INTOR WORKSHOP: DESIGN CONCEPT, CRITICAL ISSUES, INNOVATIONS, DATABASE ASSESSMENT

(Summary)

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This paper is one of two presented at this conference that summarize the final phase (1985–1987) of the INTOR Workshop. It reports work on the INTOR design concept and how it would be modified in the light of new information, analysis of critical technical issues that affect the feasibility of next generation tokamak experiments, innovations that could improve the tokamak as a reactor concept and an analysis of the database that supports a next generation tokamak experiment. A companion paper by G. Grieger (IAEA-CN-50/F-I-2) reports the critical analysis of INTOR-like designs that was performed by the INTOR Workshop.

INTOR is viewed as the next major step in the world tokamak programme and has the general objectives of serving as an engineering test reactor which demonstrates the physics and integrates the technology required for a subsequent DEMO and which provides an engineering testing capability. This general objective leads to several technical objectives: long, controlled burn (> 100 s) with a neutron wall load of $P_n \gtrsim 1$ MW/m^2; a maximum availability of 25%; reactor relevant technologies; and a neutron fluence of ~ 3 MW·a/m^2.

An initial INTOR design concept was developed via a conceptual design in 1981 and was subsequently updated in 1983 and again in 1985. The most recent design concept is characterized in Table I.

During 1985–1987 the workshop carried out a continuing assessment of the evolving tokamak physics and technology databases, analysed several critical technical issues and evaluated a number of possible innovations, all in relation to the INTOR design concept.

Results from tokamak experiments and from extensive analysis continue to support the conclusion that a high recycling poloidal divertor with a tungsten target is the best available impurity control system, although innovations such as a liquid metal divertor target and helium extraction by surface burial are extremely promising. The value of beta chosen for INTOR in 1985 has turned out to be somewhat optimistic (requiring a Troyon factor of 4 with the present parameters), and some increases in plasma current and elongation are probably necessary. On the other

TABLE I. INTOR DESIGN PARAMETERS

Major radius	4.9 m	Average density	1.4×10^{20} m^{-3}
Plasma radius	1.2 m	Average temperature	10 keV
Maximum toroidal field	11 T	Beta	4.9%
Plasma current	8 MA	Safety factor	1.8
ICRH	50 MW	Tritium breeding ratio	0.6
LHR current drive	20 MW	Structural material	SS
Neutron wall load	1.3 MW/m^2	Coolant	H_2O
Single null poloidal divertor		Total volt-seconds	112

hand, the INTOR density seems to be safely below the limit extrapolated from experiment, and the 1.4 s energy confinement time required for ignition should be achievable with the present parameters. The experimental database and extensive calculations of non-inductive current drive would now justify reliance upon either a combination of NBI and LHR (preference) or NBI alone, with a consequent reduction in inductive volt-second requirements.

Substantial progress has been made in the computer modelling of transient electromagnetic phenomena. Control of vertical and radial instabilities can be accomplished by a combination of the passive stabilization provided by the torus structure and a set of active control coils (20–50 MW required) placed outside the blanket-shield but inside the toroidal field coils.

An extensive comparison of configurations based upon horizontal (present reference) and vertical access to the torus for assembly and maintenance has led to the conclusion that horizontal access would be simpler, and therefore preferable, for moderate plasma elongation, but that vertical or oblique access would be preferable for plasma elongations ≥ 2.

Austenitic stainless steel is the only reasonable structural material, although uncertainties related to aqueous stress corrosion and to disruption characteristics are of concern, and low Z protective tiles may be required for protection of the first wall against fatigue cracking due to disruptions. It appears possible to achieve tritium self-sufficiency with a ceramic breeding blanket covering $\sim 60\%$ of the upper and outboard parts of the plasma chamber.

If the INTOR design were to be updated again, on the basis of the work of 1985–1987, it is likely that some combination of plasma current increase, magnetic field increase and change in dimensions would be made to enhance the likelihood of MHD stability. Non-inductive current drive would be relied upon to achieve the basic performance objectives, if this resulted in a better design. If it proved necessary to increase the plasma elongation above about 2, a vertical or oblique maintenance scheme would replace the present horizontal scheme.

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