J. Plasma Fusion Res. SERIES, Vol. 5 (2002) 137–142

Status of the ITER Project

TSUNEMATSU Toshihide and NAGAMI Masayuki
Japan Atomic Energy Research Institute, Ibaraki 311-0193 Japan

(Received: 14 December 2001 / Accepted: 10 April 2002)

Abstract

The status of the ITER Project is presented. The objective of ITER is to demonstrate the scientific and technical feasibility of fusion energy for peaceful purposes. The Engineering Design Activities (EDA) was concluded in July 2001 to finalize the nine-years design and R&D activities. The design of the ITER has been developed not only to satisfy these requirements but also to have the flexibility of plasma operations. This flexibility will provide an advantage for coping with uncertainties in the physics database, in studying burning plasmas, in introducing advanced features. The governmental negotiations has been initiated by aiming at the ITER Construction, Operation, Exploration and Decommissioning Activities in early 2003.

Keywords:

ITER, engineering design, R&D, operation region, negotiation

1. Introduction

The history of co-operation on ITER started in 1985 when government leaders in summit meetings called for more substantial international co-operation in order to increase the efficiency and minimize the cost of fusion power development. In response to the summit initiatives, the Conceptual Design Activity (CDA) began in 1988 after the four Parties formally accepted the invitation from the Director-General of the IAEA to participate, in accordance with the terms of reference, under the auspices of the IAEA. The CDA were successfully completed in 1990. After completing the CDA, the four Parties, Euratom, Japan, Russian Federation and the USA, entered into negotiations on how ITER should proceed further, resulting in the ITER Engineering Design Activity (EDA). The Agreement was signed on July 21, 1992 in Washington by representatives of the four Parties and entered immediately into force. Joint work during the six years period originally set for the Agreement led, by July 1998 to the development of a design for ITER that was adjudged to fulfill the overall programmatic objective and the detailed technical objectives and cost target as originally set. With the approaching end of the original period set for the EDA, the Parties negotiated an agreement extending the EDA period for three years which was signed by three Parties, Euratom, Japan and the Russian Federation. The USA agreed to participate for one additional year. At that time, by recognizing the possibility that they might be unable, for financial reasons, to proceed to the construction of the then foreseen device, the Parties adopted a less demanding set of detailed technical objectives that would still meet the overall programmatic objectives but at a significantly reduced cost (about 50%). The EDA extension period was primarily devoted to developing a design under the revised technical objectives set in 1998, as well as completion of planned technology R&D projects. The results of these activities have been reflected in the report of the ITER design [1] presented by the ITER Director in July 2001. At the last moment of the EDA period, Euratom, Japan and Russian Federation initiated a non-committal discussion on the

Corresponding author's e-mail: toshit@naka.jaeri.go.jp

©2002 by The Japan Society of Plasma Science and Nuclear Fusion Research

implementation of the construction, operation and decommissioning of ITER and the formal negotiation has started in November 2001among the above three Parties and Canada.

2. ITER EDA Objectives

The purpose of the ITER Agreement is to produce a detailed, complete, and fully integrated engineering design of ITER and all technical data necessary for future decisions on the construction of ITER. Such design and technical data shall then be available for each of the Parties to use either as part of an international collaborative programme or in its domestic programme.

2.1 Programmatic objectives

The overall programmatic objective of ITER is to demonstrate the scientific and technological feasibility of fusion power for peaceful purposes. ITER would accomplish this objective by demonstrating controlled ignition and extended burn of deuterium-tritium plasmas, with steady-state as an ultimate goal, by demonstrating technologies essential to a reactor in an integrated system, and by performing integrated testing of the high-heat-flux and nuclear components required to utilize fusion energy for practical purposes.

2.2 Technical objectives

Detailed technical objectives to determine the best practicable way to achieve the programme objectives were adopted in 1992 by the ITER Council and served as the focus for design work for the first six years of the EDA Agreement. In 1998 revised detailed objectives were adopted by reducing the size of device as well as the capital cost, while maintaining the programmatic objectives and by taking account of the progress in physics and technology R&D. Detailed technical objectives adopted by ITER Council in 1998 are summarized as follows:

1) Plasma Performance:

- to achieve extended burn in inductively driven plasmas with the ratio of fusion power to auxiliary power injected into the plasma $Q \geq 10$ with an inductive burn duration between 300 and 500 s,
- to aim at demonstrating steady-state operation using non-inductive current drive with $Q \geq 5$,
- controlled ignition should not be precluded.

2) Engineering Performance and Testing:

- demonstrate availability and integration of essential fusion technologies,
- test components for a future reactor,
- test tritium breeding module concepts; with a 14 MeV neutron average power load on the first wall ≥ 0.5 MW/m^2 and fluence ≥ 0.3 MWa/m^2,
- the option for later installation of a tritium breeding blanket on the outboard of the device should not be precluded.

3) Operation requirements:

- the operation anticipated over an approximately 20 year period should address the issues of burning plasmas, steady-state operation and improved modes of confinement, and testing of blanket modules.

3. Validating Research and Development

The overall design philosophy for ITER has been to use established approaches and to validate their application through wide-ranging physics and engineering work. This included detailed analyses, experiments in existing fusion research facilities, and dedicated technology developments including fabrication and test of full scale or scalable models of key components [2]. Unavoidable uncertainties remain in the extrapolation of performance from current experience to the ITER size and parameters; these can only be fully resolved through experiments at ITER scale. On a voluntary basis, the Parties have conducted well-focused physics investigations that strengthen further the physics database, reduce the ranges of uncertainty in extrapolation and explore wider options for possible ITER operation. The results of ITER Physics R&D has been summarized [3] and contributed to the prediction and the design of the ITER plasma.

The design also addresses demanding technical challenges including:

- the unprecedented size of the superconducting magnets and structures,
- high neutron flux and high heat flux in the first wall,
- extremely high heat flux in the divertor,
- remote handling for maintenance and interventions in an activated tokamak structure,
- equipment unique to fusion reactors such as fuelling and pumping, heating and current drive systems and diagnostics.

Major technology R&D projects therefore have been undertaken for the purpose of confirming performance and understanding operating margins. Assessment of results indicates, from an engineering and technology point of view, that the design is feasible, that it can be manufactured to specifications, and that it will be capable of meeting its operating objectives. In

Table 1 Main Parameters of ITER

Total Fusion Power	500 MW *(700 MW)*
Q – fusion power/additional heating power	≥10
Average 14MeV neutron wall loading	≥0.5 MW/m^2
Plasma inductive burn time	≥400 s
Plasma major radius (R)	6.2 m
Plasma minor radius (a)	2.0 m
Plasma current (Ip)	15 MA *(17 MA)*
Toroidal field at 6.2 m radius (B_T)	5.3 T

Note: The machine is capable of plasma current up to 17 MA, with the parameters shown in parentheses, within some limitations on other parameters such as pulse length.

particular, seven large projects, undertaken jointly by the Parties have demonstrated the feasibility, performance, and maintainability of the main engineering components and systems of ITER under the international collaboration. Some key technologies such as in the central solenoid and the vacuum vessel were manufactured and tested by a commingled procedure among several countries and industries. In addition to the seven large R&D projects, development of components for fuelling, pumping, tritium processing, heating/current drive, power supplies, plasma diagnostics as well as safety related R&D have significantly progressed.

4. Safety and Environmental Assessments

A central goal of ITER is to demonstrate the safety and environmental potential of fusion as an energy source. A consensus across the participant Parties on safety principles has been attained, based on internationally recognized safety criteria and radiological limits following ICRP and IAEA recommendations, and in particular on the concept of defense in depth and the As Low As Reasonably Achievable (ALARA) principle. Comprehensive safety and environmental assessments of the ITER design and operation have been completed under generic site assumptions. The conclusion indicates that ITER will meet the objective of demonstrating the safety and environmental potential of fusion power.

5. ITER Design and Performance

5.1 ITER design

The design was developed to satisfy the technical objectives and to include the latest progress in physics and technologies mainly from the ITER R&D activities.

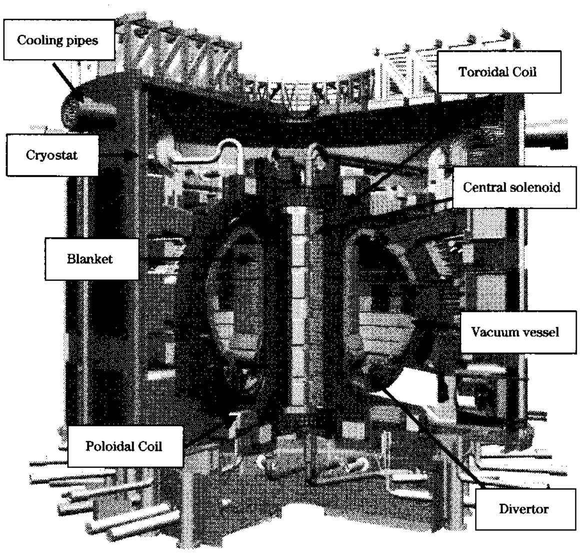

Fig. 1 Cutaway view of the ITER device

The joint work has yielded a mature design, cost estimate and safety analysis. The main parameters of ITER are shown in Table 1. Figure 1 shows a cutaway view of the ITER device inside the cryostat. Detailed designs for major specific components have been developed, coherence of parts or subsystems with the whole has been achieved, and all major outstanding design issues are resolved except site-specific adaptations, based on the design work and fabrication experience of R&D components. The essential engineering features on the tokamak core include:

- A segmented center solenoid coil with six coils to control strongly shaped plasmas;
- An integrated structural arrangement in which super conducting magnets (18 toroidal field coils,

and 9 poloidal field coils) and vacuum vessel are linked to provide an overall assembly which simplifies the equilibration of electromagnetic loads in all conditions, relying largely on the robustness of strong TF coil cases ; and

- Modular in-vessel components (blanket modules and divertor cassettes) designed to be readily and safely maintainable by a practical combination of remote handling and hands-on techniques.

The tokamak is contained in a cryostat vessel, situated in a pit, inside a building (the tokamak building) of about 50 m height. Peripheral equipment such as fueling and pumping, heat transfer, auxiliary heating and remote handling systems are arranged in galleries around the main pit. The seismic condition depends on the site. If the seismic ground peak acceleration is larger than 0.2 G, the tokamak building will be supported by flexible isolators. This seismic isolation minimizes the design changes due to different seismic conditions. The main services required for ITER such as the electrical power, cooling water, fuel treatment, information flow, assembly and maintenance facilities, waste treatment, etc. are distributed in ancillary buildings and other structures throughout a site of 25 hectares overall.

5.2 Plasma performance

The design of the ITER tokamak has flexibility of plasma operations which will allow exploration in a large operation space of fusion power, plasma density, plasma beta, pulse length and Q value in various operation modes and will provide a wide range of opportunities to study burning plasmas as well as to develop reactor core plasmas. This paper describes shortly possible plasma operation modes in ITER to study issues of burning plasma physics and fusion reactor core development.

Figure 2 shows the operation space in the fusion power vs the confinement capability for Q=10, 20 and 50. The HH factor denotes the deviation from the fitting formula (HH=1) to the experimental data. The plasma confinement time (τ_E), HH factor, L-H transition power ($P_{L\text{-}H}$), Greenwald density (n_G) and normalized beta (β_N) are given as follows [4]:

$$\tau_E = H_H \, \tau_{E,th}^{IPB98(y,2)}$$

$$\tau_{E,th}^{IPB98(y,2)} = 0.144 I_P^{0.93} B_T^{0.15} P^{-0.69} n_e^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa_a^{0.78}$$

where the units are s, MA, T, MW, 10^{20} m^{-3}, amu, m and $\kappa_a = V/(2\pi^2 R a_2)$

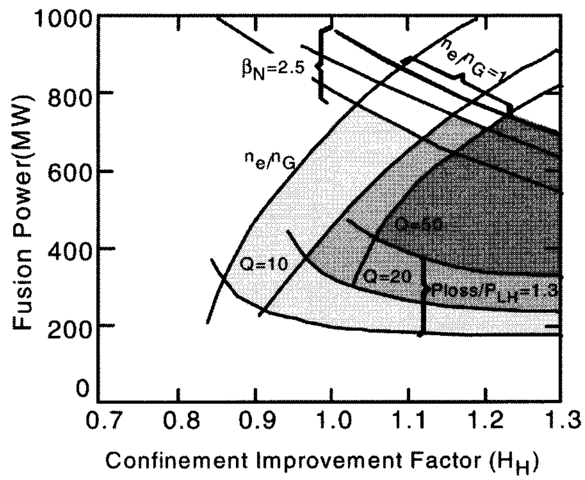

Fig. 2 Operation region of ITER, Q=10, 20 and 50 for I_p=15MA. The regions are bounded by the density limit, beta limit and the power threshold to the H-mode transition.

$$P_{L-H} = 2.8 M^{-1} B_T^{0.82} \bar{n}_e^{0.58} R^{1.00} a^{0.81} ,$$

$$n_G = I_P / \pi a^2$$

and

$$\beta_N = \beta(\%) / (I_P / a B_T) .$$

For Q=10 with I_p=15MA, the predicted performance includes about 15% of the margin in the confinement. The burn time is 400–500 seconds. As Q increases, the margin in HH decreases and about 10% of the improvement in the confinement is needed for Q=50. By increasing the plasma current up to 17MA, the operation space has about 10% of the confinement margin shown in Fig. 3. The burn time of this operation is limited to be about 100 second due to the capability in the burn flux and in the cooling system. To prolong the pulse length, the ITER has a capability to install a non-inductive current drive. Figure 4 shows the operation points of Q=5 for different loop voltages. As an ultimate state, a steady state operation in ELMy H-modes is attainable for 100 MW. If the confinement is improved in e.g. a negative shear configuration and b$_N$ can be increases up to 3.5, the steady state operation with Q=10 is available in ITER.

Many design studies of possible or desirable fusion demonstration reactors and power plants have been done and these provide requirements for reactor core plasmas and a good basis of experimental plans. The global energy confinement would not be a major issue, and the

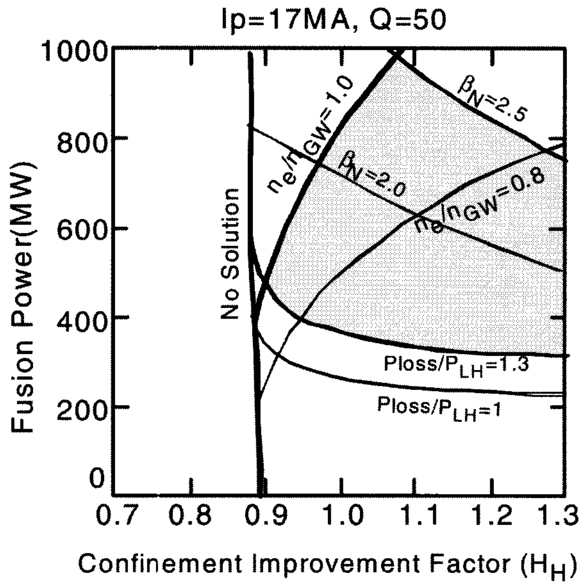

Fig. 3 The enhanced operation region for Q=50 and I_p=17 MA. The region has a margin for the confinement, compared to that in Fig. 2.

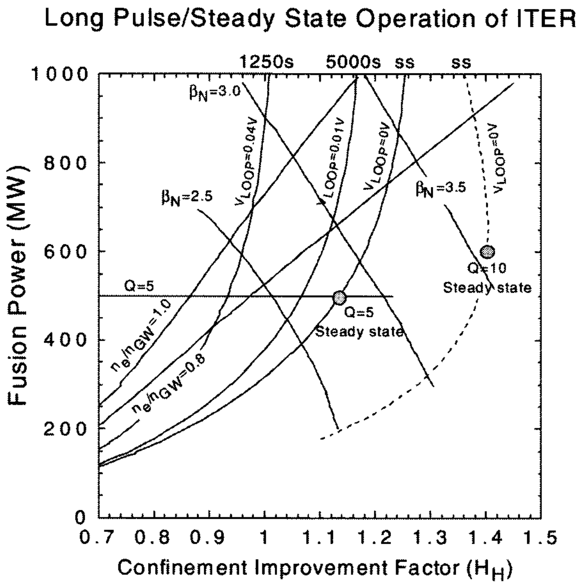

Fig. 4 The operation points of Q=5 for different loop voltages. As an ultimate state, a steady state operation in ELMy H-modes is attainable for 100 MW. If the confinement is improved in e.g. a negative shear configuration and β_N can be increases up to 3.5, the steady state operation with Q=10 is available in ITER.

enhancement factor (H_H) from the ELMy H-mode confinement should be around 1. Reducing recirculating power, increasing fusion power density and increasing reliability are major requirements. These requirements impose the following plasma parameters in a steady-state reactor (SSR) and in a pulse reactor (PR):

- high normalized beta (β_N) typically more than 3 in SSR and 2.5 in a PR,
- high bootstrap current (I_{BS}) fraction typically more than 60% in a SSR,
- low divertor heat load typically less than 20% of alpha and external heating power, and
- avoidance of disruptions.

These requirements are not always necessary to achieve the required performance in ITER. In order to explore regimes close to fusion power plants with such as a high normalized beta and a high bootstrap current fraction in ITER, methods to improve parameters are needed especially in the confinement, the plasma density profile and the plasma beta, compared to the assumed standard mode in ITER, i.e., the ELMy H-mode. At this moment, only indicative definite database exists for the steady state of the improved plasma region. The exploration of this region in a burning condition is one of the research items in ITER and the present design includes the capability in the magnet and control system for accommodating such advanced research.

6. Status after the EDA

Delegations from Canada, the European Union, Japan and the Russian Federation started formal negotiations on the joint implementation of the ITER project. Canada presented its bid to host the ITER project in Clarington facing Ontario Lake. European Union is preparing a proposal of Cadarache site in south France. The Russian Federation is preparing a proposal of in-kind contributions. In Japan, a possibility to host the project has been intensively studied. Following the work plan accepted by the Delegations, consensus on site preference, cost sharing and procurement allocation, and understandings on Nominee Director General will be achieved in 2002. The agreement is expected in 2003. The experimental operation is expected to start in 2013. About 20 years operation is planned. Variety of physics and engineering researches will be done by various scientists and engineers. In order to use ITER efficiently as a worldwide fusion research facility, the exploitation of ITER by the laboratories and universities and personnel including graduate students participating of the Parties is planned as well as by the ITER Legal Entity staff. It is also important to enhance personnel mobility between the project and the Parties domestic programs as well as to have remote access to the facility.

References

[1] "Summary of the ITER Final Design Report", to be published by IAEA in the ITER EDA Documentation Series, IAEA, Vienna (2001). http://www.iter.org/ITERPublic/ITER/

[2] ITER Joint Central Team and Home Teams, Fusion Eng. Des. **55**, 97 (2001) Issues 2-3.

[3] ITER Physics Basis Editors, ITER Physics Expert Group Chairs and Co-Chairs, ITER Joint Central Team and Physics Integration Unit, ITER EDA, Nucl. Fusion **39**, 2137 (1999).

[4] Shimada, M. *et al.*, *"Physics Basis of ITER-FEAT", Fusion Energy 2000* (CD-ROM, IAEA, Vienna), ITER p.105 (2001), IAEA-CSP-8/C, ISSN 1562-4153.