MASTER

SUMMARY

USA CONTRIBUTION TO SESSION III, PHASE 1 INTOR WORKSHOP

W. M. Stacey, Jr.	(Ga. Tech)
C. A. Flanagan	(Westinghouse/ETF)
G. L. Kulcinski	(Univ. Wisconsin)
J. A. Schmidt	(PPPL)
T. E. Shannon	(ORNL/ETF)

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TABLE OF CONTENTS

INTRODUCTION

SUMMARY

1.	INTRODUCTION	1
2.	PHYSICS	1
2.1.	Poloidal Coil Configuration	1
2.2.	Impurity Control	5
2.2.1.	External Poloidal Divertor	5
2.2.2.	Bundle Divertor	5
2.2.3.	Limiter with Helium Pumping	6
2.3.	Disruptions	6
2.4.	Ripple Burn Control	7
2.5.	Plasma Scenarios	7
3.	ENGINEERING	8
3.1.	Mechanical Configuration	8
3.2.	Availability	9
3.3.	Engineering Test Configuration	9
3.4.	Maintenance Approach	9
4.	NUCLEAR	10
4.1.	First-Wall Design and Materials	10
4.2.	Limiter Design	11
4.3.	Divertor Collector Plate Design	12
4.4.	Tritium Breeding Blanket	13
5.	TEST PROGRAM	13
6.	DESIGN ASSURANCE	14
6.1.	Safety Interactions with Design	14
6.2.	Design Guidelines	14

INTRODUCTION

The International Tokamak Reactor (INTOR) Workshop is being carried out under the auspices of the International Atomic Energy Agency with participants from the European Community, Japan, USA and USSR. During 1979, the Phase-0 Workshop conducted an assessment of the technical basis, objectives, physical characteristics and feasibility of the next major experiment in the tokamak program after the generation of large tokamaks that are presently under construction. Based upon the generally positive conclusions of the Phase-0 Workshop [1], the INTOR Workshop was extended into Phase-1 with the objective of developing a preconceptual design by June, 1981.

A number of issues were identified at the end of the Phase-0 Workshop that required resolution before the starting point for the preconceptual design activity could be specified. The objective of the first six months (January-June, 1980) of the Phase-1 INTOR Workshop was to perform analyses of these issues to enable decisions to be taken at the INTOR Workshop session III, June 16-27, 1980. The USA contributions to session III of the Phase-1 INTOR Workshop are contained in this report.

REFERENCES

- [1] INTOR Group, "International Tokamak Reactor Zero-Phase Report of the International Tokamak Reactor Workshop, 1979," IAEA, Vienna, (1980); and
"INTOR Group: International Tokamak Reactor (Executive Summary of the IAEA Workshop, 1979)," Special Topic in Nuclear Fusion 20(3) 349 (March 1980).

SUMMARY

1. INTRODUCTION

The basic INTOR characteristics that were arrived at by the Phase-0 INTOR Workshop are summarized in Tables 1 and 2. These characteristics provided a point of reference for the analysis of major design issues. No requirement to change any of the suggested parameters was identified as a result of this analysis.

The major design issues that were analyzed are listed in Table 3. The results of these analyses and conclusions drawn therefrom are summarized in this chapter.

2. PHYSICS

2.1. POLOIDAL COIL CONFIGURATION

- External (to TFC) poloidal field coil system can control plasma shape. However, the external PFC system will require larger currents ($\sim 10^8$ A-T) and power (~ 1.5 GW) than an internal or hybrid PFC system. The choice of an external PFC system over a hybrid system must be based on the less demanding engineering requirements (and presumably higher availability and lower overall cost) of the former.
- The plasma vertical position can be controlled with external (to TFC) feedback coils. However, the impact of uncertainties in structural electrical characteristics on the required feedback power argue for placing this modest coil system inside the TFC.
- Startup studies indicate that 100 V/T will be required unless 5-10 MW of auxiliary heating is provided to the electrons during the initial plasma current rise.

Table 1. INTOR Suggested Parameters^a.

Major radius, R (m)	5.2
Plasma radius, a (m)	1.3
Chamber radius, r_w (m)	1.4
Elongation, κ	1.6
Burn time (s)	≥ 100
Duty cycle (%)	≥ 70
Burn average $\langle \beta \rangle$ (%)	5(6 ^b)
Av. DT density $\langle n_i \rangle$ (m ⁻³)	$1.4(1.5^b) \times 10^{20}$
Av. ion temperature $\langle T_i \rangle$ (keV)	10
Plasma current, I (MA)	6.4
DT thermal power, P_{th} (MW)	620(750 ^b)
Neutron wall load, P_n (MW/m ²)	1.3(1.6 ^b)
Lifetime (no. of burn pulses)	$5 \times 10^5 - 10^6$
NB heating power, P_B (MW)	75
NB energy, E_B (keV)	175
Fueling	pellet, gas puffing
Impurity control	divertor
Toroidal field coils	NbTi and/or Nb ₃ S _n
Toroidal field at centerline, B (T)	5.5
Poloidal field coils	NbTi
Tritium inventory (kg)	3
Maximum availability (%)	25 - 50
Tritium consumption during Stage II (kg/yr)	6

^aThe suggested parameters are tentative and suggested as input for the definition phase, during which the major parameters will be optimized.

^bFor comparison (discussed for Stage III-A).

Table 2. Features of INTOR.

-
- Relatively large ($\sim 280 \text{ m}^3$, $R = 5.2 \text{ m}$, $a = 1.3 \text{ m}$), ignited, D-shaped plasma operating at $\sim 10 \text{ keV}$ mean temperature.
 - Heated to ignition by 75 MW 175 keV NBI for 5-6 s.
 - Controlled burn for $\geq 100 \text{ s}$ with $\langle \beta \rangle = 5-6\%$.
 - Plasma current $\sim 6 \text{ MA}$ sufficient for fast alpha particle confinement.
 - Produces ~ 620 (750) MW_t thermonuclear power at a neutron wall load of 1.3 (1.6) MW/m^2 .
 - Superconducting toroidal and external poloidal field coils.
 - Water-cooled, stainless-steel shield with maximum structural temperature of $100-200^\circ\text{C}$ on basic machine.
 - Test channels and test modules (a few m^2) for blanket and materials testing.
 - Provision for testing plasma engineering components.
 - Divertor for impurity control.
 - T inventory $\sim 3 \text{ kg}$.
 - $5-10 \text{ MW}_e$ electricity production.
-

Table 3. INTOR Major Design Issues.

PHYSICS

Poloidal Coil Configuration

Impurity Control

Poloidal Divertor

Bundle Divertor

Limiter

Disruptions

Ripple Burn Control

Plasma Scenarios

ENGINEERING

Mechanical Configuration

Availability

Engineering Testing Configuration

Maintenance Approach

NUCLEAR

First-Wall Design and Materials

Stainless Steel

Aluminum

Limiter Design

Divertor Plate Design

Tritium Breeding Blanket

TESTING

Test Program

DESIGN ASSURANCE

Safety

Design Guidelines

2.2. IMPURITY CONTROL

An evaluation of poloidal divertors, bundle divertors and limiters combined with helium pumping for impurity control was carried out. A single-null poloidal divertor with external (to TFC) coils is recommended for impurity control in INTOR.

2.2.1. External Poloidal Divertor

- Space and mechanical considerations support a single-null divertor with the chamber at the bottom.
- Possible to control the separatrix to within ~ 10 cm in going from low β to high β . Difficult to provide an operating divertor prior to the heating phase.
- Providing adequate space for the scrape-off, particularly inboard, is a problem with the present INTOR parameters.
- The inside divertor throat is adequate with present INTOR parameters only if plasma triangularity ≤ 0.4 and special shielding is provided.
- The divertor pumping system dictated by the helium removal requirement is quite modest, and impurity trapping in the divertor should be good.
- Helium anti-enrichment in the divertor channel has a potentially negative impact on the tritium burnup fraction.
- Tungsten collector plates may be acceptable.
- Preliminary studies indicate that the separatrix associated with the poloidal divertor may reduce the beta limit set by MHD instabilities.

2.2.2. Bundle Divertor

- Providing adequate shielding for an INTOR-scale bundle divertor that has acceptable ripple effects remains a problem in defining a tractable engineering design. Present concepts utilize copper coils (~ 100 MW power consumption) and require organic insulators with radiation damage tolerances of 10^{10} - 10^{11} rads to achieve coil lifetimes on the order of $1 \text{ MW}\cdot\text{yr}/\text{m}^2$.

- A coil arrangement with sufficient flux expansion to accomodate a realistic collector plate design conflicts with optimizing the confinement.
- Because of the more stringent requirements for containment of injected beam ions than for containment of the fusion alpha particles, it may be necessary to divert only after the heating phase.

2.2.3. Limiter with Helium Pumping

- Limiter configurations can be designed which handle the power coming out of the plasma.
- It should be possible to meet the modest helium pumping requirements.
- Limiter erosion and plasma contamination are unresolved problems.

2.3. DISRUPTIONS

- Present experimental information is limited, and the appropriate scaling to INTOR is uncertain.
- For INTOR, the scaled energy deposition time scale is probably in the range 1-100 ms, and resistive MHD calculations predict 40 ms. A reasonable design choice is 20 ms.
- Energy is probably deposited on the inboard wall (although the top and bottom may also be impacted). For analysis, energy deposition should be over 20% of the inboard wall, will a peaking factor of 2.
- Prevention of disruptions by helical coils, feedback stabilization or current profile control should be considered further.

2.4. RIPPLE BURN CONTROL

- Dependence of ripple transport losses on T and R is favorable for burn control.
- Some automatic burn control results from Shafranov shift.
- For $\bar{n} = 10^{20}/\text{m}^3$, values of the midplane ripple at $R = R_0 + a/2$ in the range (0.15 - 0.55%) gives rise to a stable burn with $\beta < 6\%$.
- It is prudent to be able to adjust the amount of ripple by one of the following techniques.

<u>Technique</u>	<u>Design Complication</u>	<u>Time Scale</u>
Pull-back coils	support structure, power supplies	~1 s
Non-uniform TFC currents	power supplies	~1 m
Ferromagnetic shield inserts with temp.-dep. magnetic properties	flexible hydraulic system	~1 m

2.5. PLASMA SCENARIOS

Studies have been carried out to determine plausible plasma operating scenarios.

- A fully operational divertor is not required until the beginning of the neutral injection heating phase.
- The heating phase will begin when the plasma current reaches 4 MA.
- A reasonable leveling of the poloidal field power requirement is realized if the heating phase begins 2.5 s after discharge initiation.
- The current profile can be controlled by "growing" the plasma off an outside limiter.

3. ENGINEERING

3.1. MECHANICAL CONFIGURATION

- Two criteria are recommended as fundamental factors in developing the mechanical configuration:
 - 1) There should be TF coil access windows between TF coils on the outboard side that define a boundary between the cold and hot structures and that provide access for maintenance and testing.
 - 2) Access, design flexibility and maintainability are enhanced by dividing the torus into a number of segments equal to the number of TF coils so that all segments can be removed by a straight pull operation without rotation.
- The first criterion leads naturally to the concept of dedicated "bays" that utilize the access windows between TF coils:

<u>Dedicated Purpose</u>	<u>No. of Bays</u>	
	10 TFC	12 TFC
NB Injectors	4	4-6
RF Testing	1	1
Diagnostics, I & C	2	2
Fueling	1	1
Blanket and Materials Tests	2	2
Power Producing Segments	-	<u>0-2</u>
	10	12

- The second criterion tends to favor 10 over 12 TF coils from mechanical configuration considerations.
- The vacuum boundary should be located at the shield boundary.
- A single-null configuration with the null at the bottom is the preferred poloidal divertor configuration.

3.2. AVAILABILITY

- Based upon experience and anticipated results from ongoing component development programs, the statistical availability of INTOR is estimated to be $\geq 30\%$. (If certain "unacceptable" malfunctions -- e.g. TF coil failure -- are excluded from the analysis, the predicted availability is greater.)
- INTOR availability could be increased to $\sim 35-40\%$ by increasing component redundancy.
- Achievement of 50% availability in INTOR would require additional emphasis upon reliability as a criterion in component development programs.
- 50% availability should be an important goal for INTOR.

3.3. ENGINEERING TEST CONFIGURATION

- Replacement of individual test modules in special test stations is preferable to complete segment changeout for blanket and materials testing.
- The access window criterion for mechanical configuration strongly motivates the use of horizontally-inserted (rather than vertical or diagonally-inserted) test modules.
- Two dedicated bays should be sufficient for blanket and materials testing.
- Since only $\sim 10 \text{ m}^2$ are required for testing and $\sim 30 \text{ m}^2$ are required for producing $\sim 10 \text{ MW}_e$, it is preferable to produce electrical power in (dedicated) sectors rather than in test modules.

3.4. MAINTENANCE APPROACH

- All systems are designed to be maintained using fully remote procedures, but contact maintenance will be considered as a normal operation when this is possible without undue complication and expense.

- ⑥ Certain systems are designed to be very reliable and are considered semipermanent -- TF coils, trapped PF coils, bucking cylinder, inboard shield.
- ⑥ All components are designed to be maintained with existing or near-term remote maintenance equipment.

4. NUCLEAR

4.1. FIRST-WALL DESIGN AND MATERIALS

Significant analyses of bare stainless steel and aluminum alloy first-wall concepts and a limited analysis of concepts protected by a carbon "liner" were performed to assess the feasibility of designing a first-wall that would last the lifetime of INTOR (9 years at 25% availability and $P_n = 1.3 \text{ MW/m}^2$) under normal operation and taking into account the effects of disruptions. We are:

- ⑥ Able to find bare stainless steel concepts that would last the lifetime of INTOR under normal operation particle and heat fluxes.
- ⑥ Unable to determine that a bare stainless steel concept would last the lifetime of INTOR under assumed disruption conditions.
- ⑥ Able to find bare aluminum alloy concepts that would last the lifetime of INTOR under normal operation particle and heat fluxes, provided maximum temperature never exceeds 150°C.
- ⑥ Able to find a bare aluminum alloy concept that would last the lifetime of INTOR under assumed disruption erosion conditions.
- ⑥ Able to find a carbon liner (for stainless steel and aluminum alloy walls) concept that would possibly last the lifetime of INTOR under normal and disruption erosion conditions (chemical sputtering?).

- Unable to find a bare or carbon liner concept that would allow the region around the divertor throat to last the lifetime of INTOR under normal operation.
- If it were not for the evaporation during a disruption, other conditions favor stainless steel as a first-wall material.

There are three major uncertainties which make it difficult to draw definitive conclusions from these analyses:

- Modelling of the power flux to the wall in a disruption.
- Modelling of the evaporation rate and melt stability of the wall when subjected to large power fluxes during disruptions.
- Chemical sputtering of the carbon liner.

Based upon these studies and the relevant conclusions of the Phase-0 Workshop, it is recommended that:

- Three first-wall designs be carried through October, 1980:

Bare stainless steel,

Stainless steel protected by a liner on the
inboard (top and bottom?) side,

Bare aluminum alloy.

- Effort be devoted to resolving the above-mentioned uncertainties.
- Effort be devoted to developing a protected first-wall configuration in the vicinity of the divertor throat that can last the lifetime of INTOR.

4.2. LIMITER DESIGN

Two limiter designs have been investigated in order to evaluate the possibility of handling the plasma heat load without a divertor.

● Movable Tantalum (Ta-10W), H₂O-Cooled Limiter

1000 W/cm² (area ≈ 8 m²)

T_{max} = 730°C

Ta-10W thickness ≈ 2.3 mm

Erosion rate = 4.4 mm/35,000 shot

Lifetime = 8,000 shots (≈13 effective full power days)

Severe plasma contamination

● Fixed Stainless Steel, H₂O-Cooled Limiter

125 W/cm² (area ≈ 64 m²)

T_{max} = 600°C

Steel thickness ≈ 6 mm

Erosion rate = 6 mm/35,000 shots

Lifetime = 29,000 shots (≈55 effective full power days)

4.3. DIVERTOR COLLECTOR PLATE DESIGN

Two divertor collector plate designs have been investigated. Both designs are based upon a H₂O-cooled copper plate with either a tungsten tile armor plate or a renewable liquid lithium film. Both designs are intended to be replaceable and are based on a heat load of 100 MW to the divertor.

● Tungsten Tile Design

W brazed on H₂O-cooled copper plate

W thickness = 1 cm

T_{max} = 370°C

Erosion rate = 4.9 mm/y @ 25% availability

Lifetime ≈ 2 years @ 25% availability

● Liquid Lithium Film Design

Li evenly spread by capillary action through
wire mesh over H₂O-cooled Cu plate

Li film thickness = 0.05 cm

T_{max} = 310°C

Li make-up = 0.024 g/s

Lifetime ≈ INTOR lifetime?

4.4. TRITIUM BREEDING BLANKET

- There is a strong economic incentive to breed tritium in INTOR when the availability becomes more than 15%

$$\text{daily fuel cost} = \$700,000 \times \text{availability}$$

$$\text{lifetime fuel cost} = \$650,000,000$$

- Capital cost of a breeding blanket ($BR = 0.8$) would be recovered within 5 years of startup by savings in fuel cost.
- Breeding ratios of 0.8 are achievable using only the outboard regions of the torus with PbLi and Li_2O .
- Safety, tritium extraction and tritium inventory considerations favor PbLi over Li_2O .
- Breeding blankets containing PbLi must operate at $\sim 350^\circ C$, with the potential for corrosion.
- The additional risk and reduced availability associated with a tritium breeding blanket, relative to a stainless steel shield, has not been assessed.

5. TEST PROGRAM

- A preliminary definition of the INTOR test program leads to the following testing requirements:

<u>Test Purpose</u>	<u># Modules</u>	<u>Total Area</u>
Blanket	6-8	6-8 m ²
Materials	5	2-1/2
Plasma Engineering	1	1
Diagnostics	<u>1</u>	<u>1/2</u>
	13-15	10-12 m ²

- Assuming that each country will have no more than 4 blanket design concepts to test in INTOR leads to the conclusion that initial testing can be completed in 3 years, after which time emphasis will be upon fluence accumulation for reliability testing.

6. DESIGN ASSURANCE

6.1. SAFETY INTERACTIONS WITH DESIGN

- Confinement philosophy has key features
 - 1) operational exposure "as low as possible"
 - 2) accidental total machine disassembly or total tritium release not credible.
- Lithium metal acceptable only in blanket test modules.
- Magnetic field exposure criteria (Alpen committee) could be adopted, and the requirement to discharge TF coils before personnel entry into reactor cell could be imposed.
- The effect of potential major accidents (disruptions, loss of coolant, loss of flow, magnet quench, lithium fire, tritium release) should be analyzed to guide the development of design safeguards.
- Device protective systems should be integrated into a single master protective system.

6.2. DESIGN GUIDELINES

- Guidelines should be limited in scope so as not to preclude innovative approaches.
- Specific guidelines for structural design criteria (basically ASME code) are recommended for superconducting magnets and large structural systems.
- Specific guidelines for tritium handling systems are recommended
 - Annual tritium release <20 Ci/day and $<10^4$ Ci/year.
 - Operating dose limit to workers $<10^3$ mrem/year and to public $<10^2$ mrem/year.