an example.)
2. A list of the design specifications, following the format for the INTOR specifications included as appendix 2.
3. A brief discussion ($\sim$2 pages) of the main features which "drive" the design concept. (The INTOR discussion is included as appendix 3 as an example.)
4. A list of the physics and engineering design constraints, following the format of appendix 4. A brief discussion of why particular values were chosen for the design constraints.
5. Such other materials (e.g. reports) describing the design concept as are available.

REPORT OF MEETING: A report will be prepared during the Specialists' Meeting. This report will include:

1. A list of the technical and programmatic objectives for each design concept (FER, INTOR, NET, OTR, TIBER), a discussion of how different objectives cause differences in design concept, and a discussion of why the different objectives were chosen. (INTOR SC + Design team leaders)
2. A list of the design specifications for each design concept, using a common format. (Groups 1-4)
3. A discussion of the main features which "drive" each design concept.
4. A list of the physics and engineering design constraints for each design concept, using a common format, and a discussion of why particular values of design constraints were chosen. (Groups 1-4)

BACKGROUND: The International Fusion Research Council (IFRC), at their November meeting in Kyoto, recommended a change in objectives for the work of the INTOR Workshop during 1987. In view of the proposals from the USSR and the US for a new Engineering Test Reactor (ETR) design activity and supporting research initiative, and the preliminary positive reactions from the EC and Japan, the IFRC recommended that the INTOR Workshop conduct critical analyses of the existing INTOR-like designs with the aim of preparing a useful information base for future design work for the ETR. The IFRC further authorized the INTOR Workshop to ask the IAEA to hold an INTOR-related Specialists' Meeting to assist them with this new task.

The existing INTOR-like design concepts differ for a number of reasons, foremost among which are:

1) differences in objectives;
2) differences in physics and engineering design constraints;
3) differences of opinion about the incorporation of advanced, or innovative features and about the associated risks;
4) differences in the "philosophy" used to "optimize" the designs, and
5) differences in the level of detail and self-consistency.

The critical analysis of the existing INTOR-like designs can best be carried out by:

1) comparing the objectives among the designs and identifying how different objectives would cause differences in a design;
2) identifying the main "driving" features which differ among the designs and evaluating the design impact, risks, benefits, etc. associated with these features; and
3) comparing the physics and engineering design constraints among the designs and evaluating how different constraints would cause differences in a a design.

It is desirable that the principal people involved in all of the INTOR-like designs (FER, INTOR, NET, OTR, TIBER) be involved in the critical analysis in order that the most useful information base be prepared for future design work for ETR.

AGENDA

INTOR-Related IAEA Specialists' Meeting on Information on Engineering Test Reactor Design

MONDAY, March 23, 1987

9.00 - 9.30 a.m.	Introduction	PLENARY
9.30 -12.30 a.m.	Overviews (FER, INTOR, NET, OTR, TIBER)	PLENARY
2.00 - 3.30 p.m.	Comparison of Design Objectives	PLENARY
3.30 - 5.00 p.m.	Comparison of Design Specifications	GROUPS

TUESDAY, March 24, 1987

9.00 -12.00 a.m.	Comparison of Main Design Features	GROUPS 1-4
2.00 - 5.00 p.m.	Comparison of Main Design Features	PLENARY

WEDNESDAY, March 25, 1987

9.00 -12.00 a.m.	Comparison of Design Constraints	GROUPS 1-4
2.00 - 5.00 p.m.	Comparison of Design Constraints	GROUPS 1-4

THURSDAY, March 26, 1987

9.00 -12.00 a.m.	Report Preparation
2.00 - 4.00 p.m.	Report Preparation

FRIDAY, March 27, 1987

9.00 -12.00 a.m.	Meeting Summary	PLENARY
2.00 -4.00 p.m.	INTOR Team Meeting	

Georgia Institute of Technology
A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
NUCLEAR ENGINEERING AND HEALTH PHYSICS PROGRAMS
SCHOOL OF MECHANICAL ENGINEERING

Please reply to:

NUCLEAR ENGINEERING AND
HEALTH PHYSICS PROGRAMS
CHERRY EMERSON BUILDING
GEORGIA INST. OF TECH.
ATLANTA, GEORGIA 30332 U.S.A.

November 10, 1986

US Department of Energy
Oak Ridge Operations
P.O. Box E
Oak Ridge, Tennessee 37831

Dear Sir:

Subject: Foreign Travel

Approval is requested for Dr. Weston M. Stacey, Jr. (Georgia Institute of Technology, Atlanta, Georgia 30332) to travel to Vienna, Austria to participate as leader of the US delegation to the INTOR Workshop session to be held at IAEA headquarters December 1-12, 1986. This session is an integral part of work performed under financial assistance agreement Number DE-FG-05-74ER52107, and funds are available. The estimated cost to DOE is $2800 ($1500 travel plus $1300 expenses). Travel will commence November 28, 1986 and end January 4, 1987; the period December 13 - January 3 is personal time. Dr. Stacey will hold discussions with other INTOR Workshop participants from the EC, Japan and USSR.

Weston M. Stacey, Jr.

Approved
J.A. Brighton, Director M.E.

WMS/sgj
Attachments
Copy: A. Opdenaker DOE

Telephone: 404-894-3720 Telex: 542507 GTRIOCAATL Fax: 404-894-3120 (Verify 404-894-3751)

GEORGIA INSTITUTE OF TECHNOLOGY ATLANTA, GEORGIA 114103

REQUEST FOR AUTHORITY TO TRAVEL ON OFFICIAL BUSINESS OF THE INSTITUTE

VELER:

Name Weston M. Stacey, Jr.

Soc. Sec. No 258-52-3018

Title Professor

Department ME/NE&HP

Residence 825 Woodley Dr., N.W. (Street)

Atlanta, Georgia (City and State) Fulton (County) 30318 (Zip)

ESTIMATED COST OF TRIP:

Transportation	$ 1,500
Meals and Lodging	1,300
Other Travel	
TOTAL TRAVEL	$ 2,800
Registration Fees	$ 0
TOTAL ESTIMATED COST	$ 2,800

E OF TRANSPORTATION — CHECK BOX OR BOXES

- [ ] Commercial Airline
- [ ] Personal Automobile
- [ ] Rental Vehicle
- [ ] Other

TRAVEL REQUISITION NUMBER: INTOR

AMOUNT

TOTAL CHARGES $

of Departure 11/28/86 Date of Return 1/4/87

ary Vienna/London

ose of Trip INTOR Workshop/Vacation

u anticipate that expenses for meals and/or lodging will exceed the standard amounts prescribed for these items? ☐ Yes ☒ No

wer is yes, justify the excess

u want your air travel expense to be charged directly to the Institute through the approved air travel agency? ☒ Yes ☐ No

Date 11/10/86 Signed (Traveler)

ROVED BY:

Date Dept. Head or School Dir.

Date Dean or Director

Date Appropriate V/P

Date President (Only as Required)

Date V/P Bus. and Finance (For Fiscal Review)

OUNTING ONLY: Date Processed Initials

PLEASE FOLLOW THESE PROCEDURES:

. The employee requesting travel should complete this form. All information should be typewritten.

. First send all copies to the official authorized to approve your travel request. Each approving authority in turn should forward all copies to the next approving authority.

. Not all signatures are needed on each travel request. Consult directives for approval requirements.

. The Business Office will encumber the necessary funds and distribute all copies of this form. It will hold the original of the form to be matched with the travel expense statement for payment.

. Please notify the Business Office promptly in writing if this authority is not exercised so that encumbered funds may be released

GEORGIA INSTITUTE OF TECHNOLOGY

REQUEST FOR TRAVEL ADVANCE

TO: Office of Contract Administration

Name: Weston M. Stacey, Jr. Date: November 10, 1986

School/Laboratory: NE&HP/ME Research Related Purpose: Attend INTOR Workshop.

Destination: Vienna, Austria

Date of Departure: 11/28/86 Amount Encumbered on TA minus airfare: $1,300.00

Date of Return: 1/4/87

Amount Requested: $1,300.00

In accepting this advance, the undersigned agrees that the funds will be used only for the purpose stated above, and repayment will be made promptly after receipt of reimbursement for actual travel expense by the Controller's Office.

Upon demand, however, the undersigned promises to pay to the order to THE GEORGIA TECH RESEARCH INSTITUTE, at its office or at such place as it may designate and notify undersigned, the amount shown hereon as Amount.

The Georgia Tech Research Institute may, at its election and without the concurrence of the undersigned, assign all rights and privileges of this promissory note to the Georgia Institute of Technology.

Signature

Approved:

School/Laboratory Director

GTRI USE ONLY: Last Day for Payback

GTRI Check No. ( ) PC ( ) GF Date Issued:

R/8-14-86

GEORGIA INSTITUTE OF TECHNOLOGY
NUCLEAR ENGINEERING AND HEALTH PHYSICS PROGRAMS

REQUEST TO TRAVEL OR BE ABSENT FROM CAMPUS

TRAVELER Weston M. Stacey DATE 11/10/86

DATE OF DEPARTURE 11/28/86 DATE OF RETURN 1/4/87

ITINERARY Vienna/London

PURPOSE INTOR Workshop/Vacation

I CAN BE REACHED AT THE FOLLOWING:

LOCATION Vienna/London

PHONE NUMBER 011-43-222-2360 / 011-441-422-8756

CLASS ARRANGEMENTS

COURSE NUMBER	DATE AND TIME	CLASS TO BE TAKEN BY
NE4610	MWF-12	Thomas

SIGNATURE OF TRAVELER

APPROVED:

W. M. STACEY, CHAIRMAN

ENERGY RESEARCH FOREIGN TRAVEL REQUEST COVER SHEET

1. TRIP NO.: ------
2. HQ APPROVAL REQUIRED: Y
3. TRAVELER'S NAME: Weston M. Stacey
4. EMPLOYER: Ga. Inst. Technology
5. EMPLOYER DESIGNATION: -
6. DEPARTURE DATE: 11/28/86
7. RETURN DATE: 01/04/87
8. NO. DAYS PERSONAL: 20
9. FISCAL YEAR: 86
10. DESTINATION(S): AU
11. TRIP CATEGORY: IC/__/__/__

    -- __/__/__/__

    -- __/__/__/__

    -- __/__/__/__

    -- __/__/__/__
12. CONFERENCE NO.: ----
13. B&R CLASSIFICATION: -- ----
14. ESTIMATED COST TO DOE: $ 2,800
15. ESTIMATED COST TO OTHERS: $---,---
16. JUSTIFICATION/PURPOSE: Attend INTOR Workshop and Vacation.

COVER SHEET

FOR TRIP REPORTS SUBMITTED TO THE

OFFICE OF ENERGY RESEARCH

Destination(s) and Dates for
Which Trip Report Being Submitted: Vienna, Austria - September 9-13, 1985

Name of Traveler: Weston M. Stacey, Jr.

Joint Trip Report [ ] Yes

[x] No

If so, Name of Other Traveler(s): —

—

Georgia Institute of Technology
A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
SCHOOL OF MECHANICAL ENGINEERING

Please reply to:

NUCLEAR ENGINEERING AND
HEALTH PHYSICS PROGRAM
CHERRY EMERSON BUILDING
GEORGIA INST. OF TECH.
ATLANTA, GEORGIA 30332 U.S.A.

October 29, 1985

Report on Foreign Travel

Grant No. DE-FG05-84ER-52107

Traveler - Weston M. Stacey, Jr.

Project No. E-25-621

Visit to Vienna, Austria - (9/9-13/85)

The traveler attended the INTOR Workshop as the US Steering Committee Member. The primary purposes of this meeting were to proofread the edited final version of the Phase 2A Part 2 INTOR report and to plan in detail the work of Phase 2A Part 3. The purposes were successfully accomplished. Emphasis in the planning was placed on preparations for the INTOR - related IAEA Specialists Meeting on Tokamak Concept Innovations to be held in January, 1986.

WMS/sgj

Telephone: 404-894-3720 Telex: 542507 GTRIOCAATL Fax: 404-894-3120 (Verify: 404-894-4850)

F-N-1500.3

REQUEST FOR APPROVAL OF OFFICIAL FOREIGN TRAVEL

hous Editions are Obsolete)

PART A—To be completed by traveler's administrative officer

lget and Reporting Classification to be charged: E-25-621
e Appendix 1101, Part II)

ic category of a trip, either I, II, or III, is determined by the initions set forth in Appendix 1501, Part VIII)

TRIP CATEGORY: _____

PART B—To be completed by traveler

NAME OF TRAVELER Weston M. Stacey, Jr. CITIZENSHIP	c. DATE AND PLACE OF BIRTH July 23, 1937; Birmingham, AL USA d. PASSPORT NUMBER (If available)
HOME ADDRESS 825 Woodley Dr., N.W. Atlanta, Ga. 30318	b. BUSINESS ADDRESS Georgia Institute of Technology Atlanta, Ga. 30332
EMPLOYER Georgia Institute of Technology Atlanta, Ga. 30332 USA	c. CONTRACT NUMBER
ORGANIZATIONAL UNIT School of ME/NE & HP	d. POSITION TITLE (Including profession) Callaway Professor Nuclear Engineering

PURPOSE OF TRAVEL—Include all pertinent background information leading to travel and attach copies of invitations and correspondence regarding travel to present papers, give speeches, or to attend conferences or symposia. Also identify by name and organization other DOE and contractor personnel who, to the traveler's knowledge, are going to the same destination at the same time as the traveler. In addition, specify nature and classification of information to be disclosed including titles of papers to be presented; nature of information to be obtained at each of the places to be visited and conferences to be attended and its relation to traveler's work. Travelers are responsible for obtaining clearances for papers or speeches when necessary. If more space is required, attach a separate sheet. NOTE: IF THIS INFORMATION IS CLASSIFIED BE SURE TO CLASSIFY THIS FORM APPROPRIATELY.

Attend INTOR Workshop session at IAEA in Vienna as US Steering Committee member.

...mation Statement. Collection of the information is authorized by the Department of Energy Organization Act of 1977, P.L. 95-91. ... personal information requested is mandatory to support authorization for official travel to foreign countries, and to obtain a

...rnished will be used by DOE to authorize travel and payment of travel expenses; by the Department of State to issue a passport; ... Accounting Office to audit and verify the accuracy and legality of disbursements.

ENERGY RESEARCH FOREIGN TRAVEL REQUEST COVER SHEET

1. TRIP NO.: ------
2. HQ APPROVAL REQUIRED: Y
3. TRAVELER'S NAME: Stacey, Weston M., Jr.
4. EMPLOYER: Georgia Institute of Technology
5. EMPLOYER DESIGNATION: U
6. DEPARTURE DATE: 09/06/85
7. RETURN DATE: 09/22/85
8. NO. DAYS PERSONAL: 04
9. FISCAL YEAR: 85
10. DESTINATION(S): AU -- -- -- --
11. TRIP CATEGORY: IC __/__/__/__ __/__/__/__ __/__/__/__ __/__/__/__ __/__/__/__
12. CONFERENCE NO.: ----
13. B&R CLASSIFICATION: -- ----
14. ESTIMATED COST TO DOE: $---,---
15. ESTIMATED COST TO OTHERS: $ 3,520.00
16. JUSTIFICATION/PURPOSE: Attend INTOR Workshop session at IAEA in Vienna as US Steering Committee member.

London 9/14-22/85 personal days.

Seminar and Technical Talk (England, Culham Laboratory)

COVER SHEET

FOR TRIP REPORTS SUBMITTED TO THE

OFFICE OF ENERGY RESEARCH

Destination(s) and Dates for
Which Trip Report Being Submitted: Vienna, Austria - April 15-May 3, 1985

Name of Traveler: Weston M. Stacey, Jr.

Joint Trip Report [ ] Yes

[x] No

If so, Name of Other Traveler(s): --

--

Georgia Institute of Technology
A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
SCHOOL OF MECHANICAL ENGINEERING

Please reply to:
NUCLEAR ENGINEERING AND
HEALTH PHYSICS PROGRAM
CHERRY EMERSON BUILDING
GEORGIA INST. OF TECH.
ATLANTA, GEORGIA 30332 U.S.A.

October 29, 1985

Report on Foreign Travel

Grant No. DE-FG05-84ER-52107

Traveler - Weston M. Stacey, Jr.

Project No. E-25-621

Visit to Vienna, Austria - (4/15-5/3/85)

The traveler attended the INTOR Workshop as the US Steering Committee Member. The primary purposes of this Workshop session were to take final decisions on the INTOR design modifications and to write the report of the Phase 2A Part 2 Workshop. These purposes were successfully accomplished. A secondary purpose was to develop a revised set of recommendations for the work of the next phase, taking into account IFRC comments at their September meeting in London. A set of recommendations was developed and presented to the IFRC, who endorsed them.

WMS/sgj

Telephone: 404-894-3720 Telex: 542507 GTRIOCAATL Fax: 404-894-3120 (Verify: 404-894-4850)

ENERGY RESEARCH FOREIGN TRAVEL REQUEST COVER SHEET

1. TRIP NO.: ------
2. HQ APPROVAL REQUIRED: Y
3. TRAVELER'S NAME: Stacey, Weston M., Jr.
4. EMPLOYER: Georgia Institute of Technology
5. EMPLOYER DESIGNATION: U
6. DEPARTURE DATE: 04/12/85
7. RETURN DATE: 05/04/85
8. NO. DAYS PERSONAL: --
9. FISCAL YEAR: 85
10. DESTINATION(S): AU
11. TRIP CATEGORY: IC/__/__/__
12. CONFERENCE NO.: ----
13. B&R CLASSIFICATION: -- ----
14. ESTIMATED COST TO DOE: $---
15. ESTIMATED COST TO OTHERS: $4,520.00
16. JUSTIFICATION/PURPOSE: Attend session XI of INTOR Workshop as US Steering Committee Member, in Vienna, Austria

---

79)
E-N-1500.3

REQUEST FOR APPROVAL OF OFFICIAL FOREIGN TRAVEL

vious Editions are Obsolete)

PART A—To be completed by traveler's administrative officer

dget and Reporting Classification to be charged: E-25-621
ee Appendix 1101, Part II)

he category of a trip, either I, II, or III, is determined by the finitions set forth in Appendix 1501, Part VIII)

TRIP CATEGORY:

PART B—To be completed by traveler

NAME OF TRAVELER Weston M. Stacey, Jr. CITIZENSHIP USA	c. DATE AND PLACE OF BIRTH July 23, 1937; Birmingham, AL USA d. PASSPORT NUMBER (If available)
HOME ADDRESS X-4 Royal Coach Apts. Atlanta, GA 30332 USA	b. BUSINESS ADDRESS Georgia Institute of Technology Atlanta, GA 30332
EMPLOYER Georgia Institute of Technology Atlanta, GA. 30332 USA	c. CONTRACT NUMBER
ORGANIZATIONAL UNIT School of ME/NE & HP	d. POSITION TITLE (Including profession) Callaway Professor Nuclear Engineering

PURPOSE OF TRAVEL—Include all pertinent background information leading to travel and attach copies of invitations and correspondence regarding travel to present papers, give speeches, or to attend conferences or symposia. Also identify by name and organization other DOE and contractor personnel who, to the traveler's knowledge, are going to the same destination at the same time as the traveler. In addition, specify nature and classification of information to be disclosed including titles of papers to be presented; nature of information to be obtained at each of the places to be visited and conferences to be attended and its relation to traveler's work. Travelers are responsible for obtaining clearances for papers or speeches when necessary. If more space is required, attach a separate sheet. NOTE: IF THIS INFORMATION IS CLASSIFIED BE SURE TO CLASSIFY THIS FORM APPROPRIATELY.

Attent session XI of INTOR Workshop as U.S. Steering Committee Member.

vacy Act Information Statement. Collection of the information is authorized by the Department of Energy Organization Act of 1977, P.L. 95-91. closure of the personal information requested is mandatory to support authorization for official travel to foreign countries, and to obtain a sport

e information furnished will be used by DOE to authorize travel and payment of travel expenses; by the Department of State to issue a passport; l by the General Accounting Office to audit and verify the accuracy and legality of disbursements.

PART E—To be completed at DOE Field Organization and/or Laboratory

11. APPROVAL AND/OR REMARKS BY HEAD OF DOE FIELD ORGANIZATION OR DIRECTOR OF LABORATORY:

(Signature) (Date)

PART F—To be completed at Headquarters

12. CONCURRENCE AND/OR REMARKS OFFICE OF SAFEGUARDS AND SECURITY
(Note compliance with Chapter 2502)

(Signature and Title) (Date)

13. CONCURRENCE AND/OR REMARKS OFFICE OF INTERNATIONAL SECURITY AFFAIRS

(Signature and Title) (Date)

14. APPROVAL OR CONCURRENCE/REMARKS BY ASSISTANT SECRETARY FOR INTERNATIONAL AFFAIRS

(Signature and Title) (Date)

15. CONCURRENCE AND/OR REMARKS—OTHERS

(Signature and Title) (Date)

16. CONCURRENCE/REMARKS OF DIRECTOR OF DIVISION OR OFFICE

(Signature) (Date)

17. APPROVAL/REMARKS BY ASSISTANT SECRETARY/DIRECTOR

(Signature) (Date)

Georgia Institute of Technology
A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA
SCHOOL OF MECHANICAL ENGINEERING
ATLANTA, GEORGIA 30332

April 14, 1986

Report on Foreign Travel — Grant No. DE-FG05-84ER-52107

Traveler - Weston M. Stacey, Jr. — Project No. E-25-621

Visit to Vienna, Austria - (3/10-21/86)

Dr. Stacey attended session XIII of the INTOR Workshop at IAEA headquarters in Vienna. A meeting notification (which includes the name and affiliation of the participants), an agenda and a description of the organizational structure of the Workshop session are attached.

The principal tasks accomplished at this session were: 1) review of ongoing work in six critical issue areas and planned detailed homework tasks for the next session; 2) reviewed results of the IAEA meeting on Tokamak Concept Innovations and planned detailed homework tasks for the next session; and 3) the INTOR work plan for the remainder of Phase 2A Part 3 (through CY 1987) was developed.

The tasks of the INTOR Workshop for 1985-87 are:

1. Evaluate innovations which could lead to a significantly improved commercial tokamak, with emphasis upon those that could be implemented on the INTOR time-scale (1990-95).

2. Study critical technical issues that affect the feasibility of an INTOR-like device.

3. Review the objectives of INTOR (or any tokamak ETP).

4. Review the scientific and technological data base for INTOR (or any tokamak ETR).

5. Improve the INTOR conceptual design to incorporate the results of the above work, with the objective of developing an international consensus on the features of a tokamak ETR.

The work plan for the INTOR Workshop is indicated in an attachment. This plan calls for concentration upon critical issues studies and evaluation of innovations which could impact the INTOR concept during this year, and concentration upon incorporating the results of this work into an improved conceptual design during CY 1987.

A number of INTOR -- related IAEA Specialists' Meetings are being held during 1985-87, as indicated in an attachment, to provide additional expert advice to the INTOR Workshop.

The dates for the INTOR Workshop sessions and for the Specialists' Meetings are given in an attachment.

Division: RIRL

1986-03-07
F1-TC-433.34

NOTIFICATION OF A MEETING HELD AT HEADQUARTERS

Title of Meeting: XIIIth INTOR Workshop

Dates Inclusive: 10 - 21 March 1986
Place: B-07, B-08, B-09

Opening Meeting: 9.30 a.m.
Responsible Officers: M. Leiser
Ext 1756, A2375
A. Shurygin
Ext 1702, A2305
Secretary: N. Pawel A2374, Ext 1755

PARTICIPATING AND DESIGNATING MEMBER STATES AND ORGANIZATIONS	ADDRESSES ABROAD	IN VIENNA	FOR THE PERIOD
EURATOM			
G. Grieger	Max Planck Institut für Plasmaphysik D-8046 Garching FRG	Rothensteiner Apartments Neustiftgasse 66 1070 Vienna	10-21 March 1986
F. Engelmann	NET Team Max Planck Institut für Plasmaphysik D-8046 Garching FRG	"	10-21 March 1986
F. Farfaletti-Casali	Joint Research Centre Ispra Establishment I-21010 Ispra (Varese) Italy	"	10-21 March 1986
M. Harrison	Culham Laboratory Abingdon, Oxfordshire OX14 3DB UK	"	10-21 March 1986
A. Knobloch	Max Planck Institut für Plasmaphysik D-8046 Garching FRG	"	10-21 March 1986

H. Iida	Naka Fusion Research Establishment JAERI Naka-machi, Naka-gun Ibaraki-ken 311-02	Hotel Royal 1010 Singerstr. 3	10-21 March 198
M. Kasai	Mitsubishi Fusion Centre 2-3, Marunouchi, 2-chome Chiyoda-ku, Tokyo 100	"	10-21 March 198
T. Kobayashi	Energy Research Laboratory Hitachi, Ltd. 1168 Moriyama-cho, Hitachi-shi Ibaraki-ken 311	"	10-21 March 198
T. Okazaki	Energy Research Laboratory Hitachi, Ltd. 1168 Moriyama-cho, Hitachi-shi Ibaraki-ken 311	"	10-21 March 198
T. Tsunematsu	Naka Fusion Research Establishment JAERI Naka-machi, Naka-gun Ibaraki-ken 311-02	"	10-21 March 198
USSR			
V.I. Igitkhanov	I.V. Kurchatov Institute of Atomic Energy 46 Ulitsa Kurchatova D-182 Moscow		10-21 March 198
B. Kolbasov	I.V. Kurchatov Institute of Atomic Energy 46 Ulitsa Kurchatova D-182 Moscow		10-21 March 198
A. Kostenko	D.V. Efremov Scientific Institute P.O. Box 42 Leningrad		10-21 March 198
R. Litunovski	D.V. Efremov Scientific Institute P.O. Box 42 Leningrad		10-21 March 198

T. E. Shannon	Fusion Engineering Design Center Oak Ridge National Laboratory P.O. Box Y Oak Ridge, TN 37830	10-21 March 1986
D. Smith	Argonne National Laboratory 9600 South Cass Avenue Argonne, Illinois 60439	10-21 March 1986
P. Spampinato	Fusion Engineering Design Center Oak Ridge National Laboratory P.O. Box Y Oak Ridge, TN 37830	10-21 March 1986

10 March 1986

Tentative Agenda

Session XIII
INTOR Workshop Phase IIA Part 3
March 10-21 1986

Monday, March 10

09:00 -09:30	Steering Committee
09:30 - 10:30	Report of Innovation Meeting
11:00 - 12:00	Plenary Session (National Summary)
14:00 - 14:30	Exchange of National Reports
14:30 - 17:00	Reading Reports
16:30	Coordinating Committee

Tuesday, March 11

09:00 - 12:00	Group Discussion on Innovations
14:00 - 17:00	Group Discussion

Wednesday, March 12

09:00 - 10:30	Plenary Session (Innovations)
10:30 - 12:00	Group Discussion
14:00 - 17:00	Group Discussion
17:00	Coordinating Committee

Thursday, March 13

AM & PM	Group Discussion

Friday, March 14

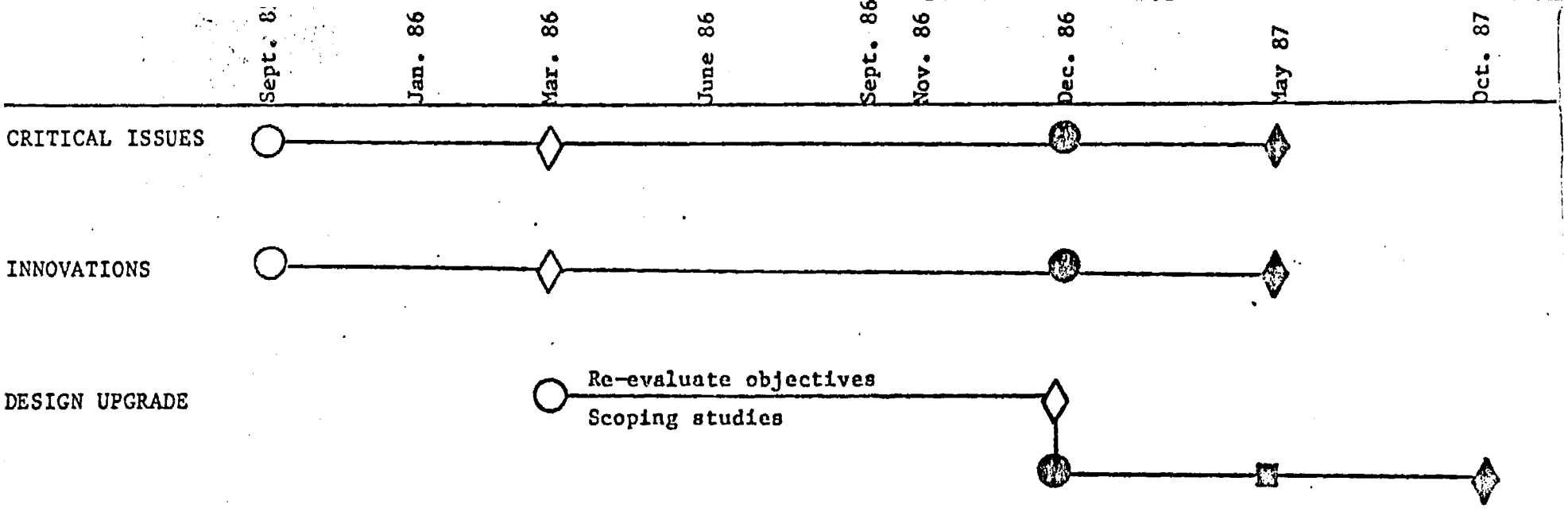
09:00 - 12:00	Group Discussion
14:00 - 17:00	Plenary Session
17:00 - 17:30	Coordinating Committee

PHASE 2A, PART 2, SESSION XIII - INTOR WORKSHOP ORGANIZATION

(March 10 - 21, 1986)

March 10, 1

	Chairman	EC	Japan	USA	USSR
A. Impurity Control	Post	Harrison Schiller	Okazaki	Post	Mazul Igitkhanov Shatalov
B. Beta Enhancement and Confinement	Engelmann	Engelmann Knobloch	Tsunematsu	Peng	Igitkhanov Litunovskij
C. RF Heating and Current Drive	Fujisawa	Engelmann	Fujisawa	Ehst	Vdovin
D. Electromagnetics	Salpietro	Salpietro	Kasai	Henning Shannon	Litunovskij Kostenko
E. Configuration and Maintenance	Iida	Farfaletti-Casali Leger	Iida Honda	Spampinato	Kostenko Kolbasov Yakunin
F. Blanket and First Wall	Shatalov	Vieider Leger	Kobayashi	Smith	Shatalov Kolbasov Mazul Yakunin
G. Physics	Peng	Engelmann Harrison	Fujisawa Tsunematsu Okazaki	Peng Post Ehst	Igitkhanov Vdovin
H. Engineering	Shannon	Farfaletti-Casali Salpietro Knobloch	Iida Kobayashi Honda Kasai	Shannon Henning Spampinato Smith	Kostenko Litunovskij Shatalov Kolbasov Mazul Yakunin
Steering Committee	Mori	Grieger	Mori	Stacey	Kolbasov



INTOR WORK PLAN 1985-87

LEGEND

- Initial planning
- Review work
  Detailed planning
- Preliminary conclusions
  or design
- Intermediate design
- Final Conclusions or design

IAEA Specialists' Mtgs.

[1]	Imp. Cont.	- Vienna
[2]	Innovations	- Vienna
[3]	DEMO Requirements	- Yalta 6/3-4/86
[4]	Current Drive	- Munich 9/15-17/86
[5]	Confinement	- Kyoto 11/21-22/86

INTOR WORKSHOP SESSIONS PHASE 2A, PART 3

XII	September 9-13, 1985
XIII	March 10-21, 1986
XIV	December 1-12, 1986
XV	May 5-15, 1987
XVI	October 19-30, 1987

INTOR RELATED IAEA SPECIALISTS' MEETINGS

Impurity Control	September 16-18, 1985
Tokamak Innovations	January 13-17, 1986
DEMO Requirements	June 3-4, 1986
Current Drive	September 15-17, 1986
Confinement	November 21-22, 1986

Georgia Institute of Technology

A UNIT OF THE UNIVERSITY SYSTEM OF GEORGIA

SCHOOL OF MECHANICAL ENGINEERING

Please reply to:

NUCLEAR ENGINEERING AND
HEALTH PHYSICS PROGRAM
CHERRY EMERSON BUILDING
GEORGIA INST. OF TECH.
ATLANTA, GEORGIA 30332 U.S.A.

February 10, 1986

Department of Energy
Oak Ridge Operations
P.O. Box E
Oak Ridge, Tennessee 37831

Dear Sir:

Subject: Foreign Travel

Approval is requested for Dr. Weston M. Stacey, Jr. (Georgia Institute of Technology, Atlanta, GA 30332) to travel to Vienna, Austria to participate in the INTOR Workshop session to be held March 10-21, 1986. This session is an integral part of the work performed under financial assistance agreement No. DE-FG-5-84ER52107, and funds are available. The estimated cost of the trip to DOE is $4,511.00 ($2,311.00 travel plus $1500 per diem). Travel will commence March 5 and end April 1; the periods March 6-7 and March 22 - March 31 are personal time. Dr. Stacey will hold discussions with other INTOR Workshop participants from the EC, Japan and USSR.

WMS/sgj
Attachments

Weston M. Stacey, Jr.

Approved:
J.A. Brighton
Director ME

Telephone: 404-894-3720 Telex: 542507 GTRIOCAATL Fax: 404-894-3120 (Verify: 404-894-4950)

April 11, 1986

Progress Report Period 1983-1985

DOE Grant Number DE-FG05-84ER52107

Title: Perform Work in Support of the International Tokamak Reactor Workshop

Contractor: Georgia Institute of Technology

Principal Investigator: W.M. Stacey, Jr.

The PI organize and technically directed the US work in support of the INTOR Workshop and participated in that Workshop as the US Steering Committee member.

The principal tasks of Phase Two A, Part 2 of the INTOR Workshop were: (1) work upon certain critical technical issues essential to feasibility and improvement of the INTOR concept; (2) reassessment of the scientific and technical data base supporting the INTOR concept, and (3) revision of the INTOR design. Critical issues studies were carried out in impurity control, RF heating and current drive, transient electromagnetics, maintainability, and technical benefit. The physics, engineering and nuclear data bases supporting the INTOR design concept were re-evaluated, and the design concept itself was, accordingly, slightly modified.

The following publications resulted from this work:

1. INTOR Group, Nuclear Fusion, 25, 1791 (1985).
2. US INTOR Group, INTOR Phase 2A, Part 3, Georgia Institute Technology report (1985).
3. INTOR Group, INTOR, Phase 2A, Part 3, IAEA Report STI/PUB/714, Vienna (1986).

DOE F 1332.16 (10-84)
(Formerly RA-427)

OMB Approval
No. 1910-1400

U. S. DEPARTMENT OF ENERGY

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See Instructions on Reverse Side

1. DOE Report No.	3. Title
2. DOE Contract No. DE-FG05-74ER52107	International Tokamak Reactor Workshop

4. Type of Document ("x" one)
- ☐ a. Scientific and technical report
- ☐ b. Conference paper:
  - Title of conference _____
  - Date of conference _____
  - Exact location of conference _____
  - Sponsoring organization _____
- ☒ c. Other (Specify) Progress Report

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8. Submitted by — Name and Position (Please print or type)
W.M. Stacey, Jr. Callaway Professor of Nuclear Engineering

Organization
Fusion Research Center, Georgia Tech, Atlanta, GA 30332

Signature	Phone	Date
	404/894-3714	May 22, 1986

FOR DOE OR OTHER AUTHORIZED USE ONLY

9. Patent Clearance ("x" one)
- ☐ a. DOE patent clearance has been granted by responsible DOE patent group.
- ☐ b. Report has been sent to responsible DOE patent group for clearance.

May 22, 1986

Progress Report Period 1983-1985

DOE Grant Number DE-FG05-84ER52107

Title:	Perform Work in Support of the International Tokamak Reactor Workshop
Contractor:	Georgia Institute of Technology
Principal Investigator:	W.M. Stacey, Jr.

The PI organize and technically directed the US work in support of the INTOR Workshop and participated in that Workshop as the US Steering Committee member.

The principal tasks of Phase Two A, Part 2 of the INTOR Workshop were: (1) work upon certain critical technical issues essential to feasibility and improvement of the INTOR concept; (2) reassessment of the scientific and technical data base supporting the INTOR concept, and (3) revision of the INTOR design. Critical issues studies were carried out in impurity control, RF heating and current drive, transient electromagnetics, maintainability, and technical benefit. The physics, engineering and nuclear data bases supporting the INTOR design concept were re-evaluated, and the design concept itself was, accordingly, slightly modified.

The following publications resulted from this work:

1. INTOR Group, Nuclear Fusion, 25, 1791 (1985).
2. US INTOR Group, INTOR Phase 2A, Part 3, Georgia Institute Technology report (1985).
3. INTOR Group, INTOR, Phase 2A, Part 3, IAEA Report STI/PUB/714, Vienna (1986).

DOE Grant DE-FG05-84ER52107

WORK ACCOMPLISHED DURING GRANT PERIOD (1/1/86-12/31/87)

The US work in support of Phase 2A Part 3 of the INTOR Workshop was organized and technically directed. The US work in support of several INTOR-related IAEA Specialist's Meetings was organized. The PI represented the US as Steering committee member at INTOR Workshop sessions and in various capacities at the above-mentioned Specialist's Meetings. The PI participated in a number of DOE activities related to ETR in which his knowledge of INTOR was relevant, in particular ETOC. The IAEA INTOR Phase 2A Part 2 report and other reports generated in the course of the work were distributed within the US fusion community. US and IAEA INTOR reports on the Phase 2A Part 3 work were prepared under the supervision of the PI.

Publications:

INTOR Group, INTOR Phase Two A Part II, IAEA report STI/PUB/714, Vienna (1986).

Tokamak Concept Innovations, IAEA-TECDOC-373, Vienna, (1986).

W.M. Stacey, "International Tokamak Reactor", Proc. 11th Int. Conf. Plasma Physics and Controlled Nuclear Fusion Research (Kyoto), IAEA, Vienna (1987).

W.M. Stacey, et al., "The US INTOR Activity", Fusion Techn., 11, 317 (1987).

Progress in Nuclear Energy, Vol. 22, No. 2, pp. 119–172, 1988
Printed in Great Britain.

0149-1970/88 $0.00 + .50

THE INTOR WORKSHOP: A UNIQUE INTERNATIONAL COLLABORATION IN FUSION

WESTON M. STACEY

Fusion Research Center and Nuclear Engineering Program,
Georgia Institute of Technology,
Atlanta, Georgia 30332, U.S.A.

ABSTRACT

The history and technical accomplishments of the International Tokamak Reactor Workshop during its nine years of existence are reviewed.

1. INTRODUCTION

In 1977, the USSR proposed to the Director General of the International Atomic Energy Agency (IAEA) that the Agency sponsor a collaborative activity among the countries with major fusion programs to design, construct and operate the next (after the then upcoming generation of large tokamaks) major experiment in the world fusion program as an international project. This proposal was referred to the International Fusion Research Council (IFRC), an advisory body to the Director General of the IAEA consisting principally of persons with responsibility for the fusion programs in their respective governments. The IFRC recommended that a workshop be established under IAEA auspices and that the activity of this workshop be carried out in phases: Phase Zero -- determination of the scientific and technical feasibility of such a project in the near term; Phase One -- definition of the experiment through a conceptual design; and Phase Two -- design of the experiment. At the end of each phase, the participating governments would review, through the IFRC, the progress of the activity and decide whether to proceed to the next phase and upon the objectives of the next phase. The IFRC drew up terms of reference to guide the workshop.

The International Tokamak Reactor (INTOR) Workshop was established by the IAEA as a collaborative effort among the European Community, Japan, the USA and the USSR. The broad objective of the INTOR Workshop was defined by the IFRC: "To draw on the capability in all countries to prepare a report to be submitted to the IFRC describing the technical objectives and nature of the next large fusion device of the tokamak type that could be constructed internationally." The more specific objectives were:

(1) to identify the objectives and characteristics of the next major experiment (beyond the present generation of large tokamaks) in the world tokamak program;
(2) to assess the technical data base that will exist to support the construction of such a device for operation in the 1990s;
(3) to define such an experiment through the development of a conceptual design;
(4) to study critical technical issues that affect the feasibility or cost of the INTOR concept;
(5) to define R and D that is required to support the INTOR concept;
(6) to carry out a detailed design of the experiment; and, finally,
(7) to construct and operate the device on an international basis.

Each participating government designated a Steering Committee Member. The Steering Committee, acting jointly, was responsible for the organization of the INTOR Workshop, and each Steering Committee Member was responsible for the organization of the scientific and technical work in his respective country*. In Japan and the USSR, where the tokamak programs are

* For the sake of stylistic simplicity, the European Community will be referred to in this paper as a "Country".

centrally organized and located, in the Japan Atomic Energy Research Institute and the Kurchatov Atomic Energy Institute, respectively, the leading people with responsibility for the tokamak program in those institutes, Sigeru Mori and Boris Kadomtsev, respectively, were designated as Steering Committee members and the bulk of the people who contributed to the INTOR work were from those institutes. In the European Community and the USA, where the tokamak programs are much less centrally directed and much more dispense geographically, the INTOR work was organized on a multi-institutional basis. Gunter Grieger of the Max Planck Institute for Plasma Physics in West Germany and myself were designated as Steering Committee Members for Europe and the USA, respectively.

The INTOR Steering Committee first met at IAEA headquarters in Vienna on November 20-23, 1978 to organize the INTOR Workshop. A mode of operation was defined in which a limited number of participants met periodically in workshop sessions in Vienna to define the home tasks, to review the work performed at home, to take decisions and to prepare reports. The bulk of the work in support of the INTOR Workshop was then performed between workshop sessions by experts working in their home institutions under the direction of these Participants. The sessions of the INTOR workshop are given in Table 1.1. The number of Participants from each of the four Parties has varied from 4 in the initial phase to 8-9, as shown in Table 1.2. The home country effort involved hundreds of the leading fusion scientists and engineers in each of the four countries, on a part-time basis.

The Zero Phase of the INTOR Workshop, which was conducted during 1979, addressed the first two objectives cited above; to define the objectives and physical characteristics of the next major experiment (after TFTR, JET, JT-60, T-15) in the world-wide tokamak program; and to assess the technical feasibility of constructing this experiment to operate in about 1990. Detailed assessments of the plasma physics and technology bases for such an INTOR experiment were developed, and physical characteristics were identified which were consistent with this technical basis and with the general objectives of the INTOR concept as it evolved in this process.

Each partner submitted detailed contributions to the Zero Phase Workshop, which were subsequently published. These contributions underwent extensive discussions at the Workshop sessions and formed the basis for the report of the Zero Phase Workshop [1]. This report, which represents a technical consensus of the world-wide magnetic fusion community, concluded that the operation, by the early 1990s, of an ignited, deuterium-tritium burning tokamak experiment that could serve as an engineering test facility was technically feasible, provided that the supporting research and development activity was expanded immediately, as discussed in the report. This broad international consensus on the readiness of magnetic fusion to take such a major step was in itself an important milestone in the development of fusion.

As a result of this positive conclusion, the INTOR Workshop was extended into Phase One, the Definition Phase, in early 1980, on the basis of the IFRC review and recommendation to the Director General of the IAEA. The objective of the Phase One Workshop was to develop a conceptual design of the INTOR experiment.

The Phase One INTOR conceptual design was carried out by teams working in the home countries. The starting point for the conceptual design effort was the set of reference parameters suggested by the Zero Phase Workshop. The decisions taken at each Workshop session were then incorporated into each partner's design activity, so that the four design contributions progressively converged towards a single design, at an increasingly greater degree of detail, during the course of the conceptual design activity. The national conceptual design contributions to the Phase One INTOR Workshop have been published. These contributions formed the basis for the INTOR conceptual design which was completed in mid-1981 and is documented in Ref. [2].

Concurrent with the latter part of Phase One of the INTOR Workshop, an Administrative Committee, composed of government representatives with fusion and foreign relations responsibilities, was established to address the administrative issues associated with moving the INTOR Workshop into the more intensive design Phase Two. The deliberations of this Administrative Committee were inconclusive. At the end of Phase One, the participating governments were unwilling to move forward to the originally planned Phase Two-Design, because of the implications this would have had with respect to a subsequent construction phase.

During the course of the Phase One, several critical technical issues which affected the feasibility, cost or engineering complexity of the INTOR design were identified. Many of these issues were felt to be amenable to resolution through intensive analysis.

This situation led to the decision by the IFRC to redefine Phase Two as Phase Two A--Critical Issues and Phase Two B--Design. Because of the implications that would be associated with proceeding to a design phase, the IFRC instructed the INTOR Workshop to concentrate on critical issues and not to continue design work during Phase Two A. Subsequently, Phase Two A has been twice extended by the IFRC.

During Phase Two A it became increasingly clear to all parties that INTOR would not proceed to the design and construction stage, barring drastic changes in the international political situation. Independent national design efforts for INTOR-like devices were initiated in the European Community, Japan and the USSR during this period. The USA has maintained an independent effort throughout all Phases of INTOR.

During Phase Two A Part 1 of the INTOR Workshop, which extended from mid-1981 to mid-1983, critical issues were studied in five areas (plasma performance, impurity control and first-wall, tritium and blanket, mechanical configuration, and electromagnetics), the engineering testing requirements were defined, and a cost-risk-benefit study of design options with different fluence objectives was conducted. These critical issues studies improved our understanding of major technical issues which affect the feasibility, cost and engineering design tractability of a next-generation tokamak reactor, advanced our knowledge of how to design such a device and led to improvements in several aspects of the INTOR design concept. Some of these intensive studies were carried to a point where further significant progress must await additional experimental information. Specific R and D recommendations were formulated to this end. In other areas, such as impurity control, a continuation of the intensive study was warranted. In addition, several new areas were identified for which intensive studies held the promise to lead to additional improvements of the INTOR concept. The new information that was developed in these critical issues studies led to certain improvements in the INTOR concept. The work in the Phase Two A Part 1 INTOR Workshop was reported in the national contributions to the Workshop and in the report of the Phase Two A Part 1 Workshop [3].

Phase Two A Part 2 (mid-1983 through mid-1985) was concentrated on four critical issues (impurity control, plasma heating and current drive, transient electromagnetics, maintainability) and on a study of the technical benefit of partitioning INTOR component design and fabrication. The last item consisted of a rather detailed examination of the merits and demerits of various forms of international collaboration in constructing and operating an INTOR-like device. Also, a reassessment of the scientific and technical data base supporting the INTOR concept was undertaken. As a consequence of these studies, some of the major parameters of the INTOR design concept were modified. This was done in order to illustrate in which way the INTOR design concept should be modified to take account of the new results, leaving the self-consistent updating to be done in a later INTOR Phase. The work of Phase Two A Part 2 Workshop is reported in the national contributions and in the report of the Phase Two A Part 2 Workshop [4].

Phase Two A Part 3 (mid-1985 through-1987) started with the intention (i) to address the following critical issues: impurity control, beta and confinement, heating and current drive, electromagnetics, configuration and maintenance, and first wall and blanket; (ii) to reassess the DEMO requirements: (iii) to study potential innovations which were not yet supported by developed physics or technology, but which hold the promise to lead to improvement of the tokamak concept; and (iv) to incorporate the results of all the work done during Parts 1 to 3 of Phase Two A in an updated INTOR conceptual design.

During the course of Phase Two A Part 3, the prospects for international collaboration on the design and construction of an INTOR-like experiment increased dramatically as a result of a summit-level initiative by the USSR and a favorable response by the USA. Subsequent government level discussions involving the European Community and Japan as well as the USA and USSR evolved towards a new international design activity for an International Thermonuclear Experimental Reactor (ITER) with objectives and general characteristics similar to those of INTOR. At this moment, the work orientation of the INTOR Workshop was rediscussed by the IFRC, and it was decided to stick to the work on critical issues, DEMO requirements and innovations, because these items were of immediate relevance to any near-term tokamak design activities, but to replace the intention to introduce the results of the work into an updating of the INTOR conceptual design by an analysis of how those results would affect the INTOR design concept, and rather to use this part of the capacity of the INTOR Workshop for a critical analysis of all existing INTOR-like designs. The results of Phase Two A Part 3 are reported in Ref. [5].

Table 1.1
Sessions of the INTOR Workshop

PHASE ZERO	PHASE ONE
February 5-16, 1979	January 16-18, 1980
June 11 - July 5, 1979	March 24-28, 1980
October 1-23, 1979	June 16-27, 1980
December 10-19, 1979	October 20-31, 1980
	January 19 - February 4, 1981
	March 30 - April 10, 1981
	June 22 - July 3, 1981
PHASE TWO A PART 1	PHASE TWO A PART 2
September 7-11, 1981	January 16-27, 1984
December 7-18, 1981	May 21 - June 1, 1984
March 22 - April 2, 1982	October 15-27, 1984
July 12-23, 1982	April 14 - May 3, 1985
April 11-22, 1983	September 9-14, 1985
August 1-5, 1983	
PHASE TWO A PART 3	
March 10-21, 1986	
December 1-12, 1986	
July 13-24, 1987	
November 9-20, 1987	

2. INTOR ROLE AND OBJECTIVES

2.1 INTOR in the Fusion Program

Apart from the more general INTOR role to be the maximum reasonable step beyond the present generation of large tokamaks (TFTR, JET, JT-60, T-15), its programmatic objectives followed from a deeper consideration of its potential role in the world fusion program. The target of all fusion programs is the successful construction and operation of a fusion reactor DEMO. A definition of the DEMO objectives and a discussion of the broad, general prerequisites for the design and construction of DEMOs have led to the working hypothesis that one single step between the present generation of large tokamaks and the DEMO might be sufficient, together with the other parts of the fusion program, to develop all the information necessary for the construction of DEMOs. The characteristics of a DEMO were developed early in the INTOR Workshop, for the purpose of guiding the determination of the INTOR objectives, and have been essentially confirmed by an IAEA consultants' meeting on DEMO requirements [6]. The DEMOs are generally understood to have the following objectives:

(a) Production of several hundred megawatts electricity and achievement of net electrical power production
(b) Production of tritium in the blanket, with a net breeding ratio greater than unity
(c) Demonstration of the development and integration of full-scale components which can be extrapolated to a commercial reactor
(d) Demonstration of component and system reliability, availability and lifetime at a level that would be acceptable for a commercial reactor
(e) Demonstration of safe and environmentally acceptable fusion reactor operation that would satisfy the requirements for a commercial reactor
(f) Demonstration of commercial feasibility (although the DEMO would not need to be itself economically competitive).

The role of INTOR in the fusion program can be defined upon identifying the physics and technology prerequisites for the design and construction of the DEMOs. Then, those prerequisites which can best be satisfied by INTOR and those for which complementary physics experiments and technology test facilities are needed can be distinguished.

Table 1.2
Principal INTOR Participants

EC	Japan	USA	USSR
Phase Zero (1979)			
G. Grieger	S. Mori	W.M. Stacey, Jr.	B.B. Kadomtsev
F. Engelmann	T. Hiraoka	J.R. Gilleland	G.E. Churakov
R. Hancox	K. Sako	G.L. Kulcinski	B.N. Kolbasov
D. Leger	T. Tazima	R.H. Rutherford	V.I. Pistunovich
P. Reynolds			G.F. Shatalov
Phase One (1980-81)			
G. Grieger	S. Mori	W.M. Stacey, Jr.	B.B. Kadomtsev
G. Casini	N. Fujisawa	M.A. Abdou	C.F. Churakov
F. Engelmann	T. Hiraoka	J.A. Schmidt	B.N. Kolbasov
F. Farfaletti-Casali	K. Sako	T.E. Shannon	V.I. Pistunovich
R. Hancox	K. Tomabechi		D.V. Serebrennikov
A. Knobloch			G.E. Shatalov
D. Leger			
P. Reynolds			
P. Schiller			
Part Two A Part 1 (1981-83)			
G. Grieger	S. Mori	W.M. Stacey, Jr.	B.B. Kadomtsev
G. Casini	N. Fujisawa	M.A. Abdou	G.F. Churakov
F. Engelmann	T. Hiraoka	J.M. Rawls	B.N. Kolbasov
F. Farfaletti-Casali	K. Miyamoto	J.A. Schmidt	V.I. Pistunovich
M. Harrison	S. Nishio	T.E. Shannon	D.V. Serebrennikov
A. Knobloch	Y. Sawada	R.J. Thome	G.E. Shatalov
D. Leger	K. Tomabechi		
P. Reynolds			
P. Schiller			
Part Two A Part 2 (1983-85)			
G. Grieger	S. Mori	W.M. Stacey, Jr.	B.B. Kadomtsev
H. Chazalon	N. Fujisawa	C.C. Baker	B.N. Kolbasov
F. Engelmann	T. Honda	P.L. Colestock	A.I. Kostenko
F. Farfaletti-Casali	H. Iida	C.A. Flanagan	A.S. Kukushkin
M. Harrison	M. Kasai	R.F. Mattas	R.N. Litunovskij
A. Knoblock	M. Seki	M.Y. Peng	V.I. Pistunovich
D. Leger	M. Sugihara	D.E. Post	D.V. Serebrennikov
E. Salpietro	K. Tomabechi	P.T. Spampinato	G.E. Shatalov
P. Schiller		R.J. Thome	
Phase Two A Part 3 (1985-87)			
G. Grieger	S. Mori	W.M. Stacey, Jr.	B.B. Kadomtsev
E. Cocorese	N. Fujisawa	D. Ehst	A. Epinatiev
F. Engelmann	T. Honda	C.A. Flanagan	B.N. Kolvasov
F. Farfalletti-Casali	H. Iida	Y.M. Peng	A.I. Kostenko
M. Harrison	B. Ikeda	N. Pomphrey	A.S. Kukushkin
A. Knobloch	T. Kobayashi	D.E. Post	R.N. Litunovski
D. Leger	T. Mizoguchi	D.L. Smith	I.V. Mazul
E. Salpietro	T. Okazaki	P.T. Spampinato	V.I. Pistunovich
P. Schiller	T. Tsunematsu		G.E. Shatalov
W. Spears			V. Vdovin
G. Vieider			S. Yakunin
J. Wegrowe			

The broad, general prerequisites for the design and construction of DEMOs are:

(a) Development of an adequate plasma physics and engineering data base for prediction of the performance of the DEMOs
(b) Demonstration of the plasma physics performance required for the DEMOs
(c) Development of fusion reactor components
(d) Testing of component integration into an overall fusion reactor system
(e) Testing of fusion reactor maintainability
(f) Testing of component and overall reactor system reliability, at least to some significant fraction of the availability and design lifetime of the DEMOs
(g) Testing of electricity and tritium production by fusion
(h) Testing of the safety and environmental characteristics of a fusion reactor.

The INTOR concept is predicated upon the existence of a vigorous worldwide fusion R and D program that is constrained only by technical considerations.

An extensive plasma physics experimental and theoretical programme will be needed to support the design and construction of INTOR and to supplement INTOR in providing the physics basis for the design and construction of DEMOs. In this context, INTOR is viewed as the maximum reasonable physics step beyond the present generation of large tokamaks towards DEMO and is intended to demonstrate the achievement of most of the plasma conditions that will be required for tokamak DEMOs. Primary physics objectives of INTOR then are to investigate the operation of an ignited D-T plasma and to achieve long, controlled, reproducible burns with optimized plasma parameters. Achievement of these objectives requires satisfactory impurity control, power and particle balance control, and profile control for parameter optimization. A closely related objective is the achievement of high-duty cycle operation. INTOR may also be used to perform certain plasma physics experiments not directly related to learning how to operate INTOR, but such experiments should be carried out in other plasma physics devices, if possible.

An extensive technology and component development and testing program will be required in the development of fusion power reactors to the demonstration reactor stage. This program will both support INTOR in providing the basis for its design and construction, and supplement INTOR in providing the basis for the design and construction of the DEMOs.

In general, it is anticipated that a thorough screening of candidate materials and component design concepts will be carried out in test facilities and that, before the final design and construction of INTOR, components will be developed and tested under conditions that, at least partially, simulate a fusion reactor environment. INTOR will then serve principally to:

(a) Test the compatibility of components within an integrated reactor system
(b) Test the remote maintainability of a fusion reactor system
(c) Test components and materials in a fusion reactor environment
(d) Test the reliability of components under sustained operation in a fusion reactor environment, i.e. to some significant fraction of the component design lifetime against the limiting phenomenon (e.g. neutron damage, fatigue).
(e) Irradiate materials samples to moderate fluences in a fusion neutron spectrum
(f) Test the production of electricity and tritium in a fusion reactor
(g) Test the safety and environmental acceptability of a fusion reactor.

Much will be learned in carrying out these investigations that will be utilized to improve the design of components and the overall reactor system for the DEMOs. It is then the role of the DEMOs to provide convincing demonstrations with full-size, fully developed components that can readily be extrapolated to commercial reactors.

Within the fusion programs also other magnetic confinement concepts besides the tokamak are being developed. There is a good chance that one or more of these concepts will be developed to the commercial stage, and there is even a possibility that some other concept will supplement the tokamak as the frontrunner before the DEMO stage. Thus, it is important that INTOR serve also to test technology that is required for these other magnetic fusion concepts. Fortunately, the technologies required for the principal magnetic confinement concepts are, to a high degree, common to all systems.

2.2 INTOR Objectives

The programmatic objectives of INTOR follow directly from the foregoing considerations of its role in the fusion program and from an assessment of the technical basis which could exist within the next several years for its design. The technical objectives have been developed to support

the achievement of the programmatic objectives while being consistent with the anticipated technical basis. These objectives will be achieved at different stages of INTOR operation. The programmatic objectives of INTOR are:

(a) INTOR should be the maximum reasonable step beyond the present generation of large tokamaks (TFTR, JET, JT-60, T-15) in the world fusion program
(b) INTOR should demonstrate the plasma performance required for the tokamak DEMOs
(c) INTOR should test the development and integration into a reactor system of those technologies required for the DEMOs
(d) INTOR should serve as a test facility for the blanket, tritium production, materials and plasma engineering technology development
(e) INTOR should test fusion reactor component reliability
(f) INTOR should test the maintainability of a fusion reactor
(g) INTOR should test the factors affecting the reliability, safety and environmental acceptability of a fusion reactor.

The technical objectives of INTOR have been developed to support the achievement of the programmatic objectives, while being consistent with the anticipated technical basis for the design and construction of such an experiment to initially operate in the mid 1990s. These technical objectives are given in Table 2.1.

TABLE 2.1 INTOR TECHNICAL OBJECTIVES [4]

A. Reactor-relevant mode of operation	
1.	Ignited D-T plasma
2.	Controlled burn pulse of > 100 s
3.	Reactor-level particle and heat fluxes ($P_n \geq 1$ MW·m^{-2})
4.	Optimized plasma performance
5.	Duty cycle $\geq$ 70%
6.	Availability 25%
B. Reactor-relevant technologies	
1.	Superconducting toroidal and poloidal coils
2.	Plasma composition control (e.g. divertor)
3.	Plasma power balance control
4.	Plasma heating and fuelling
5.	Blanket heat removal and tritium production
6.	Tritium fuel cycle
7.	Remote maintenance
8.	Vacuum
9.	Fusion power cycle
C. Engineering test facility	
1.	Testing of tritium breeding and extraction
2.	Testing of advanced blanket concepts
3.	Materials testing
4.	Plasma engineering testing
5.	Electricity production ~5-10 MW (e)
6.	Fluence ~3 MW·y·m^{-2}

An important question was the extent to which materials and component testing should be included in the INTOR objectives. It was clear from the beginning that end-of-life testing of DEMO components or materials was far above all possibilities with the technologies available for the basic INTOR machine. Even one-third of the intended DEMO fluence seemed to be out of scope.

This led to the concept to study radiation damage of reactor materials by simulation methods and to use INTOR for calibration of the simulation methods by a 14 MeV neutron environment. Since it was expected in addition that within the available time and budget only stainless steel could be developed to all manufacturing details required for DEMO components, a fluence goal of 3 MWy/m^2 was considered adequate. In the meantime, ferritic steels also are being considered for use in DEMO. In this case, the fluence needed for calibration would be around 6 MWy/m^2, which might be too large to aim at in INTOR. In this case the minimum fluence goal is related to the development and test of blanket modules, which is about 0.8 MWy/m^2. Hence, 3 MWy/m^2 is retained as the goal to allow also for some valuable component testing and for extended blanket module testing.

Self-sufficiency in tritium production was not an original INTOR objective. In this respect, module and sector tests were considered adequate to allow reliable extrapolations. A tritium breeding blanket was introduced in the INTOR basic machine upon request of the IFRC in order to reduce the demands on tritium supply and thus the operating cost. For this blanket the request for full reactor relevance was replaced by the condition not to interfere with the reliable operation of the basic INTOR machine. In order to achieve this objective and to maximize the shielding capability on the inboard of the torus, the breeding blanket was restricted to the outboard and upper sections of the plasma chamber, covering about 60% of the available space. The initial INTOR blanket designs achieved a tritium breeding ratio of 0.6 with this coverage. However, recent designs have been developed which are capable of achieving a tritium breeding ratio in excess of unity with this coverage, so tritium self-sufficiency appears to be obtainable without interfering with the reliable operation of the machine.

3. DESIGN CONCEPT

3.1 Guiding Parameters

The initial INTOR design concept and philosophy were embodied in the set of guiding parameters that were developed in Phase Zero [1] to guide the data base assessment of that Phase and to provide a starting point for the conceptual design activity of Phase One. These guiding parameters were evolved in part on the basis of previous national studies of "next-step" devices and in part on the basis of the ongoing data base assessment.

The concept for a machine that is capable of meeting the technical objectives of INTOR and supported by the data base assessment was developed by the Workshop. The concept recommended as input for the definition phase was characterized by a relatively large (~280 m^3) D-shaped plasma heated to ignition by neutral-beam injection power of about 75 MW for 5 - 6 s up to a mean temperature of about 10 keV. Controlled burn should then be achieved for more than 100 s at a plasma-to-magnetic pressure ratio $\langle\beta\rangle$ = 5%, which was thought at that time to be marginally stable theoretically. A divertor is needed to exhaust helium and control impurity influx in order to achieve such a long burn. The plasma current should exceed 6 MA, which is quite sufficient to confine fast alpha particles. Both the toroidal and external poloidal field coils should be superconducting. INTOR would require about 200 MW (e) power. The neutron power load on the first wall would be about 1.3 MW·m^{-2}, and the total fusion power would be about 620 MW(th). A stainless-steel, water-cooled shield operating with relatively low (100 - 200°C) maximum structural temperature would be employed on the basic machine. The guiding parameters for INTOR are given in Table 3.1 under the column head "1979".

Since a principal function of INTOR is to test the feasibility of reactor components and to serve as a test facility for technology development, simplicity and reliability of the basic machine are of the utmost importance, even if this is in conflict with certain aspects of reactor relevance. The technical feasibility of INTOR is then intimately related not only to the supporting technology development programs but also to the manner in which each technology is incorporated into INTOR.

Certain reactor-relevant technologies must be incorporated as intrinsic -- not easily replaceable -- components (e.g. superconducting magnets, remote handling system) in order to be tested in INTOR. These technologies must be developed to a high level before the construction of INTOR since failure of such intrinsic components would cause INTOR either to fail or to be significantly delayed in achieving any of its objectives. For such intrinsic technologies, INTOR would test an already-developed technology in a fusion reactor environment.

With certain other reactor-relevant technologies (e.g. electricity and tritium-producing blanket), there exists the option of incorporation as intrinsic components or as relatively easily

replaceable test components. Inclusion of a new technology as an intrinsic component at the outset places a greater demand upon the development program supporting that technology and entails an increased risk that failure of that component would cause INTOR to fail or to be significantly delayed in achieving any of its objectives. On the other hand, incorporating a new technology as a test component that could be inserted after INTOR had operated for some time and that could be relatively easily replaced places less of a demand upon the supporting

TABLE 3.1 INTOR PARAMETERS

	1987[5]	1985[4]	1983[3]	1981[2]	1979[1]
GEOMETRY					
Chamber major radius, R (m)	*	4.9	5.2	5.2	5.2
Chamber volume (m^3)		285	320	320	320
Chamber surface area (m^2)		352	380	380	380
PLASMA					
Plasma radius, a (m) *		1.2	1.2	1.2	1.4
Plasma elongation, K		1.6	1.6	1.6	1.6
Plasma aspect ratio, A		4.2	4.4	4.4	4.0
Burn average beta, $\langle\beta\rangle$(%)		4.9	5.6	5.6	5
Poloidal beta, β_I		1.5	2.6	2.6	TBD
Average ion temperature, $\langle T_i \rangle$ (keV)		10	10	10	10
Average ion density, $\langle n_i \rangle (m^{-3})$		1.4×10^{20}	1.4×10^{20}	1.4×10^{20}	1.4×10^{20}
Energy confinement time, τ_E (s)	*	1.4	1.4	1.4	1.5
Plasma current, I_p (MA)	*	8.0	6.4	6.4	6.4
Field on plasma axis, B_T (T)	*	5.5	5.5	5.5	5.5
Safety factor (separatrix), q_I		1.8	2.1	2.1	2.5
Peak thermonuclear power, P_{th} (MW)		585	620	620	620
Neutron wall load, P_n ($MW.m^{-2}$)		1.3	1.3	1.3	1.3
Toroidal field ripple at outboard edge (%)		1.2	0.9	TBD	<0.75
OPERATION					
Burn time, Stage I/Stage II (s)		100/150	100/200	100/200	>100
Duty cycle, Stage I/Stage II (%)		~60/~70	70/80	70/80	>80
Fluence goal ($MW.y.m^{-2}$)		3	4	6	TBD
Number of pulses (lifetime)		4×10^5	7×10^5	7×10^5	$5\text{-}10 \times 10^5$
Maximum availability goal (%)		25	50	50	25-50
HEATING - NEUTRAL BEAM					
Number of injectors (active/spare)		-	-	4/1	TBD
Beam power (MW)		-	-	75	75
Beam energy (keV)		-	-	175	175
Pulse-length capability (s)		-	-	10	10

*Parameter might be expected to change is design was updated.

TABLE 3.1 INTOR PARAMETERS (cont.)

		1987[5]	1985[4]	1983[3]	1982[2]	1979[1]
HEATING: ICRF						
Number of launchers (active/spare)			3/1	3/1	-	-
Power at start-up (MW)			50	50	-	-
Frequency (MHz)			85	85	-	-
Pulse-length capability			CW	CW	-	-
STARTUP ASSIST: ECRH						
Power (MW)			10	10	-	-
Frequency (GHz)			140	TBD	-	-
Pulse length (s)			3	TBD	-	-
CURRENT DRIVE: LHR		*				
Number of launches			1	-	-	-
Power (MW)			20	-	-	-
FUELLING						
Method			pellet injection and gas puffing	pellet injection and gas puffing	pellet injection and gas puffing	pellet injection and gas puffing
IMPURITY CONTROL						
Method			single-null poloidal divertor	single-null poloidal divertor	single-null poloidal divertor	single-null poloida diverto
Plasma current, I_p (MA)		*	5.0	6.4	6.4	
Collector			W bonded to copper	W or Be on SS	W on SS or Cu 80	TBD TBD
Power to divertor (MW)			80	80	80	TBD
FIRST WALL						
Power to first wall (MW)			49	44	44	TBD
Outboard: material		*	H_2O-cooled SS316	D_2O-cooled SS316	D_2O-cooled SS316	H_2O-cooled SS
thickness (mm)			12	12	11.7	TBD
Inboard: material		*	H_2O-cooled SS316	H_2O-cooled SS316	H_2O-cooled SS316	H_2O-cooled SS
thickness (mm)			14	14	13.5	TBD
Lifetime (y)			12	15	15	TBD

HEATING - NEUTRAL BEAM

TABLE 3.1 INTOR PARAMETERS (cont.)	1987[5]	1985[4]	1983[3]	1981[2]	1979[1]
BREEDING BLANKET		H_2O,SS316	D_2O or H_2O, SS316	H_2O,SS316	TBD
Material		Li_2O	Li_2O	Li_2O,Pb,c Pb,CT	TBD
Breeder temperature (°C)		410-800	400-650	400-600	TBD
Thickness (m)		0.5	0.5	0.8	TBD
Location		outboard and top	outboard and top	outboard and top	TBD
Breeding Ratio	*	>0.6	>0.6	0.65	TBD
Tritium extraction		cont. He purge	cont. He purge	cont. He purge	TBD TBD
TRITIUM FUEL SYSTEM					
Tritium flow rate ($g.h^{-1}$)		61	64	64	55
Annual tritium consumption at 25% availability ($kg.y^{-1}$)		7	7	7	TBD
Isotopic enrichment		cryogenic dist.	cryogenic dist.	cryogenic dist.	TBD
TRITIUM INVENTORY					
Breeding blanket (kg)		0.5-1.0	0.5-1.0	0.5-1.0	
Storage (kg)		1.1	2.3	2.3	2.5
First wall/divertor (kg)		0.1-1.0	0.1-1.0	0.2	
Tritium handling systems (kg)		1.4	1.4	0.4	
TORUS VACUUM SYSTEM					
Initial base pressure (torr)		10^{-7}	10^{-7}	10^{-7}	10^{-7}
Pre-shot base pressure (torr)		$3x10^{-5}$	$3x10^{-5}$	$3x10^{-5}$	10^{-6}
Pumps		Compound cryopumps through divertor chamber	Compound cryopumps through divertor chamber	Compound cryopumps through divertor chamber	TBD
Pumping					TBD
Pumping speed ($l.s^{-1}$)		2×10^5	TBD	TBD TBD	
TOROIDAL FIELD COILS					
Number	*	12	12	12 12	
Bore (m)	*	6.3x9.3	6.6x9.3	7.7x10.7	TBD
Conductor		Nb_3Sn	Nb_3Sn, NbTi	Nb_3Sn, NbTi	Nb_3Sn &/or NbTi
Stabilizer		Cu	Cu	Cu	Cu
Maximum field (T)		~11	<12	~11	10.6

TABLE 3.1—INTOR PARAMETERS (cont.)

	1987[5]	1985[4]	1983[3]	1981[2]	1979[1]
POLOIDAL FIELD COILS					
Total flux ($\dot{V}$.s)		112	110	110	110
Breakdown voltage (V)		35 for 0.3s	35 for 0.3	TBD	100
Location	*	external to TF coils	external to TF coils	external to TF coils	external to TF coils
Conductor		NbTi or Nb_3Sn	NbTi	NbTi	NbTi
Maximum allowable field (T)		8	8	8	9
POWER SUPPLIES					
Stationary loads (MW)		200	200	241	TBD
Pulsed energy storage (GJ)		14	14	22.5	TBD
SHIELDING					
Inboard (m)		0.8	1.1	0.8	1.1
Outboard		1.5	1.5	1.65	1.4
Neutral beam drift tube (m)		-	-	1.0	TBD
NBI box (m)		-	-	0.5-0.75	TBD

MECHANICAL CONFIGURATION

1987 *

1985

Twelve or twenty-four blanket sectors assembled with straight-line or translational horizontal motions through windows between TF coils
Semi-permanent inboard, upper and lower shield forming primary vacuum boundary on outer boundary of removable torus sectors
Final closure of primary vacuum boundary on outer boundary of removable torus sectors
Test modules inserted horizontally at mid-plane
All superconducting coils in a common cryostat, except lower ring coil in a separate cryostat

Dedicated sectors:	4 ICRF	2 fuelling
	1 ECRF	2 testing
	1 LHRF	2 I & C

1983

Twelve or twenty-four blanket sectors assembled with straight-line or translational horizontal motion through windows between TF coils
Semi-permanent inboard, upper and lower shield forming primary vacuum boundary or inner surface
Final closure of primary vacuum boundary on outer boundary of removable torus sectors
Test modules inserted horizontally at mid-plane
All superconducting coils in a common cryostat, except lower ring coil in a separate cryostat

Dedicated sectors:	4 ICRF	2 fuelling
	1 ECRF	3 testing
		2 I & C

1981

Twelve blanket sectors assembled with straight-line horizontal motion through windows between TF coils
Semi-permanent inboard, upper and lower shield forming primary vacuum boundary or inner surface
Final closure of primary vacuum boundary on outer boundary of removable torus sectors
Test modules inserted horizontally at mid-plane
All superconducting coils in a common cryostat

Dedicated sectors:	5 NBI	3 testing
	2 fuelling	2 I & C

development program and entails considerably less risk that component failure will significantly delay INTOR in achieving its other objectives. A further advantage of inclusion of a new technology in test components is that INTOR could serve as a test bed for the development of that technology, with successively improved test components being tested in INTOR.

The philosophy of INTOR is to minimize the risk of failure and the demands upon the technology development programs by including new technologies in test components to the maximum extent that is consistent with achieving the programmatic and technical objectives. This philosophy leads to the concept of staged operation, with the possibility of incorporating some of the technologies that have been tested in test components during an early stage, into intrinsic components during the last stage in order to provide a full-scale engineering demonstration.

3.2 Conceptual Design

Based on a realistic assessment [1] of the then anticipated status of plasma physics research and technology development a few years hence, during Phase One a conceptual design was developed for an INTOR device which could fulfil the objectives listed in the previous section. Emphasis was given to developing the design self-consistently and in sufficient detail in certain important areas so that the critical problems could be identified and resolved and so that the consequences of certain major design decisions could be investigated.

The major features for the INTOR conceptual design are specified in Table 3.1 under the column headed "1981". It is interesting to note that it was possible to develop a self-consistent conceptual design with major parameters very close to those specified in Phase Zero as guiding parameters.

An analysis of the magnetics, MHD equilibrium and stability, energy transport, plasma heating and impurity control was made to support the plasma physics parameters specified for INTOR. Based upon the then existing knowledge, it was felt that the INTOR plasma, operating with the indicated parameters, should achieve an ignited burn with an average thermonuclear power output of 620 MW(th). The plasma current, in excess of 6 MA, should adequately confine alpha particles. The value $\langle\beta\rangle = 5.6\%$ was somewhat greater than the then current theoretical limit, but experimental evidence that tokamaks can operate in excess of this theoretical limit supported this choice. A divertor was specified to exhaust helium and to prevent heavy impurities from reaching the plasma in order to achieve a 200 s burn time, which was set at about one-fifth the theoretical magnetic surface diffusion time. Based upon the then current best estimate of plasma energy transport losses, the predicted alpha-heating power exceeded that required for ignition by a factor of about two. The 75 MW of neutral beam heating power allows the plasma to be heated to ignition for energy losses up to about twice as large as the then current best estimate for heat conduction. This provided a margin for coping with radiation losses and existing uncertainties.

A single-null poloidal divertor, with the chamber at the bottom, was chosen for impurity control. Analyses indicated that it should be possible to magnetically form the divertor channels and to control the separatrix motion to within several centimeters with coils external to the toroidal field coils. A relatively short channel length should be adequate because of the high-density mode of divertor operation. Two divertor collector plate designs were developed. In one, tungsten was brazed to a copper heat sink. In the other, tungsten tiles were mechanically attached to a stainless-steel heat sink. The tungsten and copper would need to be replaced every few years; the stainless steel should last the design life-time of INTOR. The analysis in support of the single-null divertor included self-consistent treatments of the magnetics for separatrix control and divertor channel formation, the plasma physics of the divertor channel and scrape-off region, the nuclear design of the divertor collector plate, and the engineering design of a maintainable divertor.

The mechanical configuration design was driven from the outset by the requirement to provide maximum access to facilitate maintenance and assembly/disassembly. A semi-permanent inboard, upper and lower shield formed the primary vacuum boundary. Twelve torus sectors fit within this semipermanent shield. These torus sectors were partially (outboard and upper) tritium-producing blanket and partially (inboard and lower) heat-removal shield. The final closure of the vacuum boundary on the outboard was at the outer boundary of the blanket, inside of the outboard bulk shield. Once the outboard bulk shield is removed and the vacuum boundary is cut, each torus sector could be withdrawn horizontally with straight-line motion through a 'window' between adjacent toroidal field coils. The divertor channel was broken up into twelve modules which are removable with straight-line horizontal motion between the toroidal field coils. The single-null divertor was chosen over the double-null divertor in order to achieve

this more simply maintainable mechanical configuration.

Semi-permanent, superconducting toroidal and poloidal field coils would be enclosed in a common, semi-permanent cryostat, thus completely separating the cold and warm structures. All poloidal field coils would be external to the toroidal field coils and superconducting, except for a set of cryoresistive coils internal to the Ohmic heating solenoid which provide the voltage pulse for plasma breakdown. Both forced-flow and pool-boiling conductor designs were developed for the toroidal and poloidal field coils, and in addition a superfluid pool-boiling conductor design using NbTi was developed for the toroidal-field coils. Each of these conductor concepts was under active development at the time, and a final decision could await results from the development programs.

The rather demanding structural requirements for the toroidal field coils were met by a combination of design strategies. Coil wedging, intercoil support structure and a bucking cylinder would be used to handle in-plane and centering forces. Gussets, intercoil support structure, a ring girder, the bucking cylinder and shear ties would be used to handle out-of-plane forces and the overturning moment. A built-up laminated structure would be used. A major accomplishment of the INTOR design effort was to develop a credible structural design for a high-field, pulsed tokamak.

Extensive analysis supported the design of the first wall, blanket and shield. A water-cooled, stainless-steel first wall with a panel-type construction was specified. This first wall was expected to last the full life-time of the device, provided that the melt layer which was predicted to form on the inboard section during a plasma disruption is stable. A tritium-producing blanket would be installed from the outset of operation in order to reduce the operational cost. A solid breeder (Li_2SiO_3) blanket that covered the outboard and upper surfaces of the plasma chamber was shown to be able to produce more than 60% of the tritium consumed in INTOR. Shielding for the torus and major penetrations was determined on the basis of several one-dimensional and one extensive three-dimensional radiation transport calculations. Adequate shielding would be available for component protection and to allow access 24 hours after shut-down.

The availability goal for INTOR was specified as 50% during the last stage of operation. Reliability analyses based upon component reliability estimates provided by the component developers indicated that achievement of this goal would require increased emphasis on component reliability in the component development programs. Extrapolation of the then present reliability data led to availability estimates of ~ 30 - 40%, depending upon the degree of redundancy.

It should be noted that the purpose of the work of Phase One was to define the INTOR concept. Specifications for the different systems were naturally evolved in a parallel and iterative fashion, with the result that not all systems were specified to the same level of consistency and detail. The Phase One work sufficed to define the concept and further work to achieve consistency on a detailed level was left as a task for a later phase.

In retrospect, a few factors can be identified which had a major influence upon the INTOR conceptual design of Phase One. It is useful to review those factors which most contributed to imposing demanding engineering design requirements. The emphasis on maintainability was the major factor in determining the mechanical configuration. The requirement of a relatively simple torus and divertor assembly/disassembly procedure led to somewhat larger toroidal field coils than otherwise would be necessary, imposed certain constraints upon the toroidal field coil structural support system and led to the choice of a single-null (rather than a double-null) poloidal divertor (which imposed additional requirements upon the poloidal field coil system). The requirement of maintainability led to a choice of an all-external poloidal field coil system. The all-external PF coils and the large TF coils necessitated by the accessibility requirement, together with the high toroidal field specified, combined to produce large overturning moments on the toroidal field coils and thus led to additional requirements for the structural support system. The requirement of personnel access for ance at the outer boundary of the reactor led to considerably more outboard shi ld be necessary for component protection. The objective of achieving an ac action of the design life-time neutron fluence for first-wall and blanket n a DEMO had a significant influence upon the specification of the neutron wall availability goal, upon the inboard shield thickness, and upon the design of the all and magnet structural system with respect to fatigue and crack growth limits.

3.3 Phase Two A Part 1 Design Concept Update

The INTOR design concept that was developed in Phase One was updated at the end of Phase Two A Part 1 based upon the results of the critical issues work performed during that Phase. This update was based upon detailed analysis of the system involved, but a self-consistent design iteration was not performed, so that the resulting design concept was not optimized nor necessarily self-consistent on the detailed level.

The updated design concept that was developed in Phase Two A Part 1 is shown in Table 3.1 under the column head "1983". The major changes with respect to Phase One were the replacement of neutral beam heating with ICRF heating and the use of ECRH for startup assist to reduce the required loop voltage for current initiation. The power supply requirements and the toroidal field coil size were both reduced somewhat.

3.4 Phase Two A Part 2 Design Concept Update

A second update to the INTOR design concept was made at the end of Phase Two A Part 2, based upon the critical issues studies of that Phase. This updated concept is shown in Table 3.1 under the column headed "1985". The major changes with respect to Phase Two A Part 1 were a reduction in size (major radius), an increase in plasma current and the inclusion a LHR system for non-inductive current drive. An elevation view of the INTOR design concept is shown in Fig. 3.1.

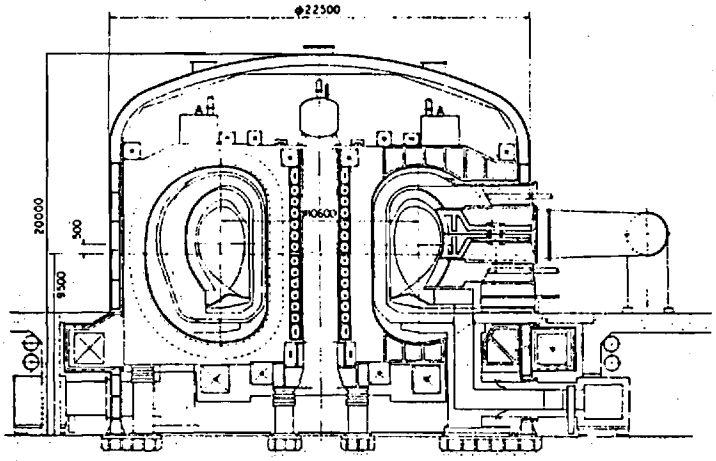

Figure 3.1. Elevation View of INTOR

3.5 Phase Two A Part 3 Design Concept Re-evaluation

The INTOR design concept was not updated at the end of Phase Two A Part 3 because it was felt that a self-consistent design iteration and optimization was needed at this point, and there was not time to conduct such an activity. However, the design concept was re-evaluated based upon the conclusions of the work of this Phase. Parameters which might be expected to change are indicated in Table 3.1 by an asterisk under the column with heading "1987". Anticipated changes are discussed in the following sections.

3.5.1 Impurity Control

Analytical studies and experimental data continue to support the choice of the poloidal divertor for impurity control and the choice of a high-Z (tungsten) divertor collector plate surface. Thus, the major aspects of the recommended impurity control system are unchanged from the INTOR design concept. Several modifications to the INTOR design concept may be necessary, however. A low-Z_{eff} limiter for startup may be required. If there remains the present uncertainty about being able to limit the disruption frequency to the presently assumed

reference value, then it may be necessary to install protective tiles on the first wall, at least during the initial phase. The reference value of Z_{eff} may have to be increased from 1.5 to 2.0, which would require allowance for up to a 50% increase in the power radiated to the first wall.

3.5.2 Operating Limits and Confinement

A variety of H-mode energy confinement scaling laws have been proposed over the past few years. Based upon these, the INTOR design concept is considered to have adequate confinement capability to achieve ignition, if there is no substantial degradation with heating power.

Although the INTOR design concept somewhat exceeds both the Murikami-Hugill and the Greenwald density limits, it is noted that these limits are exceeded by as much as a factor of 2 in experiments with intense auxiliary heating. So, the density in INTOR is very probably below the density limit.

Analytical and experimental results suggest that the Troyon beta-limit g-factor must be reduced from the value of 4 used in the INTOR design concept to 3.0 - 3.5 and that the safety factor q_I in INTOR must be increased from 1.8 to at least 2. Since

$$\beta\,(\%) = \frac{gI(MA)}{a(m)B(T)}, \quad q_I = \frac{5\left(\frac{1}{2}\left(1+\frac{b}{a}\right)^2\right)^{1/2} B(T)\, a^2(m)}{R(m)\, I(MA)},$$

some combination of increasing the plasma current, the magnetic field and the plasma elongation ($\frac{b}{a}$) and/or reducing the major radius in the INTOR design concept is probably necessary in order to achieve the performance objective (e.g. neutron wall load).

3.5.3 Current Drive and Heating

There is now a substantial experimental and supporting theoretical data base to support the use of non-inductive current drive, either by a combination of lower hybrid waves and neutral beams, or by neutral beams alone to achieve the basic performance objectives of INTOR, however, the predicted efficiency is low, and the required power may be on the order of 200 MW. This required power can be reduced substantially if the design is optimized for current drive. Thus, while inductive current drive is retained as the reference option in the INTOR design concept, the use of non-inductive current-drive to achieve the basic performance objectives of a new INTOR-like design concept is suggested, provided that such a design could be shown to be feasible and to have substantial advantage over an inductively-driven design.

New experimental data supports the previous choice of ICRH as the reference heating scheme in INTOR. However, if neutral beams and lower hybrid wave were chosen to contribute to current-drive in a new INTOR-like design concept, it would be suggested to use that system also for heating (and for added impurity control by flow reversal).

3.5.4 Electromagnetics

Analytical studies have established that the active control coils should be located inside the toroidal field coils and outside the shield and have confirmed that the first-wall/blanket structure is adequate for passive stabilization.

Analytical studies indicate that the INTOR poloidal field coil system could be designed more optimally. In particular, coils should be placed closer to the mid-plane. Leaving a large mid-plane window for horizontal access imposes a moderate penalty in terms of stored energy for small - to moderate - elongated plasmas, but imposes a large penalty for highly elongated plasmas.

3.5.5 Configuration and Maintenance

The reference INTOR maintenance concept is horizontal removal of large torus segments, which requires that a rather large "window" for access be left at the mid-plane, with the consequence, among others, that no poloidal field coils could be located near the mid-plane. Analysis of this maintenance scheme versus a vertical or oblique removal concept led to the conclusion that the simpler maintenance procedures associated with horizontal maintenance outweigh the penalty in poloidal field coil optimization for low-to-moderate plasma elongations, but that the vertical or oblique maintenance scheme would be preferred for moderate-to-large plasma elongation, for which the penalty in poloidal field coil optimization becomes too large for the horizontal maintenance approach based upon the removal of large torus segments. Thus, if it is necessary to increase the plasma elongation above about 2, as it seems it might be to satisfy plasma operating limits (section 3.5.2), then it may be necessary to change the horizontal maintenance concept to a vertical or oblique concept. Also, a combination of both concepts might have its merits.

For the reference INTOR maintenance concept, the use of a transfer cask for containing tritium and dust is recommended. This is required in order to meet the requirement for personnel access to the reactor hall.

Because of recent developments, an in-situ maintenance scheme is now recommended for plasma-facing components (e.g. protective tiles on the first wall).

Analysis of iron inserts to reduce the field ripple indicate that they would allow a reduction of about 50cm in the toroidal field coil bore or a reduction in the number of toroidal field coils from 12 to 10, without complicating the configuration significantly. Thus, the use of iron inserts would be recommended.

3.5.6 First-wall and Blanket

Analysis of the divertor collector plate, the first-wall and the breeding blanket confirms the choices that were made in the INTOR design concept. The reference divertor concept of tungsten bonded on copper cooled by water is predicted to have a lifetime of 2×10^4 cycles, limited during normal burn by fatigue and erosion. This implies that the divertor plate must be replaced 10 times during the lifetime of INTOR. A bare, water-cooled austenitic stainless steel first wall remains the recommendation, unless new information indicates that the frequency of disruptions would be much greater than is assumed in the present disruption scenario.

The reference breeding blanket concept - a ceramic breeding material, austenitic stainless steel structure and water cooling - is still recommended. It is possible to use water at relatively low pressure, which is recommended to enhance reliability.

Analysis indicates that by the use of beryllium multiplier and other stratagems, it is possible to achieve a tritium breeding ratio greater than unity, and hence to make INTOR self-sufficient in tritium production, without increasing the inboard dimension or the level of risk. Accordingly, it would be recommended to equip INTOR with a non-reactor relevant tritium producing blanket adequate to provide tritium self-sufficiency.

3.5.7 Design Sensitivity

Systems analyses indicate that the size and cost of an INTOR-like design is very sensitive to the ignition margin, Z_{eff}, plasma elongation, safety factor, value of the g-factor in the Troyon beta-limit, the neutron wall load, the shield attenuation and the allowable stress in the toroidal field coils. Thus, the size and cost of INTOR could be reduced by future developments that demonstrated improved energy confinement, improved impurity control, stability at higher plasma elongation, lower safety factor and larger values of the Troyon g-factor, higher radiation damage limit on the magnet insulators, and magnet structural materials that could operate at higher stress levels.

4. DATA BASE ASSESSMENT

A major task of the INTOR Workshop was the review of results from the worldwide R&D programs in fusion and related areas and an assessment of the data base for the design and construction of INTOR. Additional R&D programs that would be required to provide an adequate data base were

identified. This process went on throughout the course of the INTOR Workshop and had a major impact on the world-wide fusion R&D programs. In this section, the data base assessment activity over the course of the INTOR Workshop is described, and some of the most recent scientific and technical results are summarized.

In 1979 the adequacy of the then existing and planned world-wide R&D programs for providing the data base needed for the design and construction of INTOR was assessed. These results are summarized in very broad terms in Tables 4.1 - 4.5 and are given in detail in Ref. 1. The assessment was generally positive with respect to being able to provide the necessary data for designing and constructing INTOR to operate within 10-15 years, on the basis of the substantial amount of physics knowledge and technology then existing and the then-planned R&D programs. Certain crucial information was identified which would not be developed in then-planned programs, but which could be developed in time by a substantial expansion of existing programs and the initiation of new programs. This assessment, with its identification of which R&D areas were most important for a next-step experiment and of those R&D areas in which existing and planned programs were inadequate, provided a valuable focus for worldwide fusion R&D.

TABLE 4.1 INTOR PLASMA PHYSICS R AND D NEEDS [1]

R and D need	Status of world program: Adequate[a]	Inadequate[b]
Energy and particle confinement	X[c]	
β-limits and effect on transport	X	
Shape and profile control	X[d]	
Disruptions (detection and control)		X
Impurity control and divertor physics and technology		X
Neutral-beam heating technology		X
RF physics and technology	X[e]	
Fueling	X	
Plasma engineering (start-up, shut-down, burn control)	X[f]	

[a] programs are either planned or could be planned on existing or authorized facilities (with some upgrading if required) and using existing trained personnel.
[b] Either the available and authorized manpower and facilities are inadequate or a substantial increase in effort is needed.
[c] Except ripple effects and helium transport.
[d] Except current profile control.
[e] If near-term experiments are successful, an accelerated program would be needed.
[f] Except burn control.

TABLE 4.2 INTOR MAGNETICS AND POWER SUPPLY R AND D NEEDS [1]

R and D need	Status of world program	
	Adequate[a]	Inadequate[b]
SC toroidal-field coil	X	
SC poloidal-field coil		X
Energy storage and transfer	X	

[a] programs are either planned or could be planned on existing or authorized facilities (with some upgrading if required) and using existing trained personnel.
[b] Either the available and authorized manpower and facilities are inadequate or a substantial increase in effort is needed.

TABLE 4.3 INTOR BLANKET, MATERIALS AND TRITIUM R AND D NEEDS [1]

R and D need	Status of world program	
	Adequate[a]	Inadequate[b]
First-wall, limiter and divertor plates		X
Structural materials	X[c]	
Insulators and coatings		X
Integrated tritium fuel handling system	X	
Tritium clean-up, safety and monitoring systems	X	

[a] programs are either planned or could be planned on existing or authorized facilities (with some upgrading if required) and using existing trained personnel.
[b] Either the available and authorized manpower and facilities are inadequate or a substantial increase in effort is needed.
[c] For stainless steel only.

TABLE 4.4 INTOR ENGINEERING R AND D NEEDS [1]

R and D need	Status of world program	
	Adequate [a]	Inadequate [b]
Remote maintenance and assembly	X	
Penetration shielding for personnel access and component protection	X	
Vacuum pumping of helium		X
Diagnostics, data acquisition and control	X	

[a] programs are either planned or could be planned on existing or authorized facilities (with some upgrading if required) and using existing trained personnel.

[b] Either the available and authorized manpower and facilities are inadequate or a substantial increase in effort is needed.

TABLE 4.5 INTOR SAFETY AND ENVIRONMENTAL R AND D NEEDS [1]

R and D need	Status of world program	
	Adequate [a]	Inadequate [b]
Tritium containment, clean-up, conversion and biological effects	X	
Activation product generation, release and transport	X	
Experimental accident data and analytical techniques		X
Magnetic-field personnel exposure limits		X

[a] Programs are either planned or could be planned on existing or authorized facilities (possibly with some upgrading is required) and using existing trained personnel.

[b] Either the available and authorized manpower and facilities are inadequate or a substantial increase in effort is needed.

Table 4.6. Specific Short-Term Research and Development to Support INTOR Design

		Priority [a]
Physics		
P.1.	Beta limit scaling with triangularity and asymmetry	1
P.2.	High-density divertor operation (a) Divertor channel model tests (b) High-Z impurity backflow	1
P.3.	Atomic cross-sections for plasma edge processes	2
P.4.	Ripple-induced hot-ion transport model tests	2
P.5.	Ripple-induced thermal transport model tests	1
P.6.	Disruption characterization	1
P.7.	Low start-up voltage and volt-second experiments	2
P.8.	Vertical feedback control theory and experiments	1
P.9.	Pellet fuelling studies	2
P.10.	RF heating modelling and experiments	3
P.11.	Code development for consistency checks and optimization (including transport, equilibrium, stability, poloidal field configuration, etc.)	2
P.12.	Current profile control	1
Nuclear		
N.1.	Stainless steel surface behaviour under plasma disruptions	1
N.2.	Aluminium surface behaviour under plasma disruptions	3
N.3.	Graphite tile attachment	3
N.4.	Chemical sputtering of graphite	3
N.5.	In-situ recoating	3
N.6.	First-wall grooves	3
N.7.	Solid breeder development	1
N.8.	Liquid breeding material development	3
N.9.	Solid breeder/coolant tube interfaces with predictable thermal conductance	2
N.10.	Tritium permeation	1
N.11.	Chemical sputtering of refractory metals (tungsten and molybdenum)	1
N.12.	High-temperature fatigue of tungsten and molybdenum	1
N.13.	Behaviour of brazing under thermal cycle conditions	1
Engineering		
E.1.	Fatigue and fracture mechanics data for room-temperature and 4 K materials	1

[a] Priority
1 – Required for the reference INTOR design, highest priority
2 – Required for the reference INTOR design, secondary priority.
3 – Not required for the reference INTOR design, but of importance for other design options.

Table 4.7. Additional Short-Term Research and Development to Support INTOR [a]

		Priority [b]
Physics		
P.1	Plasma behaviour near beta limits	1
P.2	Confinement scaling in auxiliary heated tokamaks	1
P.3	Plasma equilibrium control	1
P.4	Plasma profile control	1
P.5	Reactor prototypical ICRF heating	1
P.6	ICRF code development	2
P.7	RF start-up assist	2
P.8	High power LH and EC heating	(2)
P.9	Quasi-steady-state mode of operation	(1)
P.10	Characterization of high- and low-temperature edge regimes	1
P.11	Edge particle and energy fluxes	1
P.12	Divertor channel behaviour	1
P.13	Impurity transport	1
P.14	Limiter pumping characteristics	(1)
P.15	Molecular and low-temperature charge-exchange data	2
Nuclear		
N.1	Self-sputtering yield of main candidate materials	1
N.2	Sputtering by tritium	2
N.3	Properties of re-deposited metals	1
N.4	Irradiation effects on non-replaceable high-flux materials (60 dpa)	1
N.5	Irradiation effects on replaceable high-flux materials (30 dpa)	1
N.6	Tritium permeation and inventory, including irradiation effects	1
N.7	Eutectics development	(1)
N.8	14 MeV neutronics integral experiments	2
Engineering		
E.1	High-power ICRF system demonstration	1
E.2	Improved structural concepts for first wall/divertor/limiter	1
E.3	First wall outgassing procedure	2
E.4	Tritium pellet injector	1
E.5	Superconductors for fields above 10 T	1
E.6	Low-loss, high-current 8 T superconductors	1
E.7	TF coil mechanical and electrical properties	1
E.8	Safety circuits to cope with short-circuiting coils	2
E.9	Low-loss poloidal field coil concept	1
E.10	Intermediate-scale PF coil demonstration	1
E.11	Computational tools for transient electromagnetics	1
E.12	Torus maintenance methods and procedures	1
E.13	Adequate torus resistance	1
E.14	Voltage withstand criteria for components within the torus	2
E.15	Pump development	2
E.16	PF coil power supply system optimization	2

[b] Priority
1 – Required for the INTOR reference design: highest priority.
2 – Required for the INTOR reference design: secondary priority.
Numbers shown in parentheses refer to tasks which are not required for the INTOR reference design but which are of importance for other design options.

4.2 Phase One Assessment

The Phase Zero data base assessment was confirmed during the conceptual design phase of 1980-81. The development of technology to support a solid tritium breeding blanket was identified as an additional required R&D area and the detailed of the required development program were specified [2].

During the Phase One conceptual design activity, a number of critical uncertainties were identified which prevented firm design decisions from being taken. In most instances, it was apparent that these uncertainties could be resolved by specific, short-term R&D programs. These R&D needs, which are given in Table 4.6, provided a detail focus for worldwide fusion R&D.

4.3 Phase Two A Part 1 Assessment

Review of the new results from the R&D programs continued to confirm the physics and technology bases for the Phase One INTOR conceptual design. By 1983, the physics data base on ICRF heating was sufficient that ICRF could be recommended as the reference heating method, because of its engineering advantages with respect to neutral beams. Several additional specific, short term R&D programs (Table 4.7) were recommended to resolve uncertainties that prevented the taking of definite decisions on the INTOR design.

4.4 Phase Two A Part 2 Assessment

The assessment in 1985 generally continued to support the Phase Two A Part 1 INTOR design concept, with a few exceptions. For example, new experimental data on beta limits, which can be expressed by the "Troyon" formula

$$\left(\frac{\text{plasma pressure}}{\text{magnetic pressure}}\right)_{\max} \equiv \beta_{\max} = \frac{g\ I\ (MA)}{a(m)\ B\ (T)}\ ,\quad g = 3\text{-}4\ \%$$

indicated that it was necessary to increase the plasma current (I) in order to achieve the performance objectives.

4.5 Phase Two A Part 3 Assessment

The most recent (1987) INTOR data base assessment [5] is summarized in this section.

4.5.1 Impurity Control

There has been a substantial amount of new data from both poloidal divertor and from limiter experiments in tokamaks. There is further evidence that a divertor with an open geometry, of the type envisaged for INTOR, is capable of producing high recycling conditions which are desirable in order to minimize sputtering of the divertor target. The concentration of impurities within the main plasma is generally lower for a divertor than for a limiter. However, the concentration of low-z impurities (noteably oxygen) is affected less than the high impurities by a divertor. There is often substantial emission of radiation within the divertor region and this is indicative of high recycling. It appears that the H-mode can be most readily accessed by operations with a poloidal divertor. In constrast, H-mode operation has been observed only in one limiter experiment. A detrimental aspect of the H-mode is that, in certain conditions, this mode causes impurities to accumulate on the plasma axis. However in cases of limiter operation, the temperature of the plasma in contact with the limiter is high and this is likely to lead to high rates of sputtering and erosion of the limiter. Exhaust of neutral gas can be quite efficiently performed by a pumped limiter.

4.5.2 Beta Limit

Experimental results on the operational limit to the plasma beta correspond to values of the Troyon factor g in the range 3 to 3.5%Tm/MA. Improved analysis of the observations has led to a slight decrease in g with respect to earlier findings. The same range of values is

found in ideal MHD stability analyses of D-shaped plasma having an aspect ratio A $\geq$ 4, which has been extended to cover moderately large elongations K $\leq$ 2, provided that the MHD q at the plasma edge $q_a \geq 3$. For large elongation and lower q_a there is an appreciable degradation of the Troyon factor. Up-down asymmetry, for such plasmas, causes a decrease of the beta limit, typically by 10 to 20%, for cases of practical interest. Equilibria with a safety factor on axis $q_o < 1$ were considered, but a clear advantage of operating in this regime could not be identified. At extremely low aspect ratio (A $\leq$ 2), several results have indicated an enhancement of g up to about 4.5% $\cdot$ Tm/MA, provided that q_a is above a critical value, which increases with decreasing A. For indented plasma, while the ideal ballooning stability limit is enchanced, the kink modes is destabilized so that efficient wall stabilization is essential for achieving high beta. It remains uncertain whether this can be provided. The second stability regime of ideal ballooning mode can be reached either in D-shaped plasmas for sufficiently high q_o or in sufficiently indented plasma. However, in these cases kink mode instability is enhanced. Furthermore, a wide range of the plasma has to be nearly shear-free, a situation in which low-n internal modes tend to be destabilized. Also resistive destabilization of high-n modes is a concern. In conclusion, moderately elongated D-shapes (K $\geq$ 2) appear attractive for INTOR and allow enhancement of the plasma beta, but more unconventional solutions to increase beta presently are too uncertain to rely upon.

4.5.3 Density Limit

The density limit, if extrapolated according to common Murakami-Hugill-like scalings, tends to be a more stringent limitation to the plasma pressure, at temperature T $\leq$ 10 keV, than the beta limit. However, the physics understanding of this limit is incomplete, and results for discharges with intense additional heating generally show an enhancement of the density limit and indicate deviations from the Murakami-Hugill scaling. In JET, the density limit appears when the radiation losses become equal to the power input, a criterion which, when extrapolated to INTOR, predicts an appreciably higher density limit than the Murakami-Hugill scaling. Quantitative predictions, however, sensitively depend on the plasma edge parameters in this case. The limit to the safety factor, at least at modest values of beta and for conventional circular and D-shaped plasmas, is at $q_a \geq 2$ (which for poloidal divertor configurations is to be referred to 95% of the magnetic flux).

4.5.4 Disruptions

Operational limits are often due to the appearance of disruptions. In view of results from JET and TFTR, very short energy quench times, of the order of 0.1 ms, must be considered to be a possibility in INTOR. The energy deposition profile in a poloidal divertor configuration remains unknown so that deposition of up to the total plasma kinetic energy on either the divertor plates or the first wall must be considered. The current quench rate is determined by the evolution of the plasma parameters after energy quench, taking account of the electromagnetic coupling to the surrounding passive conducting structures and the capacity of the active position control device. If efficient position control is provided, a maximum current decay rate of $3 \cdot 10^8$ A/s appears appropriate for INTOR.

4.5.5 Energy Confinement

Extrapolation of energy confinement to INTOR involves large uncertainties. It is considered that relying on operating INTOR in a regime of improved confinement (H-mode) is a reasonable working hypothesis, although there are still major uncertainties with respect to the reactor relevance of this regime. These are related, e.g., to the possibility of controlled steady-state operation with limited impurity contamination and to the compatibility with RF heating and current drive, as well as with efficient power and particle exhaust for acceptable working conditions of the divertor plates and first wall. Also the scaling of energy confinement in the H-mode remains uncertain, in particular with respect to plasma size and plasma temperature (or, equivalently, heating power), but to some extent also with respect to other parameters such as plasma current and density. These issues are key research items in the ongoing tokamak physics program and are expected to be clarified within a few years.

4.5.6 Current Drive and Heating

An IAEA Specialist's Meeting on non-inductive current drive was held in September, 1986 to understand the present status of experimental and theoretical developments [6]. This specialists' meeting concluded that there has been considerable progress in research on non-inductive current drive in tokamaks. This progress includes all functions for which non-inductive current drive can be used, i.e., steady-state current drive, current ramp-up and transformer recharging, control of the current profile and MHD behaviour, as well as current initiation. The experimental data base is greatest for LH current drive. However, LH is predicted not to penetrate sufficiently in reactor grade plasma to drive current in the central region. The experimental data base for NB current drive is growing rapidly. Progress seems sufficient for a tentative extrapolation to reactor conditions for LH and NB current drive. Thus, either a combination of LH and NB or NB alone could be used for non-inductive current drive in INTOR.

Recent high power experiments have made impressive progress in plasma heating by means of different methods, as shown by the following examples:

NBI:

TFTR: T_{i0}/T_{e0}≈27/6 keV at $\bar{n}$≈3x10^{19} m^{-3} P_{inj}≈13.5 MW

LHW:

JT-60: T_{e0}≈6 keV at $\bar{n}$≈1·7x10^{19} m^{-3} P_{inj}≈2.4 MW

ASDEX: T_{e0}≥4 keV at $\bar{n}$≤10^{19} m^{-3} P_{inj}≈0.75 MW

PLT: T_{e0}≈5 keV at $\bar{n}$≈10^{19} m^{-3} P_{inj}≈0.6 MW

ICW:

JET: T_{e0}/T_{i0}≈7/5.5 keV at $\bar{n}$≈(3-4)x10^{19} m^{-3} P_{inj}≈5.5 MW

ECW:

T-10: T_{e0}≈8-10 keV at $\bar{n}$=1.5x10^{19} m^{-3}(He) P_{abs}≈2.2 MW

4.5.6.1 Lower hybrid waves

Lower hybrid waves have confirmed their efficiency for current drive in its different modes (steady-state up to the level of 2 MA, ramp-up, transformer recharge, start-up), for electron heating, for sawtooth stabilization and for profile control. Current drive efficiency of the order of that predicted for INTOR-like devices has been achieved experimentally, and current drive operation at high density, provided the frequency is adequately chosen, has also been demonstrated.

Convergence between theory and experiments is good in most domains. Lower hybrid waves, as the best documented current drive method and with its broad range of applications, thus reinforces its position as a major candidate for most of the functions to be fulfilled by external power in INTOR, with the reservation that a poor penetration in high temperature, dense plasma is predicted.

4.5.6.2 High frequency fast waves

High frequency fast waves have the potential advantage of better penetration than lower hybrid waves and have shown a comparable current drive efficiency. The data base has grown recently but is still too meager to allow firm conclusions on its potential for reactor grade plasmas.

4.5.6.3 Ion cyclotron waves

Ion cyclotron waves have confirmed their position as one of the major candidates for plasma heating in the next step and future devices. Current drive by means of low frequency fast waves in the range $\omega<5\omega_{CD}$ (avoiding ion resonances) shows favorable

prospects regarding efficiency, penetration and insensitivity to α particles. However, the present data base on low frequency fast wave current drive is now sufficient to certify this potential.

4.5.6.4 Neutral beams

The abundant data base on beam injection confirms the potential of this method for plasma heating. In recent experiments on neutral beam current drive in TFTR and JET, driven currents up to more than 0.5 MA have been achieved. In TFTR, the interpretation of these experiments is consistent with the existence of a significant bootstrap current.

4.5.6.5 Electron cyclotron waves

Electron cyclotron waves have demonstrated their usefulness for plasma heating. Their main handicap for this application is still the absence of efficient high-power generators, although the source development is making rapid progress. EC waves have also proven their utility for start-up assist and MHD stabilization and have a unique potential for local temperature control; high power coupling through the plasma edge is not a problem. This points to their use in INTOR for such functions. However, they do not appear favorable for current drive, except in the innermost plasma region.

4.5.7 Materials

New materials data have been reviewed on austenitic and ferritic steels, graphite and c/c composites, ceramic breeder materials, liquid breeder materials, divertor materials, and magnet materials.

4.5.7.1 Austenitic steels

Additional information include: sensitivity to aqueous stress corrosion, low temperature radiation effects on mechanical properties, effect of radiation on weldments, and fabrication experience of wall components. These new data show: aqueous stress corrosion cracking of austenitic steels, particularly in the presence of irradiation, is identified as a serious feasibility issue for the reference INTOR first wall/blanket structure. Significant loss of tensile ductility (the uniform elongation much less than 1%) and fracture toughness is observed after low temperature (< 300° C) irradiation to ~ 15 dpa at He/dpa ratio of ~ 15. Fatigue properties for cold-worked and solution annealed material are similar and radiation has only a modest effect on the fatigue properties of austenitic steel. The tensile and fatigue properties of austenitic steel structures bonded by the HIP process with > 95% bonding ratios (e.g. surface area) are similar to properties for the base metal. Austenitic stainless steel is the only reasonable structural material for the low temperature first wall and blanket of INTOR.

4.5.7.2 Ferritic/martensitic steels

The selection of ferritic/martensitic steel as the first wall/blanket structure for the low temperature, cyclic operating conditions of INTOR is not reasonable. The DBTT of ferritic steels is increased more than 200°C by low temperature (< 300°C) irradiation. The effect of hydrogen is of particular concern at the low temperatures where release of the internally generated hydrogen may be inhibited. The hydrogen effect may be even more critical for irradiated material, e.g. ΔDBTT, and/or for weldments.

4.5.7.3 Graphite and c/c composites

Three aspects of graphite and c/c composites, form of redeposited material, radiation effects, and tritium retention have been evaluated. A thin layer of amorphous carbon is observed on the entire chamber wall of the present machines after operation with graphite. The amorphous material has a large capacity for retaining tritium (0.4 tritium atom per carbon atom). The predicted radiation lifetime of nuclear grade graphite at 800-1200°C is < 1MW·y/m^2. The effect of high helium generation rate (He/dpa ~ 300) is unknown but may be significant at low temperatures (< 1200°C). The c/c composites provide significant tensile strength and fracture toughness advantage but are predicted to be significantly less radiation damage resistant because of the large anisotropy compared to nuclear graphites.

4.5.7.4 Ceramic Breeder Materials

Significant R and D efforts have been conducted with emphasis on the candidate ceramic breeder materials: Li_2O, $LiAlO_2$ and Li_4SiO_4. Mass transfer/weight loss of Li_2O in flowing He at low moisture contents generally correspond to thermochemical data. A significant transfer of LiOH from high temperature zones to lower temperature zones is observed. The data from FUBR experiments show that the swelling in $LiAlO_2$ remains low up to 3 at % ^6Li burnup and the other two ceramic breeders show considerable increase in swelling with burnup at irradiation temperatures of 700 and 900°C (4% for Li_2O, 2.5% for Li_4SiO_4 and < 0.5% for $LiAlO_2$ at 700°C). In-pile tritium release behavior from Li_2O, $LiAlO_2$ and Li_4Sio have been examined. From VOM results, the tritium release from Li_2O seems to be the fastest, followed by that of Li_4SiO_4 for similar temperature and grain size. Tritium release rates from ceramic breeders are sensitive to grain size and temperature. Especially small grain size in $LiAlO_2$ is critical because of low tritium diffusivity. At sufficiently high temperatures or small grain size, most of the tritium should be released from all candidate ceramic breeder materials.

4.5.7.5 Liquid breeder materials

Eutectic 17Li-83Pb alloy and aqueous Li-salt solutions are studied as liquid breeder materials. The estimated operating temperature limit based on corrosion by 17Li-83Pb is < 400°C for austenitic steel and 450-500°C for ferritic steel. Tritium permeation from 17Li-83Pb is a concern because of low solubility of tritium. There is a serious concern regarding stress corrosion cracking by Li-salts and tritium recovery.

4.5.7.6 Divertor materials

Primary candidate materials include tungsten plasma facing materials bonded to a copper heat sink. Liquid metal materials and He-burial materials are studied as innovative target materials. The cyclic fatigue data by torsion fatigue tests for the tungsten-copper brazed specimen show that the lifetime of brazed specimens in the low strain range agrees well with that of copper, and in the high strain range depends on the strength of the interface bond and tungsten. The loss rates of the liquid metal film to the plasma have been estimated for three candidate liquid materials: lithium, tin and gallium. The evaporation losses are much less than the sputtering losses up to 500°C for lithium and up to 900°C for tin. Predicted sputtering rates for a metal surface do not differ substantially from those for the same material in the solid state. The tritium inventory in the liquid lithium layer is calculated to be on the order of a few grams, and the predicted one for liquid tin is even less. The corrosion data on gallium show that the refractory metals are generally highly corrosion resistant, the steels moderately resistant and Ni, Cu and Al very low resistant. For He-burial material, candidate materials are vanadium, nickel and iron. The minimum energy for effective helium trapping (~ 30 at % trapping fraction) is estimated to be ~ 30-50 eV.

4.5.7.7 Magnet materials

The radiation limits for Nb_3Sn are estimated to be $2x10^{18}$ - $2x10^{19}$ cm^{-2} The dose limits for epoxy insulators and polyimide insulators are predicted to be in the range of 0.5 - $9x10^9$ rad and $1x10^9$ - $1x10^{10}$ rad, respectively, depending on the shear stress requirements.

4.6 Physics Design Guidelines

The limits to stable operation and the confinement properties of the INTOR plasma as well as related issues were reassessed on the basis of recent experimental and theoretical results. From this a set of new guidelines was evolved that significantly differs from those adopted in Phase Two A Part 2.

In the following the new guidelines are listed together with comments where appropriate:

4.6.1 Beta limits:

$$\beta_{max}(\%) = g\, I_p\,(MA) / a(m)\, B(T)$$

with g = 3.0 - 3.5

A value of g = 3.5 corresponds to the best experimental results achieved so far. It must be noted that g is dependent on configuration and safety factor.

4.6.2 Useful Beta:

β_{total}/β_{DT} 1.3, for <T> = 10 keV,

1.4, for <T> = 15 keV,

1.5, for <T> = 20 keV,

This is consistent with, e.g., n_α/n_e = 5% for α-particle

and n_z/n_e = 1% with Z = 6 for impurities

(a higher impurity content may have to be allowed for);

The pressure contribution of fast a-particles is respectively 13/25/32% of the pressure of the thermal component for the three values of the temperature referred to.

4.6.3 Fusion power density:

$$\langle P_{fus}(MW/m^3)\rangle = c\beta_{DT}^{\,2}\, B^4(T)$$

The coefficient c depends on profiles of density and temperature. For <T> = 10 to 15 keV, an approximation is

$$c = 1.2\,\frac{(1+\gamma)^2}{1+2\gamma}$$

where γ characterizes the pressure profile:

4.6.3 Beta limit: $p = p_o[1 - (r/a)^2]^\gamma$.

4.6.4 Plasma density:

The upper limits of the average electron density for ohmic discharges can be estimated from

$$\langle n_{e,max}(10^{19} m^{-3})\rangle = g \frac{B(T)}{R(m)\, q_I} \quad \text{(Murakami-Hugill) with } g = 15$$

or, alternatively,

$$\langle n_e(^{max}(10^{19} m^{-3}))\rangle = G \frac{I(MA)}{a^2(m)} \quad \text{(Greenwald) with } G = 2$$

taking $\langle n_e\rangle / \langle n_{DT}\rangle = 1.2$

For discharges with intense heating these limits can be about 3 times larger.

Note: The specification for $\langle n_{e,\,max}\rangle$ for extrapolation to INTOR conditions will in the future have to be replaced by one having a more convincing physics basis, such as the plasma power balance.

4.6.5 q limit (divertor operation):

Fulfill both

and $n_z/n_e = 1\%$ with $Z = 6$ for impurities

(i) $q_I = 5K^2\, a(m)^2\, B(T) / R(m)\, I_p(MA) \geq 2$

with $K = \frac{dl}{2\pi a}\, [\frac{1}{2}(1 + b^2/a^2)]^{1/2}$

and

(ii) the q = 2 surface is at distance inside the separatrix (at outboard side) not smaller than 0.1 m.

The lower limit of q_I depends on the configuration. The specification given is consistent with the present data base, for moderate beta, but the note under 4.6.4 also applies here.

4.6.6 Temperature:

$$\langle T_e\rangle = \langle T_i\rangle \leq 20 \text{ keV}$$

The coincidence of electron and ion temperatures is a reasonable approximation.

This constraint is a reminder that reaching high temperatures in a tokamak may be difficult; it may have to replaced by a specification on how a possible bound on the electron temperature gradient affects confinement.

4.6.7 Configuration:

D shape;

either moderate elongation in a single-null configuration:

$K = b/a \geq 1.6 / 2.2$,
triangularity $\delta \leq 0.3 / 0.7$

where the first number refers to the upper half and the second to the lower half, to be taken on the separatrix,

or strong elongation in a symmetric double-null configuration:

$1.7 \leq K \leq 2.2$,
$0.3 < \delta < 0.7$

on the separatrix.
Higher elongation requires a higher value of q_I.

Note: the integrity of the scrape-off layer must be ensured; the minimum width of the scrape-off layer is 0.1 m (outboard for a single-null configuration; outboard and inboard for a double-null configuration).

4.6.8 Plasma energy confinement:

Rely on a regime of enhanced confinement ("H-mode characteristics").

Relevant scalings of the global energy confinement time $\tau_E(s)$ are

ASDEX H (Gruber): $\tau_E(s) = 0.1\ I_p(MA)\ R(m)$

ASDEX H (Schliersee): $\tau_E(s) = 0.25\ I_p\ (MA)$

Kaye-Goldston (x2): $\tau_E(s) = 0.147\ I_p(MA)^{1.24}\ R(m)^{1.65}\ (b/a)^{0.28}$ x

$n_e(10^{20}m^{-3})^{0.26}\ P_{tot}(MW)^{-0.58} a(m)^{-0.49} B(T)^{-0.09}$

Mirnov: $\tau_E(s) = 0.15\ I_p(MA)\ a(m)\ (b/a)^{0.5}$

Mereshkin-Mukhovatov: $\tau_E(s) = 0.017\ n_e(10^{20}m^{-3})\ a(m)^{0.25}\ R(m)^{2.75}\ q_I\ T_e(keV)^{0.5}$

JAERI I: $\tau_E(s) = 0.19\ a(m)^2\ B(T)$

JAERI II: $\tau_E(s) = 0.071\ a(m)\ R(m)\ B(T)\ (b/a)^{1/2}$

as long as the values obtained are shorter than the neo-Alcator confinement time

$$\tau_E(s) = 0.11\ n_e(10^{20}m^{-3})\ a(m)\ R(m)^2\ q_I\ .$$

Matching with the neo-Aalcator scaling via an inverse-square interpolation is a possible procedure for some of the above scalings.

Note that the actual uncertainty about energy confinement may be larger than the spread of the predictions obtained in this way.

Required confinement capability:

$$C = \frac{P_\alpha}{W_{th}/\tau_E + P_{rad}} \geq 1.5 \quad \text{for ignition,}$$

$$C = \frac{P_\alpha + P_{ext}}{W_{th}/\tau_E + P_{rad}} \geq 1.5 \quad \text{for driven operation,}$$

to be satisfied by most of the above scalings of τ_E; here P_α is the α - particle power, W_{th} is the thermal plasma energy, P_{rad} is the power radiated from the plasma interior (bremsstrahlung, cyclotron radiation, ...) and P_{ext} is the external heating power applied for driving the discharge.

4.6.9 Fast α-particle confinement:

Power loss: 3%

This loss causes a load on the first wall localized between the TF coils and above (or below) the equatorial plane of the plasma at small poloidal angles.

Note: This guidance has an insufficient physics basis. The simulations of fast α-particle losses induced by the toroidal field ripple (see 4.6.10) have not yet led to a generally accepted quantification. In addition, α-particle induced instabilities may degrade fusion α-particle confinement further.

4.6.10 Toroidal field ripple:

τ_E ($\leq$ + 10)% at outboard edge,
$\leq$ + 0.1% at centre.

This specification is given to keep fast ion (in particular fusion α-particle) losses within acceptable limits, but it is presently not clear whether the condition is actually adequate.

4.6.11 Resistive loop voltage:

$$U(V) = 0.1\, Z_{eff}\, \frac{R(m)\, I(MA)}{K\, a(m)^2\, \langle T(keV)\rangle^{3/2}}$$

with $Z_{eff} = 1.5 - 2.$

The specification of U does not include trapped particle effects explicitely. A value of Z_{eff} of 1.5 is consistent with the case given under 4.6.2 and allows in addition a small amount of medium/high Z impurities (e.g. 0.02% Fe).

5. CRITICAL ISSUES

During the Phase One conceptual design, several areas were identified in which uncertainties raised questions about the basic feasibility of the design or in which an alternative design solution appeared to be preferable. In some areas, the uncertainties arose because of absence of fundamental information and further progress must await results from the R&D programs -- in these instances, R&D program requirements were defined, as discussed in the previous section. However,

in some areas, it seemed possible that the issues could be resolved, or at least better understood, by intensive analysis -- in these instances critical issues studies were defined. Five areas were identified for intensive analysis of critical technical issues -- impurity control, electromagnetics, current drive and heating, configuration and maintenance, blanket and first-wall. In this section, a few of the most important results of the critical issues studies are summarized.

5.1 Impurity Control

The open, high-recycling poloidal divertor concept, which is now included in most tokamak experiments that have a divertor and in virtually all designs for future experiments, was originated and developed within the INTOR Workshop. Introduction of the open, high-recycling divertor concept in INTOR has influenced the tokamak experimental programs in ASDEX, PDX/PBS, D-III and other experiments.

The development and comparison of analytical models for predicting the plasma characteristics within such a divertor has been a major task of the INTOR Workshop. Models developed by the four Parties to the Workshop have been successfully compared among themselves and to experimental data. An INTOR-related IAEA Specialist's Meeting [6] was held to bring together experts in plasma edge and divertor modeling. As a consequence, there now exists a relatively high degree of confidence in the ability to predict the performance of such divertors.

As a consequence of the work to date, the conclusion is that a high recycling divertor with a tungsten target is the best available impurity control system to maintain a clean plasma in INTOR and to ensure low target erosion during a fully inductive operational scenario. Impurity control with current drive during an ignited burn is likely to be different due to the increased power loads. A stable radiating edge layer and, in this respect, impurity flow reversal would be beneficial. Continuous current drive with a sub-ignited and clean plasma may also be difficult and possibly only achievable with very low power amplification factors. During inductive ramp-up, it is expected that adequately high recycling conditions can be established within the divertor, but this is less certain in the case of non-inductive ramp-up. Consideration has been given to the use, in the initial phase, of low-Z target material (e.g. carbon and beryllium). During an inductively driven ignited burn, such targets sputter at rates which are 10 times that of tungsten but, even so, target lifetimes exceed the likely duration of the initial phase of operation. Such materials are unlikely to be suitable for the more extended engineering testing phase unless the divertor target surface can be readily renewed.

The relative merits of single and double null poloidal divertors have been assessed. Experimental data indicate that H-more operation can be most easily achieved with a single-null divertor, whereas the conditions needed for attainment of high elongation and control of vertical position favor the double-null divertor. Edge plasma modelling predicts that the peak power loading of the divertor target is lower in the case of the double-null divertor and that, based upon present understanding, adequate exhaust of the helium ash can be obtained by pumping only the bottom divertor chamber.

The pumped-limiter was examined, as an alternative to the divertor, because it offered the advantage of a simpler and lower cost INTOR concept. The potential performance of the pumped-limiter was evaluated by an assessment of the pumped-limiter experimental data and an extrapolation of these data to predict the performance of a pumped-limiter in INTOR, using large-scale computational models. It is expected that pumped-limiters have less potential for impurity control and more of a sputtering erosion problem that poloidal divertors, owing to the greater difficulty anticipated in establishing a high-recycling, low edge temperature regime in the former than the latter. The overall conclusion is that the poloidal divertor will offer many advantages over a pumped-limiter in INTOR.

5.2 Electromagnetics

olishing the requirements for control of the plasma position against radial and hori. al lacements was a major task of the INTOR Workshop. Calculational models were de pe intercompared among the parties. It was determined that for this purpose a model he plasma as a single circuit element (as opposed to a distributed plasma current equate. For the control of the radial position of the elongated plasma only a slow em is required since the plasma is stable against radial displacements. The easiest o introduce a poloidal field amplifier to control the current in the outer PF coils. on, feedback is needed for the remaining PF coils to control the plasma current and

shape (eg. elongation, triangularity, null-point position, etc.) as functions of time during the pulse. The control of the vertical position of the plasma necessitates the use of a fast power supply since the plasma is unstable in this direction with a growth time for INTOR and INTOR-like devices of the order of 5 - 50 ms. Analyses identify the need for passive conducting material near the plasma, in order to reduce the requirements on the active control system, but indicate that the normal torus structure can provide this function, provided that the toroidal segmentation is not too great. The active radial stabilizing field is produced by a pair of axisymmetric coils in series, but these cannot be placed outside the TF coils because the time-varying field would produce very high AC losses. Estimates for the control power required for the INTOR stabilizing coils is in the range of 20 to 50 MW. The growth rate of the vertical instability is shorter for high elongation plasmas than for low elongation plasma with the same passive structure. The single-null and double-null plasma configurations differ in the requirements they impose on the control system. This is because, in the case of the single-null, there is a coupling between radial and vertical displacements, whereas this is not the case for the double-null. Also the symmetry of the double-null system eases the problems of the plasma position detection system. The overall conclusion is that the active control system for the double-null is simpler, more reliable and less demanding in power than for the case of the single-null at the same average plasma elongation.

The poloidal field (PF) coil configuration and currents are determined by the plasma current, plasma current density distribution, plasma shape and allowable coil locations. Benchmark calculations were carried out by the four INTOR participants for specified double and single-null plasma configurations. The problem to be solved was to find the optimum PF coil locations and currents for minimum objective function (eg. superconductor mass current sum or magnetic energy). The results of all four participants were in very good agreement for the double-null plasma configuration. However, for the single-null plasma configuration the results were rather more spread. This spread was due mainly to the high multipolarity of the PF coils system and the effect of mutual "screening" of closely located coils plus small differences in the plasma shape. The conclusion of this exercise was that all codes give the same results if the same constraints are used.

Extensive analyses of the impact of the various possible access schemes to the torus components have been performed. The reference INTOR concept incorporates a horizontal access approach, which requires that no PF coils be placed near the mid-plane. An alternative is a vertical access approach, which allows coils near the mid-plane. High elongation plasmas require less magnetic field energy in a vertical access scheme than is required in a horizontal scheme. For lower elongation plasmas, there is no clear-cut advantage from the magnetic stored energy point of view between the two access schemes.

If current density and field are increased in both the toroidal field (TF) and PF coils the reduction in overall machine size can be significant. However, in both coil systems, the current density is limited by structural requirements rather than by the superconductor material properties. The superconductor comprises only a small fraction of the coil area, the rest being occupied by coolant, insulation, copper to provide thermal quench protection, and structural reinforcement. Improvements in structural materials are seen as the major avenue to increasing winding pack current density.

5.3 Non-Inductive Current Drive and Heating

A series of benchmark calculations for non-inductive current drive have been performed by all four Parties to the INTOR Workshop. Analysis of studies on steady-state and cyclic current drive and on profile control for INTOR-like devices ($\bar{T}_e$=20 keV, $\bar{n}_e$=0.7x10^{20} m^{-2}) have shown comparable current drive efficiency (γ=0.3 to 0.5 (10^{20} m^{-2} MA/MW) for a number of drivers: LHW above 3 GHz; NBI; high frequency (0.2 to 1 GHz) fast waves; and low frequency (20 to 70 MWz) fast waves. For FW and NBI, this high $\bar{T}_e$ was chosen to increase γ.

Analyses indicate that LHW will not penetrate to drive current in the central regions of a full-density INTOR plasma. Thus, LHW could be used for transformer recharging or initial start-up, at reduced density. Alternatively, beam injection (or possibly IC waves or EC waves) to drive part of the current in the plasma core, with lower hybrid waves for current drive in the plasma exterior, offers an attractive possibility for steady-state current drive by using the respective techniques in the regions where their performances are optimum.

A deuteron energy in the range of 0.4 to 0.8 MeV is optimum for NBI current drive for INTOR-like devices. One necessary condition for the use of NBI is thus a successful development of the technology based on negative ions.

Bootstrap current calculations for INTOR show that external current drive power may be reduced by 50% to 90%, depending on plasma density profile, if this neoclassical current contribution should appear.

According to the modelling studies, current profile control by means of the HFFW should be flexible; and neutral beams have also shown such potential, which is restricted, however, by shinethrough constraints. Although the LHW profile control has been demonstrated experimentally, it cannot be relied upon for this function in the core of reactor grade plasma; it could be used in low density phases of the discharge. The potential of EC waves for local temperature control still appears valid.

For transformer recharging or initial current ramp-up, the LH driver is attractive. At low $\bar{n}_e$ and $\bar{T}_e$ ($\simeq 4\times10^{18}$ m^{-3} and $\simeq$2 keV) centrally peaked current (8 MA) may be maintained with a reverse emf of -0.01 V/m and $P_{CD} \simeq 20$ MW is needed for INTOR. The option of using NBI appears more restrictive. The fuelling by beams imposes operation at $n_e \gtrsim 1\times10^{19}$ m^{-3}, which enhances the power requirements. Using $E_b \simeq 0.5$ MeV, it would appear that $P_{CD} \simeq 50$ MW and $Z_{eff} \simeq 10$ is needed for the same INTOR recharge/rampup specification. A positive result is a smaller cycling of the wall loading and of β with this higher density. However, the considerable NB shinethrough fraction is a real engineering problem at such densities.

Thus, for current drive applications, among several proposed methods, only lower hybrid waves, with a large data base, and neutral beam injection, with more limited one, offer presently reliable prospects. For cyclic current drive, lower hybrid waves appear as the method choice. For steady-state applications both methods have drawbacks: the LHW because of a restricted penetration in hot and dense plasmas, and NBI because the high beam energy required imposes the use of negative ion technology, the development of which is still at an early stage. A combination of both methods appears at present the best option.

Several options for the choice of a heating method can be extrapolated confidently to the reactor level on the basis of their present physical and technical achievements. Ion cyclotron waves, lower hybrid waves and beam injection (at moderate energy level) may already be relied upon. While recent progress in the development of required generators for electron cyclotron waves could soon overcome the source deficiencies of this method. For the heating mode envisaged for INTOR, i.e. second-harmonic heating of deuterium, an intense modelling effort has been made. This includes wave coupling to the plasma, wave propagation and absorption. The main results are the following: while for plasma parameters approaching the thermonuclear regime the absorption is single-pass, this is not expected to be the case for the Ohmic target plasma. The addition of a few percent of H ensures single-pass absorption also under Ohmic conditions, but makes H minority heating dominant in the whole parameter range up to ignition. The power absorption profile is well centered in the plasma core in both cases. A point demanding further attention is wave absorption by fusion alpha-particles. The validation of the models developed will have to await the results of IC heating experiments in large tokamaks such as JET and JT-60.

A considerable modelling effort was also made to analyze LH heating to ignition. If an appropriate choice of the frequency is made, one can couple the power to the electrons during the entire heating phase or, alternatively, during most of the heating phase except when ignition is approached where ion heating sets in. This would allow driving, at the same time, a large fraction of the plasma current non-inductively. Feedback control of the spectrum is necessary to ensure power absorption in the plasma core. From the scaling of the upper density limit for electron heating and current drive, a choice of the frequency somewhat above 3 GHz is indicated if it cannot be ensured that the DT plasma has a hydrogen admixture of less than about a percent. The physics models used in this analysis were also applied to LH heating experiments in present devices and found to provide a satisfactory description of the observations.

5.4 Configuration and Maintenance

The INTOR design concept is based upon horizontal access for removal of torus segments. This maintenance concept simplifies assembly/disassembly but complicates the poloidal field coil

design by excluding coils from the near-midplane region. Comparison of the horizontal maintenance approach with an approach in which torus segments were removed vertically was a major task of the INTOR Workshop. Each party developed vertical access configurations for comparison with the INTOR reference design and other horizontal access configurations. Those configurations include purely vertical access, oblique access and a configuration with combined vertical and horizontal access. The major differences and impacts were examined considering the following features: torus segmentations and access port size, TF coil bore and supporting structure; active coils and passive structures; arrangements for plasma position control; bellows structure; reactor building size and shape; PF coil system; interfaces for heating and current drive devices; and maintenance procedures and equipment. Some differences were recognized, although there were some difficulties in developing quantitative criteria, except for the impact on the PF coil system. It was found that for low elongation plasmas, there are no clear-cut advantages for either maintenance approach with respect to stored energy and power supplies. For high elongation, locating the PF coils closer to the midplane of the machine can significantly reduce these PF system requirements. As a general conclusion, vertical access should be suitable for high plasma elongation and horizontal access for low elongation.

Access port size available for the vertical access configuration is smaller than that for horizontal access configuration, leading to greater segmentation requirements for the removable part of the torus structure. The higher segmentation permits a smaller weight of the replaceable module but, on the other hand, reduces tritium breeding efficiency and passive stabilization of the plasma.

For the vertical access configuration, the upper active control coils are segmented. The number of TF coils for the horizontal access approach should not exceed 12 in the case of providing simple straight line maintenance procedure. For the vertical access case, this number can be increased, leading to a smaller TF coil bore size.

In both cases, heating and current drive devices will be installed horizontally at the midplane. Removal of first wall/blanket segments for the vertical access configuration requires overhead maintenance handling equipment. This equipment must be capable of providing for translation, lifting and in some cases the tilting of components. Either the reactor hall must be sized to accommodate the peripheral reactor equipment for maintenance and initial assembly or a reactor hall upper deck with a tight intermediate containment where no personal access is permitted must be sized to accommodate maintenance equipment and initial assembly. The peripherals are housed in tight isolated cells around the reactor where personnel access is permitted. This latter concept allows a smaller building span. Both approaches seem to be feasible.

The effectiveness and engineering feasibility of ferromagnetic inserts for ripple reduction were investigated with the following conclusions. The ferromagnetic inserts can reduce TF ripple effectively at the plasma edge and across the plasma cross-section, permitting a reduction in TF coil size and number. Typical calculations show that the use of the inserts in the reference INTOR machine gives the possibility to reduce the TF coil size by 0.5m or TF coil number from 12 to 10. No major difficulties in the integration of ripple inserts into the reactor structure are envisaged. Forces acting on the inserts (10-25MN) seem to be manageable. The effect of the inserts upon the electromagnetics is also manageable. With respect to the INTOR Phase One design concept, two improvements in divertor and first-wall replacement procedures are suggested: 1) the baking time required before maintenance operation is approximately 2 weeks and is the biggest contribution to the down-time. Thus, vacuum or inert gas coverage should be maintained during maintenance to avoid the baking; and ii) damaged components such as first wall armor tiles should be considered to be replaceable in-situ instead of removing massive modular structures. For the divertor plate cassette replacement, it is recommended to use a transfer cask under vacuum/inert gas. For the first wall an in-situ replacement of armor tiles, using a remote handling manipulator placed in an "ante-chamber", is recommended. In order to realize in-situ replacement of first will armor tiles, easily replaceable tile attachments are necessary as well as the development of in-vessel inspection systems.

Tritium contamination in the reactor hall by outgassing of the reactor inner surface and additional contamination from the dust created by sputtering/disruption are crucial when the torus is opened for maintenance. Several solutions are proposed to limit or reduce the risk of contamination in the reactor hall. To cool the transfer casks with a chiller system in order to minimize the tritium outgassing can prevent any spread of contamination inside the reactor hall by containing the dust inside the casks. Another possibility is to use plastic shroud enclosures in order to contain withdrawn pieces to stop the spread of tritium contaminations. The only concern here deals with the metallic dust after heat. For small openings, the use of a separate aspiration system for air induction into the torus is also considered as a means to prevent the spread of tritium and dust into the reactor hall. All the proposed methods are of common use in

the tritium technology area and are expected to be applicable to later fusion reactors.

A large containment area located in an upper space of the design, called tight intermediate containment, is proposed for the vertical access concept. To this area personnel access is excluded.

5.5 Blanket and First-Wall

5.5.1 Disruption Analysis

Since disruptions have a major influence on the design and performance of the first wall and divertor, special emphasis has been placed on analyses and experimental observation of these effects. A parametric erosion analysis has been performed for the first wall and divertor materials for a range of conditions, and the predicted erosion for the postulated disruption scenario has been determined. Also, the effect of disruptions on the fatigue life of a steel wall has been evaluated and experimental observations on simulated disruptions were reviewed.

The parametric disruption analysis considered disruption times of 0.1 to 20 ms, deposited energy densities of ~ 100 - 1000 J/cm^2 on stainless steel plus graphite, and on tungsten. The extent of vaporization, melt layer thickness for the metals, and effects of vapor shielding were determined. Based on the postulated disruption scenario, in which the thermal quench was assumed to occur in ~0.1 ms with most of the energy going to the tungsten divertor plate, the predicted lifetime erosion of the tungsten was ~ 17 mm and that of the steel wall was ~ 1.7 mm. For this case, the melt layers were assumed not to erode in the short disruption times.

The effects of these disruptions on the fatigue life of a steel wall is a major concern that has not been completely resolved. Analyses indicate that surface cracking of the wall will occur as a result of severe disruptions; however, propagation of the crack will not occur, and hence, the normal fatigue life will not be significantly degraded. Other analyses indicate that stresses created at the surface by the disruptions are sufficient to severely degrade the fatigue life of a steel wall by approximately one order of magnitude. It is apparent that the ASME and RCC-MR design codes are not applicable to these conditions.

Experimental observations of simulated disruption effects with electron beams on steel indicate more melting and less vaporization then predicted by the analytical codes. The difference is attributed primarily to momentum transfer from the electron beam, which is not accounted for in the codes. It remains to be determined whether significant momentum transfer is expected in an actual plasma disruption.

5.5.2 First Wall Designs

The first wall design effort was concentrated on evaluation of critical issues associated with a bare steel wall and a graphite protected wall. The following general conclusions are suggested:

i. It is preferable to use in INTOR a non-reactor-relevant concept with low pressure/low temperature water as a coolant for better reliability;

ii. Austenitic stainless steel is the only reasonable structural material for the low temperature first wall of INTOR;

iii. There are two possible design concepts for the INTOR first wall:
   - Bare stainless steel first wall design concept, and
   - Protected first wall design concept;

iv. The choice of the extension of the protection of the first wall steel structure by low Z tiles for INTOR is based on the uncertainties regarding the effects of disruptions on the fatigue life of a steel wall and different interpretations and/or application of design codes to allowable lifetime and/or stresses.

A bare stainless steel first wall is currently the reference for the INTOR design [4]. The primary advantage of the bare steel wall includes design simplicity and a well-established data

base. Key issues identified for further study include: (1) effectiveness of vapor shield during disruptions, (2) effects of disruptions on fatigue life, (3) melt layer stability during disruptions, and (4) advantages and disadvantages of cold-worked versus solution annealed stainless steel, particularly as effected by welding/joining. For bare first wall designs with a fatigue lifetime of 2×10^5 cycles, the allowed park nominal heat fluxes are assessed to be 0.2-0.4 MW/m^2 depending on: i) The initial minimum first wall thickness of 3-6mm, where especially for the smaller value stable disruption melt layers are required; ii) the evaluation of disruption effects via different fracture mechanics methods; and iii) the coolant channel geometry (coolant direction, double containment, design pressure).

Two types of first wall protection are proposed: i) local protection by "guard" or "sub" limiters for rapid replacement and ii) extensive coverage by protection tiles with radiation cooling.

With radiation cooled graphite tiles on austenitic steel first wall structures, high nominal peak heat fluxes of 0.5 MW/m^2 at 2×10^5 cycles can be achieved, limited by both steel fatigue and graphite temperatures/erosion, considering a thin steel wall thickness, reradiation, and residual tile heating.

There are, however, also several critical issues which will limit the use of graphite tiles as first wall protection for the next generation tokamaks: i) irradiation damage is expected to lead to severe anisotropic swelling (a lifetime < 1 MW y/m^2 is predicted for nuclear grade graphite and even less for c/c composites); ii) a general concern for the integrity of the tile attachment, both for bonded and mechanical solutions; iii) the total erosion due to physical and chemical sputtering has been estimated for graphite (assuming about 1% oxygen) rising to about 15mm/year at 1800°C; iv) the form and location of redeposited material is also a key issue. (Graphite may redeposit as amorphous carbon with different properties); and v) outgassing properties of graphite will present major vacuum problems during startup.

5.5.3 Divertor Collector Plate Design Concept

The emphasis of the divertor design effort has been on detailed analysis of the collector plate reference design, the study of alternative solid state collectors, on new plasma disruption scenarios, the development of innovations on liquid metal collector and on self-pumping divertor applications.

The reference concept for the divertor plate remains an actively cooled plate consisting of tungsten tiles on the plasma side bonded to a cooled copper or copper -based alloy heat sink. By use of buffer materials between the W-armor and the Cu-alloy heat sink, it is possible to obtain a structure where ASME criteria are satisfied and the allowable number of cycles exceeds 2×10^4 under thermal loads of ~5 MW/m^2. The key issue here is the choice of optimal fabrication method of tungsten-copper bonds.

Analysis of the anticipated disruption scenario shows that significant divertor plate thermal erosion will take place. For tungsten, for example, evaporation constitutes about 1.5 mm at 2×10^4 cycles of tokamak operation.

Two innovations on applications of helium burial in divertor and on liquid metal divertor collector plate were developed. The first innovation allows a reduction in the required pump capacity, tritium processing and fueling and an increase in the helium removal efficiency and space for nuclear and tritium breeding modules. The second innovation helps solve the problem of erosion and thermal fatigue and provides for divertor plate operation without excessive damage for the total INTOR lifetime.

5.5.4 Tritium-breeding blanket

A wide range of blanket concepts have been studied with the primary objective of investigating the feasibility of achieving tritium self-sufficiency without compromising the overall reliability of the machine and with modest R&D requirements. The evaluations led to the following main conclusions for the concepts considered:

5.4.1 Ceramic breeder with water or helium coolant

Water cooling at low temperature is preferred, primarily because of the reliability goal and the achievable compact design. For this case the ceramic breeder (Li_2O, Li_4SiO_4, $LiAlO_2$) should be arranged in the form of hot pressed plates or of pebble beds surrounding the coolant tubes and contained in modules with austenitic stainless steel as structure material. Beryllium or possibly lead are required in significant quantities (up to 80% of blanket volume) as neutron multipliers in order to achieve adequate T-breeding. T-extraction is performed via a low pressure helium purge flow, possibly doped with hydrogen for reduced T-inventory. The main critical issues concern the integrity of the pebble bed, consequences of coolant tube leaks and ruptures, compatibility problems with the multiplier and stress corrosion by the water.

5.5.4.2 Lithium-lead eutectic breeder, self-cooled or water-cooled

Water cooling is preferred, mainly because of the R&D requirements, which would be significantly higher due to primarily MHD effects in the self-cooled concept. For water cooling at about 10 MPa, the coolant tubes would be submerged in the liquid breeder, which would be contained in long poloidal pressure tube modules. Beryllium may be needed in limited quantities (20% of blanket volume) for achieving T-self-sufficiency on the outboard side only. For T-extraction, the liquid breeder is slowly circulated at a rate of up to 10 times per day. The ongoing R&D effort has led to reasonable confidence on the basic feasibility of the concept in most of the previously identified critical issues such as corrosion of the structure material by the eutectic or safety consequences of a coolant tube rupture. More development work is required on the T-extraction systems and on aqueous stress corrosion of austenitic steel.

5.5.4.3 Aqueous lithium salt blanket

First conceptual studies indicated the possibility of using Li-salt (LiOH or $LiNO_3$) dissolved in low temperature coolant water as a simple blanket solution, by which an initial shielding blanket could be later transferred into a T-breeding blanket. In order to achieve T-self-sufficiency, however, at least 65% of the blanket volume would have to be beryllium. The T-extraction from the coolant water is a well-established technology based on CANDU experience. However, the large processing capacity required in order to maintain acceptable T-levels in the coolant is estimated to require up to $50M for the associated T-extraction system of INTOR. The major technical feasibility issue relates to stress corrosion of the austenitic steel structure, including irradiation effects.

It is concluded that there are several blanket concepts which have the prospects of achieving tritium self-sufficiency without major impact on the machine reliability and with modest R&D effort.

6. ENGINEERING TESTING

6.1 Testing

The INTOR operational requirements for engineering testing were evaluated. Three different aspects of testing were considered: (1) fluence requirements for structural materials radiation damage; (2) short-term operational requirements for blanket testing; and (3) long-term operational requirements for establishing component reliability.

6.1.1 Fluence requirements for structural materials radiation damage tests

The impact of materials irradiation testing in INTOR on reducing the uncertainty in the structural materials data base for DEMO design was evaluated for stainless steel and for an advanced alloy, typified by a vanadium alloy. The type of information obtainable is shown in Table 6.1. The value of information from INTOR in reducing uncertainty depends upon the availability of a high-fluence neutron source such as FMIT. The impact of different fluence levels in INTOR upon the risk associated with the performance of the structural materials in DEMO is summarized in Table 6.2.

Testing of structural materials in INTOR at a higher neutron fluence will provide a better data base for constructing the DEMO. The evaluation of the necessary neutron

fluence for such testing indicates that a fluence of 2-3 MW.y.m^{-2} will probably be the minimum.

TABLE 6.1 STRUCTURAL MATERIAL FLUENCE EFFECTS [4]

Maximum testing fluence	Benefits of testing
Stainless steel	
0-1 MW·y·m^{-2}	Little useful information
1-3 MW·y·m^{-2}	Confirmation of low-fluence effects predicted with other sources particularly for tensile properties
3-6 MW·y·m^{-2}	Model verification from observations of microstructure preceding long-term behaviour changes
Above 6 MW·y·m^{-2}	Confirmation of performance (particularly swelling) near end of life
Vanadium alloys	
0-2 MW·y·m^{-2}	Tensile properties still changing
2-3 MW·y·m^{-2}	No swelling observed, important role within an alloy development program

TABLE 6.2 EFFECT OF INTOR FLUENCE ON STRUCTURAL MATERIALS RISK[a] IN DEMO[b] [4]

High-fluence fusion test facility	with "FMIT"	without "FMIT"
Stainless steel	3 MW·y·m^{-2} low risk	3 MW·y·m^{-2} low-medium risk
Vanadium alloys	3 MW·y·m^{-2} medium risk	3 to 6 MW·y·m^{-2} high-medium risk

[a] Medium risk defined as good chance that material will exceed 50% of design life, but significant chance that 100% of design life may not be met.
[b] Assuming fission reactor test program.

6.1.2 Long-term operation component reliability

The benefit of long-term component operation was quantified in terms of the number of hours of operation that would be required to assure an 80% confidence level in predicting the mean-time-between-failure (MTBF) for that component in the DEMO. Anticipated MTBF were established for INTOR and DEMO components, then the test time required in INTOR was determined from reliability analysis, taking into account the number of such components present. The major benefits that would result from long-term component operation in INTOR are: 1) definition of failure modes; 2) determination of failure rate and distribution; 3) determination of failure recovery time; and 4) identification of design improvements. There is a substantial incentive to achieve at least 2-3 MW·y·m^{-2} fluence for component reliability testing.

6.1.3 Blanket testing requirements

The requirements upon INTOR operation were assessed for several different types of blanket tests -- neutronics, tritium recovery, materials compatibility, heat recovery and breeder lifetime. In general, the neutronics tests do not impose significant requirements upon operation. The minimum continuous operating times were estimated for solid breeder tritium recovery tests for heat recovery tests to be about 65 h (about 950 continuous cycles) and about 1250 s (about 5 continuous cycles), respectively. Solid breeder microstructural and thermophysical property changes with radiation are estimated to saturate at about 0.2 $MW \cdot y \cdot m^{-1}$ neutron fluence. Since INTOR will have a lower volumetric nuclear heating rate than DEMO, it will be necessary to simulate the DEMO thermal-mechanical conditions in specially designed test modules in order to obtain relevant information on tritium release, heat transfer and materials compatibility.

6.2 Cost-risk-benefit assessment of fluence objective

Neutron fluence is a convenient characterization of the INTOR performance objective. Many radiation-damage-related testing capabilities are directly related to fluence. In addition, the accumulation of long-term component operation reliability data can be correlated to the fluence, for a fixed neutron wall load.

A cost-risk-benefit comparison of alternatives with different fluence objectives was performed. A risk-benefit figure-of-merit was defined by taking into account the importance of the information from INTOR to the design basis for the DEMO and the probability that a given INTOR alternative could provide the information required of INTOR. This probability comprised two factors: 1) the design objective (e.g. fluence goal) of the alternative; and 2) the risk associated with achieving the design objective. The figure-of-merit was normalized so that the Phase One INTOR concept, which provided all the information necessary to supplement that from the base program and complementary facilities to complete the DEMO design basis, would have a value of unity. This figure-of-merit and the costs are shown for four different alternative, ranging in fluence capability from 0.2 to 10.0 $MW \cdot y \cdot m^{-2}$, in Table 6.3. There is relatively little difference in the base capital costs, but a rather large difference in operating costs, which results from the different operational lifetimes and tritium costs. The first case only produced about 40% of the information (in an importance-weighted sense) that is required of INTOR, while the other cases produce about 70-80% of the required information. The cost of producing the missing information elsewhere, or, alternatively, the risk of designing the DEMO without it, has not been factored into these numbers. The additional cost that might be associated with achieving the higher availability that would be necessary with the higher fluence cases has also not been factored into the capital costs.

On the basis of the cost-risk-benefit analysis, the low-fluence alternative was rejected because it produces about 40% of the information (importance-weighted) required from INTOR, and the high-fluence alternative was rejected because of the high risk associated with the design. There is no clear preference between the two intermediate fluence alternatives.

The evaluation of testing requirements in Section 1 indicated a strong incentive to achieve about (2-3) $MW \cdot y \cdot m^{-2}$ for structural materials properties (tensile, microstructural change) data and for component reliability data. While there was incentive to achieve higher fluence, this incentive was not so compelling.

It was concluded that INTOR should be designed to achieve a neutron fluence of $\phi \geq 3 MW \cdot y \cdot m^{-2}$ with a high probability of success. The design should allow for operation to achieve higher fluences, but with a lower probability of success.

6.3 Machine Operation and Test Program

A preliminary operation and test plan has been developed to provide insight into the design and operational requirements that must be imposed on INTOR. This plan has been developed using judgement as to where INTOR fits within an international fusion development plan, as discussed in Section 2, and taking into account the complementary roles that will be played by other plasma physics experiments and technology testing facilities. A summary of the types of testing included in the test plan is given in Table 6.4.

Some tests will require nearly continuous operation for some period of time. In particular, some of the tritium recovery tests will require a 70% duty cycle and continuous operation for one

week to one month in order to reach equilibrium conditions. On the other hand, thermal-hydraulics testing will require only a 50% duty cycle and continuous operation for approximately one hour. Achievement of the required fluence for reliability testing of blanket modules and other components will require machine availability of about 30%

TABLE 6.4 TYPES OF TESTING IN INTOR [3]

PLASMA PHYSICS
Vacuum vessel conditioning
Assisted start-up
Long-pulse ignition experiments
Performance optimization
Non-inductive current drive
PLASMA ENGINEERING
Impurity control and exhaust technology
RF heating technology
Burn control technology
Continuous burn methods technology
BLANKET AND ENGINEERING TESTS
Prototype module
Tritium extraction
Critical element
BULK MATERIALS
Irradiation effects upon properties of candidate structural materials, insulators, high-heat flux materials, breeders and neutron multipliers
SURFACE EFFECTS
Retention/re-emission characteristics
Plasma impurity release
Surface erosion/re-deposition
Surface microstructural changes
REACTOR MATERIAL AND COMPONENT SURVEILLANCE
Engineering performance of systems
Failure modes and rates
Maintenance experience
Reliability data
NUCLEAR TESTS
Tritium breeding ratio
Nuclear reaction rates
Volumetric nuclear heating
Neutron and gamma-ray fluxes and spectra
ELECTRICITY GENERATION
Early power generation - end of Stage II
Prototype DEMO blanket -- end of Stage III

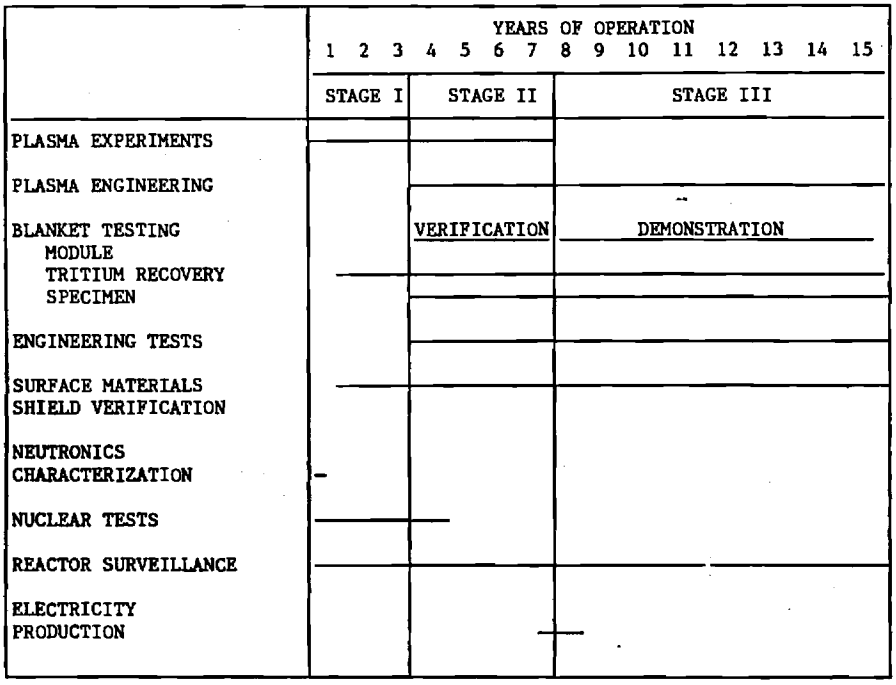

FIG. 6.1 INTOR test schedule [3]

The projected test schedule is shown in Fig. 6.1 The figure assumes that INTOR is designed to achieve 3 $MW \cdot y \cdot m^{-2}$ with high probability of success. Achieving higher fluences (6.6 $MW \cdot y \cdot m^{-1}$) is probable and is planned for. The test schedule will have to remain flexible to accommodate any changes in the device operating lifetime, which will be decided during the early phase of operation. As indicated, plasma physics testing will dominate Stage I operation. Stage II testing will consist primarily of plasma engineering and blanket engineering tests and other tests where frequent change-out is required. A minimum time of one month between scheduled reactor shutdowns has been established to permit test change-out without unduly affecting reactor availability. Stage III testing will be devoted to longer-duration tests which do not require frequent reactor shut-down.

7. INNOVATIONS

An INTOR-related IAEA Specialists Meeting on Tokamak Concept Innovations was held in January 1986. The purpose of this meeting was to identify innovations that would significantly improve the prospects of tokamak development leading to an attractive end-product -- a viable tokamak fusion reactor. Emphasis was placed upon innovations that would lead to substantial improvements in a tokamak reactor, even if they involved a radical departure from present thinking. For the most part, such innovations are not yet supported by the existing database. Innovations were contributed to the Meeting in nine categories, as indicated in Table 7.1. Fusion scientists and

TABLE 7.1 INNOVATIONS [7]

Topic	EC	Japan	USA	USSR	Total
Impurity control	5	3	3	4	15
Beta and confinement enhancement	0	6	4	3	13
Heating and current drive	2	6	2	2	12
Advanced magnetics	2	3	1	3	9
Plasma engineering	1	1	2	6	10
Configuration and maintenance	3	2	4	3	12
Advanced FW/B/S	12	5	1	8	26
Advanced materials	4	5	3	1	13
Compact reactors	1	1	4	4	10
	30	32	24	34	120

engineers from Europe (15), Japan (12), the USA (9) and the USSR (7) participated in the Meeting.

The innovations contributed to the Meeting were evaluated on the basis of substance and feasibility and those innovations which had a high priority for further consideration were identified. Factors considered in the evaluation of each innovation were:

1) how substantial an improvement would it lead to;

2) what is the feasibility of its being successfully developed;

3) what is the impact on other tokamak components; and

4) what further steps are required to evaluate its feasibility.

To provide a focus and a common basis for the evaluations, a quantitative rating system was adopted as follows:

A) SUBSTANCE (Assuming that the innovation can be successfully developed, how substantial an improvement would it lead to in a commercial tokamak reactor?)

1 - substantial improvement

2 - moderate, but worthwhile, improvement

3 - questionable net improvement

B) FEASIBILITY (Is it feasible to successfully develop the innovation for implementation on a

reasonable time-scale?)

1 - feasible to develop for implementation on the INTOR time-scale (~1990-1995)

2 - feasible to develop for implementation on commercial reactor time-scale (~ 2000 - 2050)

3 - questionable feasibility for successful development

Those innovations which have a high priority for further consideration were identified on the basis of two criteria:

1) it would lead to a substantial improvement in the tokamak as a reactor concept; and

2) it is feasible that it could be successfully developed within a reasonable time.

The evaluation of the contributed innovations is summarized in Ref. [7]. It is significant that 48 innovations were identified having a high priority for further consideration, and 25 of these were judged to be feasible on the time-scale of a next-generation experimental reactor (i.e. the INTOR time-scale). This evaluation indicates that a substantial improvement in the tokamak reactor concept, relative to our present perceptions, may be realized in the future. Moreover, there appears to be a possibility to realize a significant improvement in the INTOR concept.

The most promising innovations identified during the Meeting fell into a relatively small number of areas. A number of high priority innovations were identified which could lead to a simpler impurity control system, relative to the poloidal divertor. Among these were neutral beam driven impurity flow reversal and edge ergodicity to achieve a cool edge region that would make a pumped limiter feasible, and helium removal via burial in in-situ deposited surfaces.

Several high priority innovations were identified which could lead to enhancement of beta. Among these were accessing the second stability regime with or without plasma indentation, suppression of sawteeth oscillations and current profile control.

Fast wave and neutral beam current drive were identified as promising alternative to lower hybrid current drive for the achievement of steady state operation. Ion Bernstein wave heating was identified as a promising alternative to ICRF.

The use of sub-MeV, negative ion based neutral beams for impurity control, heating, current drive and burn control was given a high priority for further consideration. A reactor that used a single NBI system for those four purposes could be simpler than one with ICRF heating, LHR current drive, a poloidal divertor for impurity control and some other system for burn control.

Advances in superconducting magnet technology were felt to offer substantial promise for improving the tokamak reactor concept. Higher current density, higher field conductors would lead to more compact coils. Forced flow, supercooled helium conductors and fibre reinforced non-metallic structures were identified as promising areas for technology advances.

Many promising concepts for breeding blankets were identified. Among these, a self-cooled metal blanket and a high temperature He-cooled blanket appear to offer substantial improvement relative to previous blanket concepts. The development of ferritic/martensitic alloys in the near term and vanadium alloys over the longer term would lead to extended blanket lifetimes. The protection of components exposed directly to the plasma by means of liquid metal droplets promises to extend the lifetime of the most limiting components (i.e. divertor plate or limiter).

Several possibilities for achieving a more compact tokamak reactor were examined. It was concluded that improving upon the present line of moderate aspect ratio, moderate elongation, superconducting reactor concepts offered more promise than alternatives based upon very small aspect ratio, extreme elongation, or normal coils.

A proposal to locate MHD channels in the blanket introduced the possibility of taking advantage of the unique characteristics of fusion to reduce the cost of the entire energy conversion system.

The results of this Meeting led to three significant conclusions: there is a large potential for substantial improvement in the tokamak reactor concept, relative to previous perceptions; improving upon the current mainline tokamak configuration is more promising than pursuing extreme configurations, and there is a possibility of utilizing the unique features of the fusion energy source to improve the overall reactor plus balance-of-plant system. A detailed report of the Meeting [7] has been published.

The INTOR Workshop reviewed the results of the Specialist's Meeting and identified 37 items which were considered possibly to be feasible to develop and implement on the INTOR time scale. These items were further evaluated during the course of the Phase Two A Part 3 Workshop of these items, 12 were considered by the Workshop to be sufficiently promising and well-established to consider implementing within the early-to-mid 1990 time frame.

8. ANALYSIS OF INTOR-LIKE DESIGNS

8.1 Introduction

In November 1986, the IFRC recommended that the INTOR Workshop conduct, during 1987, critical analyses of existing INTOR-like designs, with the aim of preparing a useful information base for future design work for the Engineering Test Reactor (subsequently named ITER). As a first step, members of the INTOR, FER (Japan), NET (EC), OTR (USSR) and TIBER (USA) design teams* met together in an IAEA Specialist's Meeting (March, 1987) to document in a common format, discuss and compare the programmatic and technical objectives, the engineering and physics design constraints (i.e., physical limitations such as stress limits, beta limits), the main features which "drive" the design concept (i.e., choices made by the designers such as to incorporate non-inductive current drive or a horizontal maintenance and assembly scheme), and the design specifications (e.g., major parameters, choice of materials, choice of heating method) for the five designs.

The five designs are characterized by their gross parameters as listed in Table 8.1.

TABLE 8.1 MAJOR PARAMETERS OF INTOR-LIKE NATIONAL DESIGNS [5]

PARAMETER	INTOR[x]	NET	FER	TIBER	OTR
Major radius (m)	4.90	5.18	4.42	3.00	6.30
Minor radius (m)	1.2	1.35	1.25	0.83	1.50
Fusion power (MW)	585	650	406	314	500
Plasma current (MA)	8.0	10.8	8.8	10.0	8.0
Average beta (%)	4.9	5.6	5.3	6.0	3.2
Safety factor, q_I	1.8	2.1	1.8	2.2	2.1
Plasma heating method/power (MW)	ICRH/50	ICRH/50	ICRH/50 LH/20	LH/10 NBI/40	ICRH/50
No. TF coils	12	16	12	16	12
Maximum field TFC (T)	11	11.4	12	12	11.7
Volt seconds	112	181	50	58	210
Neutron wall load peak/avg. (MW/m^2)	1.6/1.3	1.5/1.0	1.5/1.0	1.6/1.0	1.05/0.
Tritium inventory (kg)	3.1-4.6	2	2	TBD	3.5-5.0
Test FW area (m^2)	12	40	9	19	

x) as of Phase Two A Part 2

*Most principal members of the national design teams were also present or past INTOR members.

The programmatic and technical objectives were discussed by the design team leaders present at the meeting. The design constraints, design "drivers" and design specifications were compiled and compared by design team members. The results of their work are summarized in Sections 8.2-8.5.

On the basis of this work, the INTOR Workshop undertook further analyses of the issues raised in the Specialists' Meeting and also undertook a rather extensive systems analysis aimed at arriving at a unique description of the various designs and at extracting the most essential design drivers. This latter work is summarized in Section 8.6. A cost comparison among the 5 designs is given in Section 8.7.

8.2 Objectives

There is virtual agreement on broad objectives among the FER (Japan), INTOR (IAEA), NET (EC), OTR (USSR) and TIBER (USA) designs. All of the designs are intended to be the maximum reasonable physics step beyond the present generation of large tokamaks and to demonstrate achievement of the type of physics performance that is required for subsequent (demonstration) reactors. All are intended to integrate essential reactor-relevant technologies into a single system and to provide component and nuclear testing capabilities. All are predicated upon a start of construction date of about 1993.

Some differences emerge, however, when the more detailed objectives are compared. FER and OTR emphasize the avoidance of new technologies not essential to the physics mission in order to minimize risk, whereas TIBER emphasizes the utilization of new technologies to minimize cost. INTOR and NET fall somewhere in between these two positions.

There are also differences in the engineering testing objectives among the five designs. FER emphasizes limiting the testing for technology development and the nuclear testing in order to achieve reasonable cost and technical risk, recommending a neutron fluence objective of only 0.3 $MW \cdot y/m^2$. INTOR, OTR and TIBER emphasize testing for component development and nuclear testing and recommend a neutron fluence objective of 3-5 $MW \cdot y/m^2$. The NET position on testing and fluence is intermediate. The tritium breeding objectives vary accordingly from no tritium breeding, except in test modules, in FER to tritium self-sufficiency in OTR and TIBER.

The physics mode of operation also differs. TIBER emphasizes non-inductive current drive to achieve "steady-state" operation, while FER, INTOR and NET emphasize ignited operation, and OTR emphasizes high-Q operation.

In addition, there is also a difference in the objective of the design studies themselves, as distinct from the objectives of the devices, which has caused differences in the designs. The TIBER design activity had as an objective to study the extent to which a compact design could be achieved by making aggressive assumptions about the development and incorporation of new technologies which are still to be developed, whereas the other designs were based upon less demanding technology development requirements.

8.3 Physics Constraints

The physics assumptions and constraints for each of the four national ETR designs and INTOR are quite similar (Table 8.2). The differences in the designs are mainly due to the choice and emphasis of different features (see Section 8.4) and the use of different engineering constraints (see section 8.3). On the whole, the national designs tended to adopt more conservative physics assumptions than INTOR, especially with regard to β and q_{edge}. All of the designs rely upon H-mode confinement and have incorporated an open poloidal divertor for this. In addition they rely upon current scaling for confinement and beta, with the result that the specified currents are in the 10 MA range. The plasma is elongated to achieve this current. The elongations vary from 1.5 to 2.4. All of the designs have adequate margin for ignition with ASDEX - H scaling, but none of the designs can ignite with most L-mode scalings.

Table 8.2

Physics Constraints [5]

	INTOR	NET	FER	TIBER	OTR
I (MA)	8	10.8	8.74	10	8
elongation	1.6	2.05	1.7	2.4	1.5 1.7
τ_E reg. (sec.)	1.4	1.9	1.7	.44	
τ_E (ASDEX-H)/τ_E, required	2.9	3.0	2.3	6.8	3
n ($10^{20}/m^3$)	1.6	1.7	1.14	1.06	1.7
Murakami* (10^{19}/ T.m^2) parameter	19	23	15	8	25
Beta Required (%)	4.9	5.6	5.3	6	3.2
Troyon coefficient (%)	4	3.5	3.5	2.8	3.5
Impurity Control divertor **	SN	DN	SN	DN	SN
Pulse length (sec)	150	350	800	55	600
Heating method	ICRF	TBD	ICRF (LH ramp up	LH + NBI	ICRH

*Using line - averaged density.
**DN = double null; SN = single null.

All of the designs rely upon densities for ignited operation that are at the high end of the present tokamak data base. The designs utilizing current drive with subignited operation can afford more conservative assumptions with respect to the density limit. The Murakami parameters range from 15 to 25 for ignited operation and ~ 8 for Q = 5 operation.

All of the designs use a "Troyon" type of scaling for the beta limit, although the choice of the Troyon coefficient is somewhat different in each design. As stated before, this type of scaling leads each design to emphasize increased current as a way to maximize beta.

The edge safety factor varies among the designs, but the cylindrical safety factors are very close for all of them.

All of the designs rely upon the operation of an open, high recycling divertor to provide

power and particle exhaust. Advanced fueling techniques, including high velocity pellets and other schemes, have been considered by all of them.

A toroidal field ripple in the range of 0.75 to 1.2% is anticipated to be adequate for fast alpha particle confinement, although major uncertainties remain.

The physics assumptions for current drive and heating are quite similar among the designs. For those that do not rely upon current drive, ICRF is generally the heating method adopted. Where non-inductive current ramp-up and transformer recharge is used, lower hybrid wave heating is the method of choice. Those designs incorporating steady-state current drive rely upon 400-500 keV neutral beams for central current drive and lower hybrid waves for current drive at the edge. The penetration of lower hybrid waves is felt to be inadequate for high density operation.

8.4 Engineering Design Constraints

Engineering constraints are those parameters and limits used in a design that are derived primarily from physical laws of nature and over which the designer has limited choice. For example, these include such elements as the radiation damage limits, heat load limits, etc. In addition, choices made in one system or aspect of a design can then pose as a design limit to be satisfied by other systems or aspects of the design. For example, the decision to use a double null, highly elongated plasma poses limits on the mechanical configuration which must be satisfied. The major engineering design constraints are given in Table 8.3

Table 8.3

MAJOR ENGINEERING DESIGN CONSTRAINTS [5]

	INTOR	NET	FER	TIBER	OTR
Field ripple (edge) (%)	1.2	1.5	0.75	0.8	1.0
Impurity control*	SN	DN/SN	SN	DN	SN
Plasma elongation	1.6	2.2/1.7	1.7	2.4	1.5
Max. rad. to TEC insulator (rad)	10^9	5×10^8	3×10^9	10^{11}	10^9
TFC Stress allowable	ASME**	600 MPa	600 MPa	600 MPa	600 MPa
Max. FW heat flux (MW/m^2)	0.4	0.4	0.4	0.3	0.4
FW Stress allowable	ASME	RCC-MR	ASME	ASME	200 MPa
Div. Peak heat flux (MW/m^2)	5	5	2	3	5

*SN = single-null divertor, DN = double-null divertor
**ASME Code

In the engineering category, the design constraints were arranged in three groupings: mechanical and configuration; electromagnetics, heating and current drive technology; and nuclear.

8.4.1 Mechanical and Configuration

In the mechanical and configuration grouping, the major engineering design constraints and the range of choice used in the five designs are as follows:

Magnet configuration: Placement of all coils in a common cryostat or placement of magnets in a self-contained cryostat.

Method reacting magnet loads: Use of a bucking cylinder, wedging of the inner legs of the TF coils, or reacting the TF coils directly from the central solenoid.

Vacuum boundary: Use of a common boundary for the plasma and the magnets or the use of separate vacuum containment for each system.

Number of replaceable modules: This varies from 12 to 48 depending upon the overall device configuration.

Component replaceability: Most designs assume many of the components to be designed to last the life of the device with no plan for replaceability; other designs make no such assumption.

Tritium breeding: This varies from no tritium breeding (other than in test modules) to full tritium self-sufficiency.

Maintenance approach: Two major approaches are considered; these are horizontal removal of torus components, or vertical removal of torus components.

Plasma configuration: The plasma configurations vary from modestly elongated (1.5) single-null divertor plasmas to highly elongated (2.4) double-null divertor plasmas.

Radial dimensions: The five designs vary dramatically in the overall plasma major radius, and naturally in the thickness of the components and space allocations comprising the major radius. For example, the allowance for plasma scrape-off varies from 9 to 30 cm, the total inboard blanket/shield thickness varies from 48 to 105 cm, the accumulated allowance for assembly gaps and spaces varies from 2 to 20 cm, and the thickness of the TF coil inner leg varies from 49 cm to 110 cm.

8.4.2. Electromagnetics, Heating and Current Drive Technology

In the area of electromagnetics and heating and current drive technology, the major differences are in the electromagnetics. All of the designs are using similar heating and current drive technologies.

In the TF coil system, there are a number of different engineering design constraints being used. These are primarily related to the environment perceived necessary for the desired performance from the superconductor. These include the total and peak nuclear heating levels. The total nuclear heating level varies from about 8 to 72 kW of nuclear heat deposition; the related peak nuclear heating levels vary from about 0.3 to 5 kW/m^3. Radiation protection requirements for the superconductor and the associated insulator also vary significantly; the radiation dose varies from about 5×10^8 to 10^{10} rads. Other significant variables are the conductor current values (from 16 to 35 kA), the average winding pack current density (from about 10 to 22 MA/m^2), the magnetic energy (from 4 to 45 GJ), and the maximum quench voltage to ground (from about 7 to 20 kV).

In the PF coil system, the dominant differences are related to the total volt-seconds the system must provide (from about 50 to 210 V-s). In addition, there are differences in the allowable maximum field rate of change, varying from about 0.5 to 3 T/s; differences in the OH current ramp time, from about 13 to 30 s; differences in the breakdown voltage being used, from 10 to 35 V; and finally, differences in the total magnetic stored energy, from about 4 to 11 GJ.

8.4.3 Nuclear

In the nuclear systems area, there are significant differences in the engineering design constraints in a number of aspects of the first wall and blanket, the divertor and in the shield area.

In the first wall and blanket systems, these differences are related to the target lifetime fluence values (which range from 0.3 to 5 MW·y/m^2), the allowable stresses in the structural material which are also tied to the number of lifetime cycles of operation, the tritium breeding requirement, the first wall protection assumptions, and finally the assumptions related to the disruption scenario.

In the divertor area, the differences are related to the incorporation of different concepts for the physics and technology phases, and to the differences in the disruption scenario.

In the shield area, the differences are related largely to the need to protect the magnets so are tied to the allowable fluence to the superconductor, to the allowable dose to the insulators, and to the nuclear heating limits. A second constraint relates to the desire to minimize the overall thickness of the inboard shield region so as to minimize the size and cost of the design.

8.5 Design Driving Features

The five INTOR-like designs differ in a number of significant features which tend to "drive" the characteristics of each design. Each of these design driving features represents an aspect of the design where the designer has a choice from among a number of options. These choices are made so as to be compatible with the overall mission and supporting programmatic and technical objectives established for each design, but are usually based on the designer's personal preferences.

The selection of each of the design driving features by each design team is also influenced by the judgement concerning a number of important considerations related to each national program. These include the perceived timing for the necessary development and construction of each device, The perceived understanding of the present scientific and technological data base and what advances can be made in the time period until the start of construction, and finally the maturity of the technology need to support each design and its stated mission.

Table 8.4 presents a comparison of the major design driving features for the five INTOR-like designs. Many of these features are related to the scientific desires and present

Table 8.4

MAJOR DESIGN-DRIVING FEATURES [5]

FEATURE	INTOR	NET	FER	TIBER	OTR
Operating mode	Ignited	Ignited	Q>20-30	Q>5	Q>5
Pulse length (s)	150	>200	800	CW	600
Current drive	induct.	induct.	hybrid	non-induct.	induct.
Fluence (MW·y/m^2)	3.0	0.8	0.3	3.0	5.0
Tritium breeding ratio	0.6	0.3	0.0	1.0	1.05
Plasma Heating Method	ICRH	TBD	TBD	NBI+LH	ICRH
Impurity control*	SN	DN/SN	SN	DN	SN
Access for maint.	horizontal	vertical	horizontal	horizontal	horizontal
Size of largest replaceable comp. (tons)	300	60	250	32	300
Max. Availability/period	25%/10 Yrs.	8%/11 Yrs.	7%/6 Yrs.	30%/12 Yrs.	50%/9 Yrs.

*SN = single-null divertor, DN = double-null divertor

understanding. These include the need to achieve ignition or not, the nature of the operating scenario (inductive or non-inductive current drive, or some hybrid combination), the pulse length, the degree of plasma shaping (elongation), the type of impurity control (single or double null divertor), the nature of startup, and the plasma heating method. The remaining major driving features result from operational and tehnological considerations such as whether to breed tritium or not (and how much), the fluence target and the nature of the desired nuclear testing, and finally the approach to maintenance of the internal torus components (horizontal or vertical access).

8.6 Systems Analysis

A systems analysis of the five INTOR-like designs was performed. This study was performed to evaluate and determine the specific impact produced in a given design of making a specific change in that design. Such a quantitative analysis of the impact of various changes provides valuable insights as to how different choices affect a given design.

Systems analysis methodology has progressed significantly during the last several years, under the aegis of the INTOR Workshop. The capability has been developed to represent a tokamak point design and much of the complexity of the interactive systems. Several of the computer codes which have been developed now incorporate numerical optimization methods that enable simultaneous change of many selected variables subject to specified constraints. Many of the codes involve interactive routines to converge to a unique solution for the specified input. Nomograph routines have also been developed to permit rapid parametric evaluation of design options for given assumptions. Finally, simplified systems of equations have been developed that represent present understanding of tokamak design physics and engineering dependencies. These systems provide the capability to quickly determine the impact of design variations.

A significant aspect of using systems analysis to study various tokamak designs is the ability to replicate various designs by a given methodology. A measure of the validity, and usefulness, of a given systems analysis methodology is the ability to reproduce the major features and performance of a variety of designs. One test of such ability is to use the systems analysis method to reproduce as accurately as possible the mechanical features, performance, physics, and engineering parameters of an existing design. Each systems analysis process requires some input to be provided, with the input, certain output is generated. The various national systems methodologies were applied to the task of replicating each of the five INTOR-like designs.

The results of calculations to replicate the four national designs and INTOR indicate good ability of the systems analysis methods to accurately represent the general, as well as many specific, features and parameters of the designs. This demonstrated ability provides confidence that parametric studies should provide meaningful indications of the impact of making a given change to a design.

Sensitivity calculations were performed by each participant to determine what changes are produced in the respective national design by making selected changes in the input. Some calculations were performed in which one aspect of the design was changed (a mechanical feature or dimension of a given component, a physics assumption or parameter, or an engineering assumption or parameter). The impact to the design resulting from this single change was determined.

The results of the individual sensitivity calculation allow determination of those items which have a high leverage impact on the overall design. Other items are determined to have considerably less impact. By performing a systematic assessment of which items have what impact on a design when changed, the highest leverage items can be identified. Once identified, this information can be factored into the detailed design process and thereby provide guidance to the designers who have to work out the design details.

From the results produced by the four participants, it was found that the items/parameters with greatest sensitivity include:

- Ignition margin, or Q
- Safety factor (q)
- Elongation
- Shield attenuation
- Z_{eff}, or Reactivity

Neutron Wall Load
Beta g-coefficient (Troyon factor)
Toroidal field coil stress

Although this general ordering was observed, individual sensitivities from the four studies do indicate appreciable scatter in the results. Various reasons account for this:

- Differences in the models (e.g. power balance level of detail, confinement time scaling law)
- Different figures-of-merit (cost, major radius)
- Differently constrained analysis (ignition margin, wall load, fusion power)

This latter point may be the most important, since each design team has evolved different sets of design constraints, including those items judged to be very important. Each design team has also derived some outputs which other design teams assume to be fixed inputs (e.g. the three items mentioned above).

In addition, each of the national designs has evolved to different levels, but in all instances the parameter space for each design has many constraints. These constraints considerably narrow the region of valid designs when one parameter is altered. Even for the same device, therefore, if different items are constrained, the parameter space over which the sensitivity analysis is conducted will be different, thereby contributing to different results.

Given these differences, it is nevertheless greatly encouraged that there is general agreement on the parameters whose sensitivity is large, and on those which are small.

There is an important distinction between sensitivity and overall design impact. To draw practical conclusions from the sensitivity studies, it is necessary to fold into the assessment the likely range of uncertainty of a parameter to determine the importance of changes in the parameter to the design. Large sensitivity parameters will have a large design impact on the design even if the range of variation is large or small. However, small sensitivity parameters could have a large design impact if the range of variation is large; this practical consideration should be recognized in the design process.

Calculations were also performed in which a collection of items was changed; for example, it was of interest to determine the impact of substituting at one time all physics-related assumptions made by one design into a second design. Similar interest exists relative to the impact of changing the engineering assumptions, or the general features of a given design.

Difference in the individual assumptions of the INTOR-like devices can be collectively grouped together into categories such as "physics" "engineering", or "features". The "physics" categories typically include terms such as beta and beta coefficient, safety factor, ignition margin, plasma temperature and density, edge ripple, plasma profile factors, etc. The "engineering" category typically includes dimensions of components (OH and TF coils, bucking cylinder, shield) stress levels, radiation levels, gaps, etc. The "features" category typically includes plasma configuration (elongation and triangularity), maintenance approach, fluence level, tritium breeding, single null or double null divertor, operating scenario, etc.

Each participant performed a limited number of these global effects analyses. These studies were performed in general in a manner similar to the individual sensitivity studies described above. For example, studies were performed in which calculational transitions were made from INTOR to FER (Japan), INTOR to TIBER (USA), INTOR to OTR (USSR), NET to TIBER (USA), and NET to FER (EC and Japan).

The results from these studies indicate that all parties have successfully demonstrated an ability to substitute global groupings of changes (all physics, engineering, features) and make the transformation from one design to another. The systems analysis codes have demonstrated the ability to differentiate the national designs by making global effect substitutions.

8.7 Cost Comparison

There is an ongoing, and important, interest in how the estimated capital cost is impacted by various aspects of a tokamak design. It is recognized that the various national designs are costed using the national procedure to account for engineering, fabrication, transportation, installation, and project management costs. These approaches are all different not only in units of currency but in how the various elements are treated in the costing process.

Recognizing these differences, a cost comparison was performed for each of the five INTOR-like designs to determine the relative ranking of capital cost as predicted by each national costing approach.

The relative capital costs were normalized to the INTOR cost estimate obtained by each delegation. The results indicated that the cost for INTOR and NET to be approximately the same, the cost of FER to be about 10% less than INTOR, the cost of TIBER to be about 35% less than INTOR, and the cost of OTR to be about 25% more than INTOR. In general, the spread among the participants in the calculated relative cost of a given design was about 5-15%.

These relative cost comparisons suggest good ability to reflect incremental differences between various designs. This ability is based on common assumptions about the designs relative to the various components and major systems. These cost comparisons must be interpreted with care, however, since various designs do make different assumptions about the timing of construction, the amount of supporting development and research required, and the aggressive or conservative posture regarding the maturity of the technology in the design. Factoring these considerations into the design could alter these cost comparisons dramatically.

9. TECHNICAL BENEFIT OF PARTITIONING INTOR COMPONENT DESIGN AND FABRICATION

INTOR combines and would test in a fusion reactor environment a number of advanced technologies such as the reactor vessel (torus, shield, blanket, superconducting magnets, plasma heating systems, tritium system), most of which occur in several modular compatible/equivalent units. Hence an international test reactor at that step of tokamak development could be the focus for gaining experience that would have to be developed in all participating countries anyway.

The study 'Technical Benefit of Partitioning INTOR Component Design and Fabrication' had to evaluate the technical feasibility of partitioning the detailed design and component production tasks so that all participants would fully benefit from the development of all advanced technologies.

The purpose of this study was to examine the implications of having different manufacturers fabricate a fraction of the components of a major system of INTOR; e.g. four different manufacturers each build three toroidal field coils. There are advantages and disadvantages for each of the participants to take part in such an international joint venture when compared to a purely national approach. These were assessed and compared with the technical benefit expected. The technical benefit is defined to be the development of industrial capability and experience needed for future fusion reactor technology, gained by partitioning all modular advanced technology components of INTOR

To provide a common basis for comparison among study results from the four INTOR delegations, a number of basic definitions and assumptions were agreed on. Specifically, two collaborative arrangements, called scenarios B and C, were selected for comparison against scenario A, in which four nations independently build INTOR-type experiments. The difference between scenarios B and C consists in splitting the advanced component design/production tasks between the participants in case B so that all participants design and produce a proportionate share of all high-technology components. In scenario C, all the components of a given type are provided by a single participant, with the component types being distributed among participants.

Expert estimates and opinions were collected as a basis for cost/schedule/benefit evaluation. Also, comments from other large international projects' representatives were sought in order to include existing experience.

TABLE 9.1 COST AND SCHEDULE IMPLICATIONS OF PARTITIONING IN DESIGN AND CONSTRUCTION OF INTOR [4]

Scenario	Four national INTORs		One international INTOR
	A	B splitting	C branching
Relative cost of one device	1	1.67	1.24
Relative cost per participant	1	0.42	0.32
Construction schedule delay (y)	0	~2.3	~1.2

The study led to the following conclusions:

(i) International partitioning in design and construction of INTOR with four participants - assuming four national R and D programs oriented towards that co-operation - has estimated cost and schedule implications as shown in Table 9.1. In scenario B/C one INTOR device is about 67%/24% more expensive than one national INTOR, whereas the cost share per participant reduces to about 42%/32%. The construction schedule delay was evaluated to be about 2.3y/1.2y.

(ii) In accordance with all cost estimates showing a substantial increase in engineering cost, which is considerably larger in scenario B than in scenario C, it is clear that the increased complexity of international partitioning will be reflected in correspondingly increased manpower.

(iii) The technical benefit of partitioning the INTOR design and construction (as defined), is gained to a large degree in the splitting approach (scenario B) for advanced technology components and to a rather low degree in the branching approach (scenario C).

(iv) A comparison of cost and degree of benefit for the scenarios shows that scenario B offers more benefit related to cost than scenario C.

(v) Scenario B is judged to be feasible, although more complex than a conventional national approach. It can be achieved if a strong central team is created, and if each participant will have sufficiently developed its own technology basis at the time of decision on INTOR construction.

10. CONCLUSIONS

The cumulative INTOR work to date has been a major factor in laying the ground-work for proceeding to the design of the next major experiment in the world tokamak program. The objectives and general characteristics of such an experiment have been identified. A preliminary conceptual design has been developed early in the INTOR process and used to identify critical technical issues and R and D requirements. The critical technical issues have been either resolved or elucidated by intensive studies. The analytical methods used in reactor design by the four parties have been further developed and compared for consistency. The national designs and the physical and technical constraints upon which they are based have been evaluated on a common basis. Based upon this work ways in which the INTOR design concept would be up-dated to provide a starting point for further design effort have been identified.

The INTOR Workshop process has paved the way for an intensified international design activity in another, more personal, way, also. Many of the scientists and engineers who would be involve in any such future effort have come to know, understand, trust and learn to work with each other through their participation in the INTOR Workshop. They understand, through their INTOR Workshop experience, that it is possible for men of different cultures and strongly held technical opinions to voluntarily work together to find technical solutions to one of the world's most challenging and potentially most rewarding technological opportunities, the realization of useful energy from nuclear fusion.

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ACKNOWLEDGEMENT: The author has made extensive use of material prepared in the INTOR Workshop in preparing this paper and gratefully acknowledges the contributions of the INTOR Workshop members (see Table 1.2).