Report on
Technical Feasibility of Fusion Energy
and
Extension of the Fusion Program and Basic Supporting Researches

May 17, 2000

The Subcommittee of The Fusion Council for Fusion Development Strategy

translation by JAERI

Contents

Introduction

Fusion development in this country is promoted in accordance with the “Third Phase Basic Fusion Research and Development Program” decided by the Atomic Energy Commission in June 1992. This program declares the achievement of the self-ignition condition and long burn as its major objectives, and chooses the tokamak experimental reactor as the central device on which the achievement of these targets can be expected from the current technology. In August 1996, the Atomic Energy Commission further recognized the International Thermonuclear Experimental Reactor (ITER), which was in the Engineering Design Activity (EDA) phase by an international collaborative effort, as that tokamak experimental reactor.

When compared with other existing fusion experimental devices, ITER is significantly larger in dimensions, it should have entire social consensus for its construction. For this reason, the Atomic Energy Commission formed the Special Committee for the ITER Project (Special Committee) consisting of knowledgeable and respected members that represent various societies of the nation to intensively evaluate whether construction of the ITER would be appropriate. The Special Committee submitted the interim report entitled “Summary of the Discussion in the Committee and the Future Issues” in March 1998, and pointed out the following six issues that must be investigated and clearly answered so that the decision to propose hosting the construction of ITER could be made;

(1) Survey of long term demand and supply of energy sources
(2) Feasibility study of alternative energy sources
(3) Technical feasibility of the fusion energy
(4) Extension of the fusion program and basic supporting research
(5) Distribution of resources for research
(6) International relations.

On June 12, 1998, the *Fusion Council* formed the Subcommittee for Fusion Development Strategy to conduct discussions on the strategy for the development and realization of fusion energy. The subcommittee consists of specialists involved in fusion research from universities, the Japan Atomic Energy Research (JAERI), and industry. In these discussions, outside specialists having expertise in specific fields are invited to provide knowledge and opinions as required.

For topic (3) above, the Special Committee additionally requested an evaluation of the feasibility of fusion energy as a safe and reliable energy source from the aspects of technical potential, management capability, and characteristics of Japanese industrial structure. The evaluation was to include the involvement of a broad range of industries. For topic (4), the Special Committee requested a detailed overall plan for the realization of fusion energy. This plan was to include the roles of universities and industry in basic research fields such as advanced reactor studies and materials development, and the education and training of personnel, all which would support the ITER project and fusion development beyond. Further, it should describe the desired cooperative structure between the universities and industry, should Japan host ITER.

Based on June 12, 1998 request from the Fusion Council, the subcommittee intensively discussed the following three issues;

(1) Technical feasibility of fusion energy,
(2) Formation of the foundation for basic research that will support fusion reactor development over a long period including education of personnel, and the roles and cooperative structure of universities and industry, and
(3) Matters on development strategy of the fusion reactor.

This subcommittee focused on the discussion of the technical feasibility of the tokamak fusion reactor whose confinement concept is same as that of ITER. The “Third Phase Basic Program for Fusion Research and Development” states that in selecting the core device for the next phase, concepts other than tokamak should also be considered and those alternative concepts that might surpass the tokamak should not be excluded. Knowledge obtained from tokamak alternatives is valuable for ITER support.

The subcommittee has held 25 discussion meetings from its inception through April 2000. This report summarizes the results.

PART 1

Technical Feasibility of Fusion Energy

Chapter 1. Future Prospects of the Fusion Energy

Fusion energy is the energy source of the sun and other stars, and if its utilization becomes possible in a controlled way, it will be the ultimate energy source for mankind. Because of the potential of this goal, research and development of fusion has been conducted individually by countries and under international collaborations among the developed countries of the world.

If Japan can lead the pursuit and succeed in attaining this magnificent goal, the realization of fusion energy, it will permanently resolve the energy problem of our country, and significantly contribute the sustainable prosperity of humankind. Our country has had little experience in being a “Founding Father” of new technologies, including nuclear fission. However in fusion research and development, Japan sustains the high level as a top runner in international fusion plasma physics and fusion technology and is qualified to play a leading role in fusion development. As in the case of the United States in the International Space Station project and the European Union in the Large Accelerator, Japan is capable of contributing to the world in fusion development. This role is suitable for Japan by the reason that it has poor energy sources.

Development of fusion requires large funding and personnel resources over a long period, and thus requires continuous support from the public, which enhances the morale of the specialists involved in the research and development. To inform and educate the public, explanations should be presented to the public about how electricity would be generated by fusion, and benefits provided to humankind in the future when fusion power becomes a realization. We should then convene public discussions addressing the advantages and disadvantages of fusion. Finally, we should request a public consensus about Japan’s role in fusion development. At the same time, the specialists must remain keenly aware of public concerns and make every effort to quell these concerns and to enhance the advantages of fusion energy source while overcoming the disadvantages. From this viewpoint, some assessments have already been made on the research and development of fusion reactors and the social acceptance of these reactors [1.1-1, 1.1-2].

The remainder of this chapter analyzes energy source issues such as, resources, the environment, safety, economics, etc., and the relationship these issues have with fusion energy in comparison with other energy sources.

1.1 Situations Surrounding Energy Enterprises and Energy Options in the 21st century and Beyond

(1) Emissions of Carbon Dioxide and Global Warming

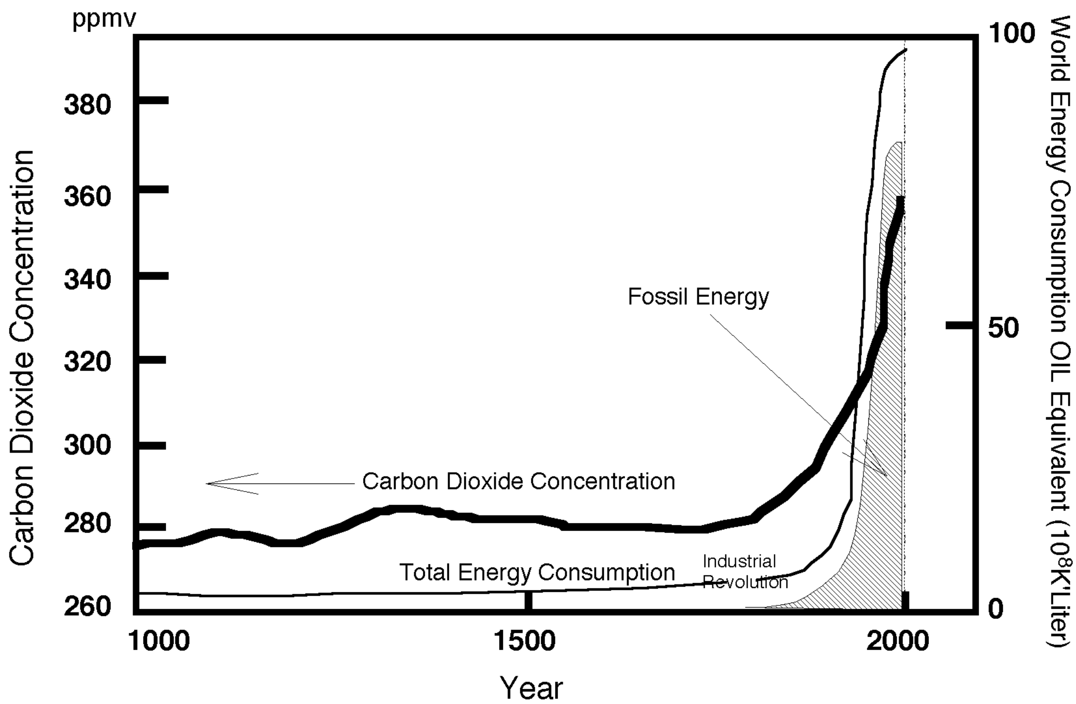

Fig. 1.1-1 Total energy consumption by humankind and variation of atmospheric CO_2 concentrations [1.1-3]

In the history of humankind, energy consumption has never increased as rapidly as it has in the last few hundred years. Most of the increase in demand was supplied by fossil fuels such as petroleum and coal. The consumption of energy for the last millennium is shown Fig. 1.1-1. Consumption of fossil fuel has drastically increased since the Industrial Revolution in the 18th century. With this situation continuing, our valuable fossil fuel resources, which nature has accumulated for hundreds of millions of years, will be exhausted in only a few hundred years.

However, it has recently been recognized that the possible climatic changes resulting from the atmospheric increase of CO_2 may be more serious than depletion of fossil fuels--and this concern may have to be faced before fossil resources are exhausted. As seen in the Fig. 1.1-1, the atmospheric concentration of CO_2 started its rapid increase at the end of the Industrial Revolution. It is well known that this escalation is linked to the combustion of fossil fuels.

Evidence shows that the increase in atmospheric CO_2 concentration has caused global warming of the climate by the greenhouse effect (refer to Section 4.3.2.). Since the change is so slow compared with a human lifetime, it is difficult a person to actually perceive. It is however considered that the air temperature has increased approximately 0.3 – 0.6°C in the past hundred years. Figure 1.1-2 shows the comparison between the predictions of three global warming models and the measured temperature. Climatic temperature changes cannot be explained only by the increase in CO_2 concentration, however, combining this concentration with the shielding effect on sunshine by volcanic smoke and debris and changes in solar activity allows the numerical model to describe the changes in air temperature more accurately. This suggests that the increase in CO_2 concentration leads to a net increase in atmospheric temperature.

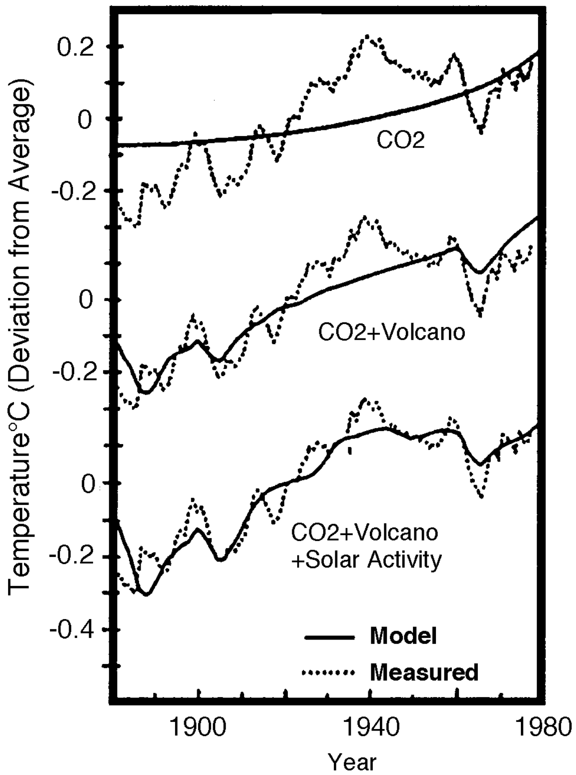

Fig. 1.1-2 Temperature variations, actual and predicted, over the last hundred years Measured value (dotted line) and calculated value by simulation model [1.1-5].

The prediction for future global warming due to the increase in CO_2 will include some uncertainty. Approximately a 2.5°C warming is anticipated according to the IPCC 95 report [1.1-4] in the scenario where the CO_2 concentration reaches 550 ppm, twice that from before the Industrial Revolution. The various consequences of such warming are also being investigated; for instance, the rise of sea level is predicted to be in the range of 50 – 100 cm. Significant ocean coastal areas might be flooded if no countermeasures were taken to prevent the sea level rise—and if the countermeasures were ineffective, huge costs will be incurred to mitigate the probable resulting flooding. There would most likely be other climatic effects; such as an increase in the severity and number of typhoons and the direct and indirect effects to the earth's ecology caused by the change of temperature. Compared with the rise in sea level, climatic changes caused by global warming are difficult to anticipate, but catastrophic changes might happen at unexpected times. Such changes may be potentially more serious than a rise in sea level, which would be slow and would allow time for sufficient preparation against it.

Much of this damage likely to be caused by climate changes is difficult to estimate quantitatively. And, regardless of the amount of monetary losses, it is difficult to assure environmental safety with a climate change of such a large scale. Moreover, such an environmental change may be irreversible. Considering above observations, global warming should not be allowed to simply take its own course; it would be prudent to suppress the concentration of atmospheric CO_2 as much as possible. It should be stressed that the understanding of our environment is limited and the future risks should not be underestimated.

(2) Energy Demand and Supply, and the Reduction of CO_2 in the 21st Century

A possible scenario has been proposed by ICPP95 to stabilize the atmospheric CO_2 concentration in late 21st century at 550 ppm, which is twice the value before the start of the Industrial Revolution. In this scenario, it is said that generation (emissions) of CO_2 must be limited at the present-day level throughout the 21st century and then be reduced after that time. However, in spite of the efforts to reduce the energy consumption and to improve its utilization efficiency, the gross energy demand in late 21st century will be at least twice as much as it was in 1990. This will result from of the population growth and improvements in the standard of living, mainly in the developing countries in Asia and Africa. Figure 1.1-3 shows an estimate of world energy demand assuming the maximum control of CO_2 as proposed by IPCC95. If this energy is supplied mainly by fossil fuels, as it is now, emission of CO_2 at the end of the 21st century will have doubled from the present-day level. To solve this CO_2 problem, corrective efforts are needed in every category of energy consumption such as electricity, transportation, daily life, and industry. In addition, fossil resources are indispensable raw materials for the chemical industries. Accelerated consumption of this fossil fuel resource as fuel would undoubtedly be regretted in the future. Let us consider the situation where an innovative energy source could be introduced in place of fossil fuel—and that source can supply a large amount of energy without CO_2 generation. It could reduce not only CO_2 originating from the production of electricity (in Japan it is 25% of the total CO_2 generation), but also the CO_2 originating from other categories, such as transportation, by converting this primary energy to secondary energy using new technologies, such as the production hydrogen for fuel. Valuable and limited fossil resources could then be reserved for purposes that are more effective.

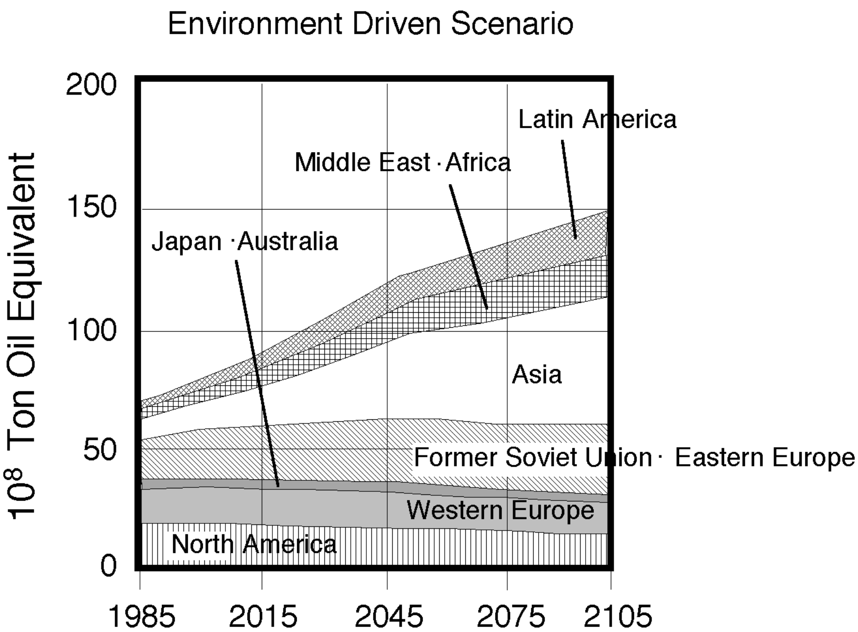

Fig. 1.1-3 Prediction of world energy demand (suppression scenario, IPCC95) [1.1-4]

Even if the so called New Energies such as solar and wind could be used fully as a substitute for fossil energy sources, reduction of CO_2 only by these sources is estimated to require a significant cost increase when compared to the cost of CO_2 sequestration at fossil fuel power stations [1.1-6]. Thus, it is unrealistic to attempt to solve the CO_2 problem only by these means. Further, supplying electricity produced by solar and wind energy, both of which are variable and intermittent sources since they are weather dependent, into the electrical network would only provide a maximum of 10% of the capacity of the network.

On the other hand, fission nuclear energy that does not emit CO_2 during its operation is expected to be a countermeasure for CO_2 reduction. However, there remain difficult problems to be solved concerning the disposal of high-level radioactive waste, which requires careful handling, storage, and disposal. In addition, careful attention should be paid on international relation from the viewpoint of the proliferation risk of nuclear materials, such as plutonium. While fission plants are already supplying large amounts of energy and so called severe accidents are extremely unlikely to occur, it is not easy to obtain and sustain good public acceptance for light water reactors judging from public opinion, which has been adversely influenced by the recent nuclear accidents. As for the world circumstance, construction of new fission reactors is not quite easy all over the world, particularly in developed countries. In the future, there will be increased needs of a system that is inherently safer than the light water reactor and that would present no major hazard to the public in any possible sequence of events.

In conclusion, it is impossible to solve global environmental problems with the present-day energy generation technology now in use. To solve them completely, development of innovative technologies is imperative.

(3) Situation Surrounding the Energy Producers and Costs

The electricity supply from energy enterprises is entering a period of free market competition, not only in Japan, but also throughout the world. For example in Japan, an IPP (Independent Power Producer) can now participate in the electricity supply business even if the IPP electricity is partial participation. This is forcing the electric utilities to make efforts to reduce generation costs to compete with them. The IPP sources of electricity are mostly thermal in origin, so most of these have a negative effect on the reduction of CO_2. This tendency is same for other countries promoting free competition of electricity producers. Some wind and solar power has been partially introduced with various financial support incentives. Such electricity is subject to an unstable output and can provide only up to 10% of the network capacity at best. To increase the fraction of power produced from these sources, electricity storage equipment will have to be added to stabilize the output--and the cost of such power will drastically increase. Disregarding the environmental impact when considering the cost of the production of electricity, coal and natural gas power are advantageous economically in the near term. However, factoring in the environmental impact, the reduction of CO_2 would be extremely difficult.

To reduce fossil fuel produced electricity, some kind of penalty for CO_2 emission will be temporarily needed. Such penalties against low-cost fossil-power stations will force the use of more expensive power sources and that, unfortunately, will unavoidably cause an increase in the cost of electricity. Since the world's countries have different circumstances, fossil fuel power stations cannot be prohibited completely. The countries adopting clean but expensive electricity will suffer disadvantages in industrial competitiveness in international markets. Therefore, it will be difficult to apply an expensive penalty for CO_2 emission for a long period in a country whose economy depends mainly on manufacturing industries, such as Japan. On the other hand, a low CO_2 tax will not be effective reducing CO_2 emissions. Development of CO_2–free innovative energy technologies is thus necessary to realize global CO_2 reduction while providing electricity at a reasonable cost.

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1.2 Criteria for Commercialization of Fusion Electricity Generation from the Viewpoint of Business Management

1.2.1 Introduction

When an electricity enterprise considers a new energy source, the issue of assessment is not the performance of each component of the power reactor but the overall performance as one of the power sources in the network. Irrespective of the process used to produce energy, important requirements from the business management are:

Low cost of electricity generation,
High reliability (availability),
Good energy security,
Wide range of site acceptability, and
Easy operation and maintenance.

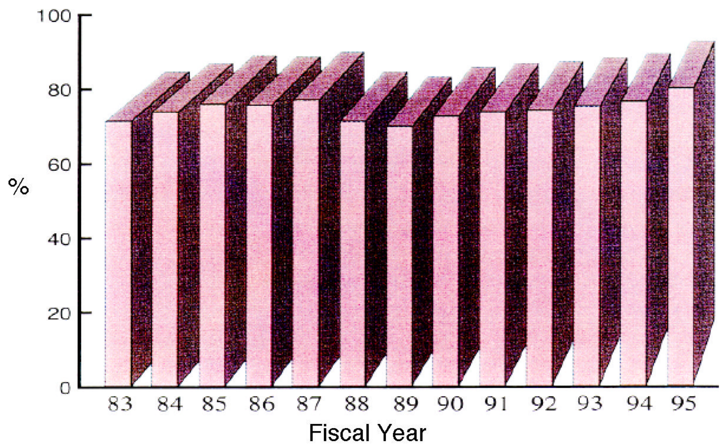
Furthermore, specifications will be decided by considering the network-specific situation in the case of construction. There will be no particular reason for fusion to have a privilege, and the requirement from the electric utility will thus be similar to that for other energy sources. The performance of each component and system of the fusion reactor must be optimized to meet these requirements.

Although it is difficult to show the desirable technical performance of fusion reactor quantitatively, several important quantities can be summarized and compared with present electricity sources. Requirements for the fusion reactor may not be quite the same as those for present sources, but these values will at least need to be regarded as a technical standard.

1.2.2 Consideration of Specifications required for Fusion Reactor in Comparison with Fossil and Fission Power Plants

(1) Capacity factor

Table 1.2.2-1 summarizes the actual capacity factor of the present-day fossil fuel and fission power plants. Present sources of both fossil fuel and fission power maintain a very high capacity factor (or availability) over 80%. The capacity factor must be maintained as high as that of fission power reactors because fusion reactor will be expected to perform as a high-duty-factor base-loaded plant. The rate of unscheduled outages must be no higher than that in light water reactors (although the details cannot be discussed now). However, even if the capacity factor of the early reactors is not as high as planned, it may be understood that the capacity factor will eventually be improved if the problems can be solved. The decrease in availability resulting from blanket replacement is an essential issue to be solved. The simplification of maintenance and replacement operations and the extension of the lifetime of the blanket and related parts will be a mandatory requirement.



$$\text{Capacity Factor} = \frac{\text{Total Electricity Generation}}{\text{Facility Capacity} \times 8760} \times 100(\%)$$

Fig. 1.2.2-1 Averaged capacity factors of light water reactors for a 13-year period [1.2.2-1]

Table 1.2.2-1 Availability of present-day power plants

	Nuclear (LWR) plants	Fossil fuel plants
General	Actual capacity factor ~ 80%(Fig. 1.2.2-1) Capacity factor = total generation power/ maximum capacity x time	85% Availability maintained Availability = time rate of the facility in available condition
Outage for Maintenance	Biannual (annual and interim) maintenance, respectively 60 days and 10 days, if no specific repair work. Actual down time is 100 days average. (Fig. 1.2.2-2)	Usual annual maintenance averages 40-60 days (maintenance of boiler ~ boiler turbine)
Forced Outage	Rate of reported events is decreasing drastically. Forced outage less than 2% (1995) Reported events 0.2/unit·year (Fig. 1.2.2-3) Early LWRs had 3% (1970-81 Sep.), 3 times (1970) - 6 times (1971), rates that were very high.	Less than 2-3% forced outage
Restraint operation	Restraint operation for changing control rod pattern reduces capacity factor 2%. (not operated at 100%)	Restraint operation for environmental purpose can occur for coal-fired plants

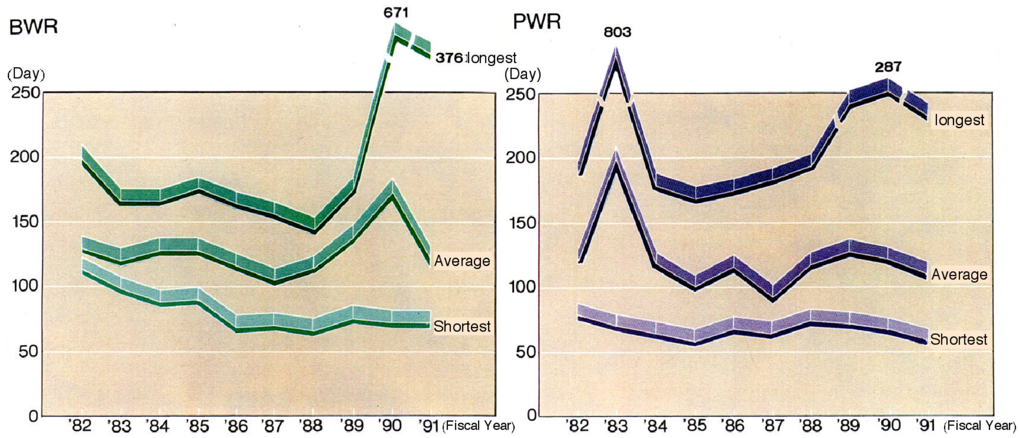


Fig. 1.2.2.2 Annual inspection down time in light water reactors [1.2.2.2]

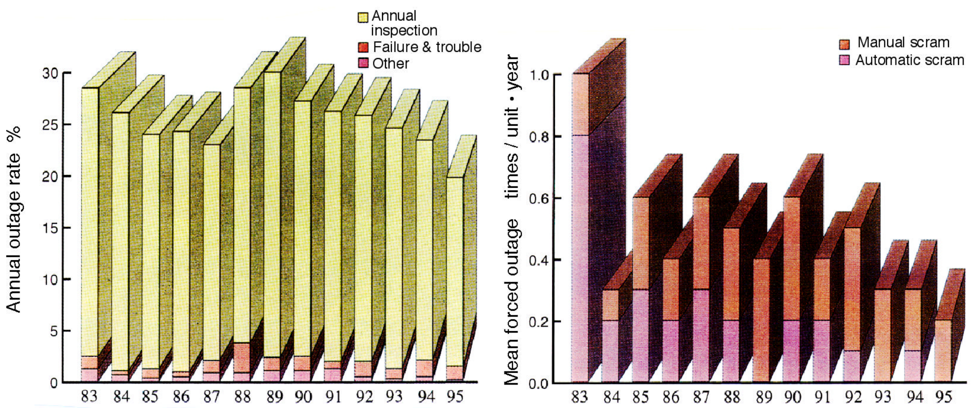


Fig. 1.2.2-3 Average annual down-time ratio and average number of unscheduled outages [1.2.2-1]

(2) Economic Efficiency

As seen in the Table 1.2.2-2, fission energy was expected to be less expensive in principle than fossil produced power in the early stages of introduction.* Similarly, it is obvious that a lower cost for fusion electricity than electricity from other sources is desirable for the commercialization of fusion reactors. If the costs are equal, other major incentives will be necessary for the electric utilities. Without such incentives, fusion energy will not be commercialized, even at equal electricity production costs. Thus, the extension of lifetime and attainment of a high capacity factor are absolutely necessary for fusion reactor plants.

(* In reality, early nuclear fission power was more costly than fossil produced power due to fission power's low availability and the low price of oil. For instance in 1967, the cost of electricity from fossil power was 2.9 yen/kWh at a 65% capacity factor while that of fission power was 7.6 yen/kWh at a 45% capacity factor).

Table 1.2.2-2 Costs of electricity from present-day power plants

	Nuclear (LWR) plants		Fossil fuel plants	
Life	Legal lifetime is 16 years, actual life until decommissioning will be ~40 years		Legal lifetime 15 years, actually maintained for much longer period	
Construction cost	(1992) 31×10^4 yen/kW	(1982, Reference) 27×10^4 yen /kW	(1992) Oil: 19×10^4 yen /kW Coal: 30×10^4 yen /kW LNG (Liquefied Nat. Gas): 20×10^4 yen /kW	(1982, Reference) 13×10^4 yen /kW 20×10^4 yen /kW
Fuel cost	(1992) 2 yen/kWh (about 20%)	(1982, Reference) 3 yen/kWh	(1992) Oil: 6 yen/kWh (60%) Coal: 3 yen/kWh (30%) LNG: 4.5 yen/kWh (50%)	(1982, Reference) 16 yen/kWh 7 yen/kWh
Cost of electricity	(1992) 8.7 yen/kWh	(1982, Reference) 6.7 yen/kWh	(1992) Oil: 10 yen/kWh Coal: 10 yen/kWh LNG: 9 yen/kWh Average: 9.6 yen/kWh	(1982, Reference) Average: 18 yen/kWh

(3) Operational characteristics

Operational characteristics of the present-day nuclear and fossil fuel stations are summarized in the Table 1.2.2-3. In Japan, it is technically possible to operate nuclear plants in a partial load mode although they are not operated in a load following mode. In general fossil-powered plants operate very well in a load following mode, but coal fired plants are usually operated in a base load mode because of high construction costs and low fuel prices.

The operational characteristics as good as that of a fossil power plant would be preferable for a fusion power plant. Base load mode operation is expected at least in early stages of fusion plant introduction, and good partial load and load following mode characteristics will not be of importance. However, in case of an accident on the electric power distribution grid, a temporary reduction of output followed by stand-by operation with the turbine by-passed would be required. For this reason, it cannot be concluded that such partial-load-mode operation would not be needed. Obviously, the protective control system that will enable the plant to shut down safely in an emergency and readily restart later is required.

(4) Site environment

In the present-day situation, both nuclear and fossil power stations are facing increasing difficulty in handling environmental problems and in maintaining public acceptance. Safety should be one of the major advantages of a fusion reactor, which means that a relaxation of siting requirements should be pursued as a high priority target to promote fusion power. To secure public acceptance, radioactive emissions from a fusion plant must be less than from a fission plant and back-end technologies must be mature. Safety in the case of off-normal events must also be assured.

(5) Others

Other criteria such as unit capacity and heat efficiency are compared in Table 1.2.2-5. Unit capacity is related to the scale of electric power grid, reserve capacity, etc. Table 1.2.2-6 summarizes the trend of nuclear power development in Asia for reference. Countries usually plan to introduce nuclear energy as they approach a total generation capacity of approximately 10 GW. Initial capacity of nuclear power is usually approximately 20% of total capacity, and unit capacity of a single nuclear plant is about 10% of the total capacity. Introduction of a power plant with a unit capacity exceeding 10% of the network (power grid) capacity is difficult from the viewpoint of the network stability. Therefore, it is desired to have no technical or economic problems that exceed a 1 GW level in a fusion reactor in these developing countries. If a unit generates more than 2 GW, 2-unit station will be selected per network capacity of 20GW. Even if the total network capacity will increase in the future, the potential areas available for introduction of an early fusion power plant will be rather limited for such large unit capacity. However, in France, where nuclear energy supplies 80% of the total generation capacity, when fusion technology matures, the introduction of fusion power is expected to far exceed 20% of the total generation capacity.

Table 1.2.2-3 Operation characteristics of present-day power plants

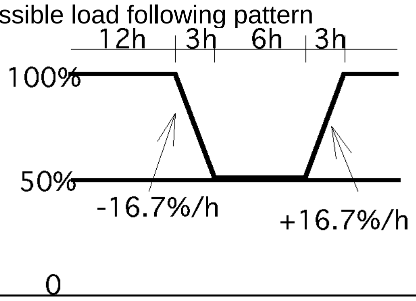
	Nuclear (LWR) plant	Fossil plant
General	Currently stable load operation by economical and legal reason, load-following is technically possible and in service in France.	Currently oil, LNG are used for most of changing load. Coal is mainly used for base load.
Start up/Partial load operation etc.	Start up time varies hot and cold start, example of PWR cold start up. Heat up -> synchronize : ca.40 hrs. Synchronize -> full power : ca.160 hrs. Total 200 hrs. Possible load following pattern 	Depending on fuel, unit, mostly as follows, Minimum load: 20-50% of maximum Start up time: (ignition -> full power) Hot start: 5 hrs. Cold start: ca. 10 hrs. Load changing (gradual change) 2% / min. (50-100%)
Off-normal events	Turbine bypass is equipped to avoid scram in the network accident. Stand-alone operation possible depending on capacity of turbine by pass.	FCB (Fast Cut Back) function available. Stand-alone operation possible.

Table 1.2.2-4 Site requirements for present-day power plants

	Nuclear (LWR)	Thermal
General	Technical site criteria, isolation from dense population (ex. Population exposure < 20000 man Sv) ・existence of firm bed rock ・obtaining cooling water are important criteria.	Criteria such as obtaining cooling water is important. Increasing importance of environmental regulation.

Table 1.2.2-5 Unit capacity of present-day generation plants and heat efficiency

	Nuclear (LWR) power plant	Fossil power plant
Unit capacity	Latest ABWR generates 1.35 GW Most of previous units are 0.5~1 GW. Next ABWR is to be 1.7 GW from economic reasons.	Up to 1 GW constructed. Recently 500~700 MW class are increasing for middle load.
Thermal efficiency	Net thermal efficiency Ca. 33% Recirculation Ca. 4%	Net thermal efficiency / Recirculation Oil 39% / 4.5% Coal 40% / 6~10% LNG 50%

Table 1.2.2-6 Nuclear energy in Asia

Nation	Total generating capacity (GW)	Nuclear generating capacity (GW)	Nuclear/Total
Nuclear existing			
Japan	227	45	0.20
Korea	32	10	0.31
Chinese Taipei	24	5	0.20
China	228	2	0.01
Nuclear planning			
North Korea	10	2 (under construction)	0.20
Indonesia	12	2 (2-3 planned)	0.16
Nuclear considering			
Thailand	15		
Vietnam	5		
The Philippines	7		

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1.2.3 Summary of the Required Specifications for Commercial Fusion Reactor

The above mentioned characteristics of present-day power plants (nuclear and fossil) and consideration of the specifications for a fusion reactor suggest the required characteristics for a fusion reactor to be used commercially. These are as summarized in Table 1.2.3-1. The right column of the table, "Target for the initial fusion reactor," summarizes the considerations of this section. The anticipated characteristics of a fusion reactor are shown considering the advantages and disadvantages of fusion power and thus not all the values are competent with, or exceed that of present electricity sources. Such values must be regarded as minimal targets of the first-generation fusion reactors.

Table 1.2.3-1 Targets of the characteristics of fusion reactor for commercial use

	Desirable for utility	Reference for fusion	Target for initial fusion reactor
Economy	30% reduction of present-day cost (~10 yen/kWh)	Initial design value of LWR 11~12 yen/kWh Costs of fossil power with CO_2 sequestration as the last solution:15~18 yen/kWh LWR with sea water uranium has ca. 11~15 yen/kWh of COE	Design value 10 yen/kWh or less desired. 15 yen/kWh as upper limit
Operation characteristics Stability fluctuation forced outage rate load following and partial load operation	 ± 0% ~ 0% ~17%/hr 100% ~ 50%	 Fluctuation on daily load curve: ~1% Forced outage: recently ca. 1.5% 0.2/unit·year in 85-90 year 0.5/unit·year	 Less than 1% 0.5/unit·year or frequency less than external causes such as lightning. At least partial load operation in case of emergency.
Siting requirements	Siting requirements more flexible than for LWR.		If possible location near high demand area
Generation capacity	1 GW class or larger Max 1.5-2 GW (In this case, demand will be limited)	ABWR:135 GW France plans 1.8 GW LWR. Economical FBR by CRIEPI targets 1 GW, but possible to design 1.5 GW.	Depends on location and situation. Replacement of LWR require less than 2 GW Desirable to be smaller
Capacity factor	Above 80%	LWR: 1975: 40% 1985: 75% 1995: 80%	Ideal design value 85% or above (non-trouble annual inspection only) Target for initial capacity factor >70%
Fuel supply and increase rate	Guaranteed fuel supply Increase as fast as LWR	History of LWR: Average ca. 15 GWe/year	Tritium stock several 10 kg. Early reactors are desirable to have TBR-1.1

TBR: tritium-breeding ratio   COE: cost of electricity

1.3 Comparison with other power plants

1.3.1 Resources

(1) Resources in mines

The amounts of mineral resources are commonly assessed as "reserve base" which is summation of "Reserve" (resources that are at an economic exploitation level in terms of amount, composition, and quality), "Marginal Reserves" (resources at a marginal economic exploitation level), and "Sub-economic Resources" (resources below an economic exploitation level). The amounts of these resources are determined in consideration of the present mineral industry technologies and thus they will change as the social or economic situations change. For example, the resource life over several tens of years of a specific mineral can remain constant as the mineral is mined as a result of discovery rates that match the mining rate of the mineral. Besides such "variable" resources, there is another definition of resources, dubbed "gross mineral resources," i.e., the amount of minerals existing in the earth's crust, which are capable of being mined with the present technologies [1.3.1-1].

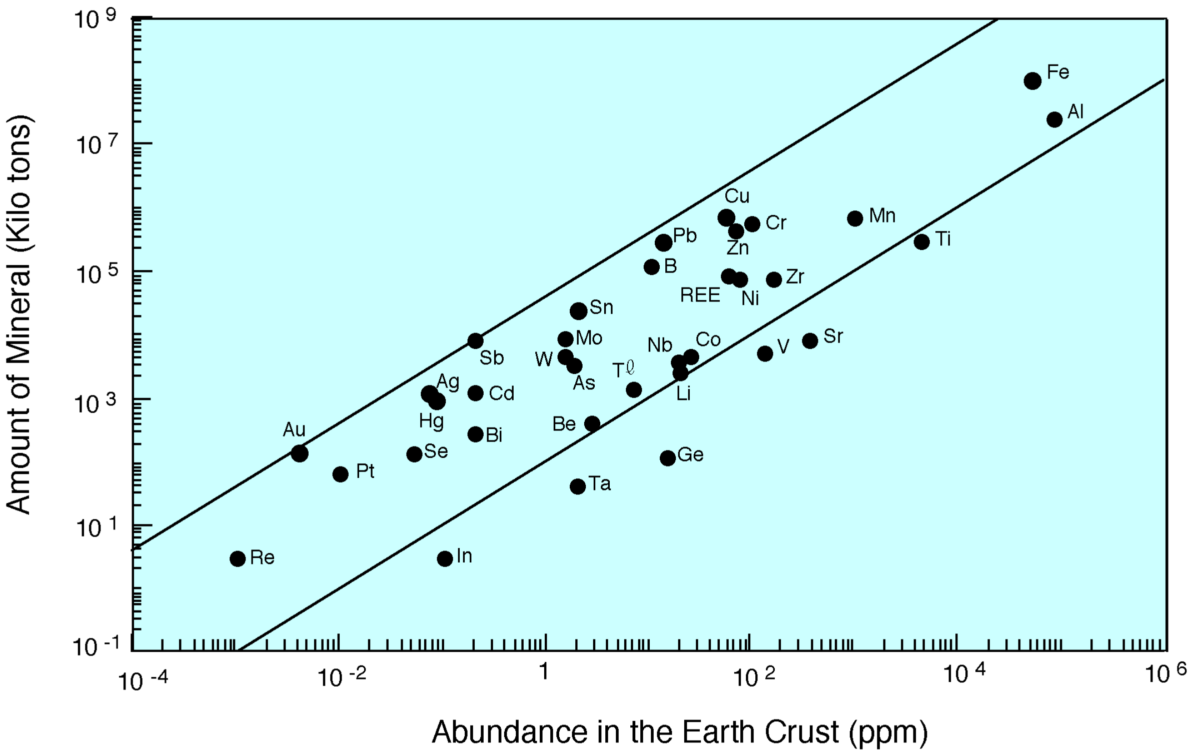


Fig. 1.3.1-1 Amount of minerals versus the abundance of the elements in earth's crust ([1.3.1-1], [1.3.1-2])

Individual minerals are known to show a distinctive trend in the logarithmic plot of the amount of minerals (including the minerals already mined) versus the average composition of elements in the earth crust (abundance in the earth crust) as shown in Fig. 1.3.1-1. This indicates a close coupling of the amount and the abundance of minerals. Gold is one of the most well identified minerals for its abundance in the earth crust, whose amount is expressed as [abundance in the crust]x$10^{13.6}$ tons.

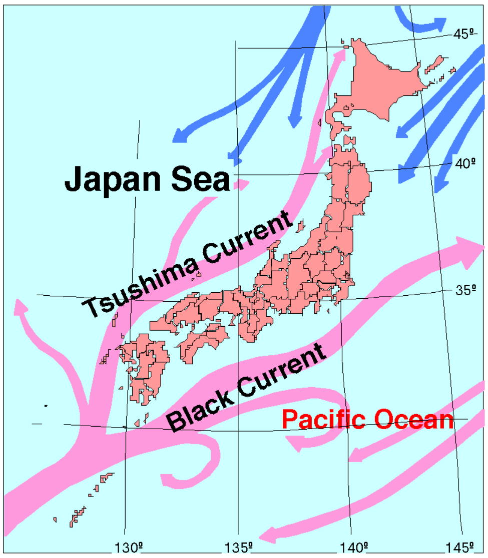
Table 1.3.1-1 Gross mineral resources, reserve base, reserves, production rate, reserve production ratio, and producing countries of main elements [1.3.1-1], [1.3.1-2]

Element	Gross Mineral Resources [1000 t] Abundancex$10^{13.6}$	Reserve Base [1000 t]	Reserve [1000 t]	Production [1000 t]	Reserve Production ratio [Year]	Main Producers
Al	3,240,000,000	28,000,000	23,000,000	114,009	202	Australia (38%) Guinea (13%)
Fe	1,990,000,000	112,000,000	68,000,000	954,900	71	China (25%) Brazil (18%)
Ti	175,000,000	440,000	270,000	3,990	68	Australia (52%) Norway (19%)
Mn	37,800,000	5,000,000	680,000	22,300	30	China (27%) South Africa (15%)
Zr	6,570,000	62,000	32,000	857	37	Australia (54%) South Africa (30%)
V	5,370,000	27,000	10,000	35	286	South Africa (46%) Russia (31%)
Cr	3,980,000	7,500,000	3,700,000	12,200	303	South Africa (41%) Turkey (16%)
Ni	2,990,000	140,000	40,000	1,010	40	Russia (22%) Canada (19%)
Zn	2,790,000	430,000	190,000	7,226	26	Canada (17%) China (14%)
Cu	2,190,000	630,000	320,000	10,756	30	Chile (28%) USA (18%)
Co	995,000	9,000	4,000	27	148	Zambia (29%) Canada (21%)
Nb	796,000	4,200	3,500	16	219	Brazil (85%) Canada (15%)
Li	796,000	9,400	3,700	21	176	Bolivia Chile
Pb	517,000	120,000	65,000	2,738	24	USA (16%) China (15%)
B	398,000	470,000	170,000	3,250	52	Turkey (48%) USA (36%)
Be	111,000	800	421	0.35	1200	USA (84%) Russia (14%)
Sn	79,600	12,000	7,700	206	37	China (26%)
Mo	59,700	12,000	5,500	127	43	USA (44%) China (20%)
W	59,700	3,300	2,100	31.9	66	China (28%) Russia (9%)
Bi	6,930[1]	260	110	4.21	26	Mexico (39%) Peru (24%)
Ag	2,790	420	280	14.5	19	Mexico (17%) Peru (13%)
Au	159	72	45:including accumulation 154	2.25	20	South Africa (22%) USA (14%)

(2) Resources in oceans

Japan is an assembly of islands surrounded by oceans rich in minerals, and ocean currents circulate these abundant minerals constantly. Figure 1.3.1-2 shows the currents around Japan, the amount of usable metal resources in all oceans, and the resources delivered by the Black Current.

Driscoll of the Massachusetts Institute of Technology has suggested the economically usable elements existing in oceans based on the market price versus the concentration in seawater (Fig. 1.3.1-3) [1.3.1-4]. Lithium, used to produce one of the fuels (tritium) for fusion reactors, is categorized as an economically usable element. Over the past several years, the technology for extracting the usable metals from seawater has developed rapidly. The extraction method using amidoxime adsorbent (applicable to uranium, vanadium, cobalt, and titanium) developed by the Takasaki Establishment, JAERI [1.3.1-3] and the ion screening crystal method (applicable to lithium) developed by the Shikoku National Industrial Research Institute (the Agency of Industrial Science and Technology) are both approaching commercial use.



	Concentration (ppb)	Resource in Seawater (10^8 tons)	Resource Delivered by Black Current (M tons / y)	Resource Traversing 1m^2 (tons / y)
Li	170	2330	268	8.5
Mo	10	140	15.8	0.5
Ni	6.6	92	10.4	0.3
U	3.3	46	5.2	0.16
V	1.9	27	3	0.09
Ti	1.0	14	1.6	0.05
Mn	0.4	5.6	0.6	0.02
Co	0.39	5.5	0.6	0.02
Ag	0.28	3.9	0.4	0.01
Au	0.01	0.14	0.015	0.0005

Fig. 1.3.1-2 Ocean currents around Japan, useful metal resources in seawater and resources carried by Black Current [1.3.1-3]

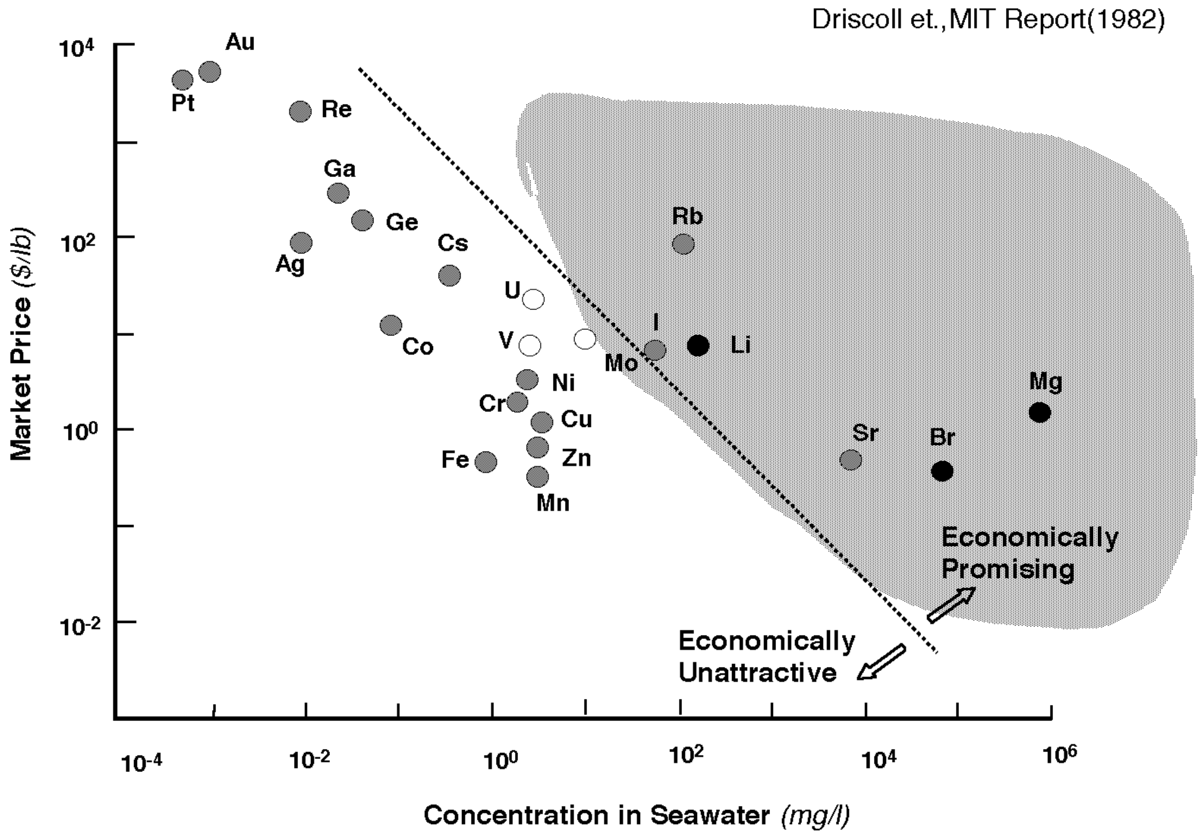


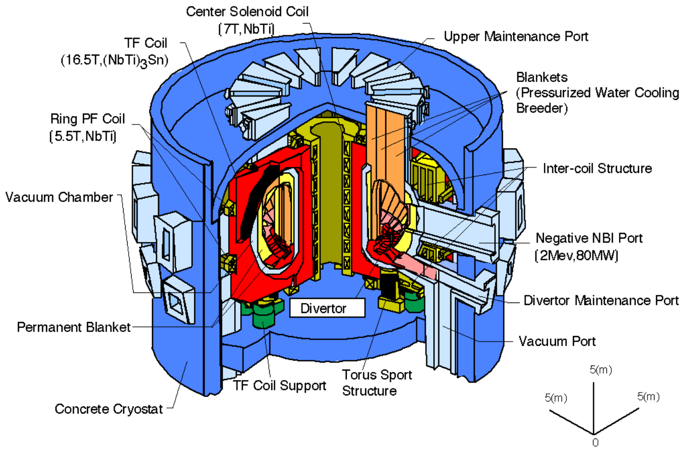
Fig. 1.3.1-3 Economic assessment of the extraction of minerals from seawater viewed from mineral concentration in seawater and market price [1.3.1-4]

1.3.1.1 Resources required for fusion reactors

Resources required for fusion reactors are surveyed in this section. Using the Steady State Tokamak Reactor (SSTR) [1.3.1-5] as a reference, the types and weights of typical metal resources are estimated. The SSTR is designed to have a fusion power output of 3 GW and generate an electrical output of 1.08 GW, as shown in Fig. 1.3.1-4. This figure also includes a cutaway view of SSTR and lists the main design parameters. Table 1.3.1-2 lists the main structural components of SSTR, their functions, and the typical metal resources and weights needed. The reactor requires a large amount of stainless steel and low-activation ferritic steel. Special materials are:

- Lithium for tritium breeding (~97 tons),
- Beryllium (Be) for multiplying neutrons (~110 tons),
- Niobium for super conductors (~207 tons), etc.

Deuterium is required as well. The SSTR's design needs a relatively large amount of Be to achieve the required tritium-breeding ratio (TBR) of 1.2. In contrast, commercial fusion reactors operated at a TBR of about 1.05 should require only half the Be required for the SSTR.



Cutaway view of SSTR

		Design Value
Plasma Current	I_p	12MA
Toroidal Field Coil	B_t	9T
Major Radius	R	7m
Aspect Ratio	A	4.1
Elongation	κ_{95}	1.85
Normalized Beta	β_N	3.5
Fusion Output	P_F	3GW
Current Drive Power	P_{CD}	60MW
Net Electric Output Power	P_E	1.08GW
Fusion Gain	Q	50
Averaged Neutron Wall Load	$P_{neut.}$	$3MW/m^2$

Fig. 1.3.1-4 Cutaway view of Steady State Tokamak Reactor (SSTR) and the main parameters

Table 1.3.1-2 Components of SSTR (Steady-State Tokamak Reactor)

Parts	Function	Material	Weight
Exchangeable Blanket	Tritium Breeding Energy Conversion	Low-Activation Ferritic Steel Lithium Oxide(Li_2O) Beryllium	~ 250 tons ~ 23 tons (Li) ~ 110 tons
Permanent Blanket	Tritium Breeding Energy Conversion	Low-Activation Ferritic Steel Lithium Oxide (Li_2O)	~ 720 tons ~ 74 tons (Li)
Divertor	Heat Removal	Low-Activation Ferritic Steel etc.	~ 360 tons
Vacuum Chamber	Vacuum Boundary, Radiator Shield	Stainless Steel and/or Low-Activation Ferritic Steel	~ 10,000 tons
Toroidal Field Coil	Magnetic Field Generation	Stainless Steel for low temperature Nb_3Sn or Nb_3Al strand NbTi strand	~ 10,000 tons ~ 150 tons (Nb) ~ 16 tons (Nb)
Poloidal Field Coil	Magnetic Field Generation	Stainless Steel for low temperature NbTi strand	~ 2,200 tons ~ 41 tons (Nb)

In the 1970s, numerous surveys were conducted to estimate the resources required for conventional fusion reactors. The surveys point out that most materials used in D-T fusion reactors (reactors, which use deuterium and tritium as fuel,) are available in sufficient quantities but that helium, lithium, molybdenum, niobium, beryllium, and lead can be problematic. Later a survey performed by the Atomic Energy Society of Japan [1.3.1-6] confirmed that the resources of helium and molybdenum are adequate. Used as a radiation shield, lead can be replaced by other materials. Thus, the resources of lithium, niobium, and beryllium as special materials, of deuterium as fuel, and of vanadium as blanket structure material should be assessed.

The following assessment of the years in which fusion energy plants would be usable assumes that approximately 1,500 of SSTR-relevant fusion reactors would be built to supply the present electricity demands of the world (~1.25×10^7 GWh in 1993).

(1) Deuterium (D_2) [1.3.1-7]

The concentration of deuterium is approximately 158 ppm in seawater and ~144 ppm in fresh water. This resource is thought to be inexhaustible. Heavy water, D_2O, is extracted from fresh water mainly by the Girdler-Spevack (GS) method. This method contacts H_2S (hydrogen sulfide) and H_2O (water) counter-currently to produce heavy water via hydrogen sulfide. This uses the reactions replacing the hydrogen of water by deuterium (H_2O + HDS --> HDO + H_2S) activated at a room temperature, and replacing hydrogen of hydrogen sulfide by deuterium (HDO + H_2S --> H_2O + HDS) activated above 100ºC. Since the method is based on equilibrium chemical reactions, little energy is consumed in the process. At present, chemical plants in Canada have the capacity to provide 800 tons of D2/year for heavy water reactors, which is 11,000 times the amount of deuterium fuel required for a one-year operation of a 1 GW fusion power plant.

(2) Lithium (Li) [1.3.1-8~10]

Tritium, the other fuel in D-T fusion reactors, scarcely exists in nature. The element is produced by the nuclear reaction of a lithium nucleus and a neutron in the tritium-breeding blanket of the fusion reactor, as shown in Fig. 1.3.1-5.

Reserve base of lithium : 9.4 M tons (Reserve: 3.7 M tons) (Table 1.3.1-1)
Gross mineral resources : 800 M tons (Exploration rate* estimated about 1%)
Resources in seawater : 233 G Tons
Annual production : 21 kilo tons (1996)
*Exploration rate = reserve base/gross mineral resources

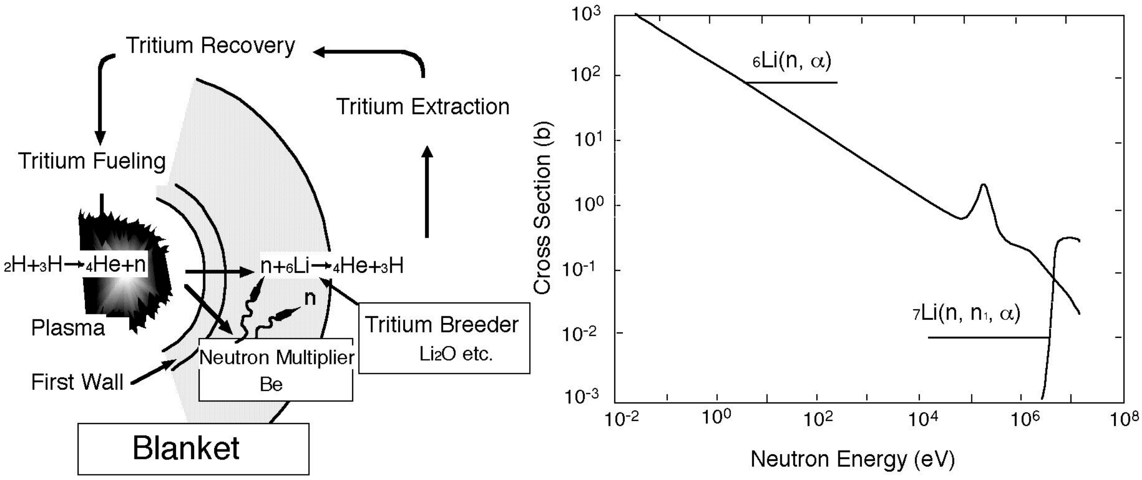


Fig. 1.3.1-5 Principle of tritium production in the blanket, and the cross sections of ^6Li and ^7Li for tritium production. The natural abundance of lithium is 7.4% of ^6Li and 92.4% of ^7Li. The reaction of ^6Li is an exothermic reaction, and that of ^7Li is endothermic. As seen in the figure, the reaction of ^6Li has 1/v dependence and that of ^7Li is a threshold reaction. Thus, ^6Li will exhibit higher burn up. For example, after 3-year operation of SSTR, over 50% of ^6Li should burn in the replaceable blanket.

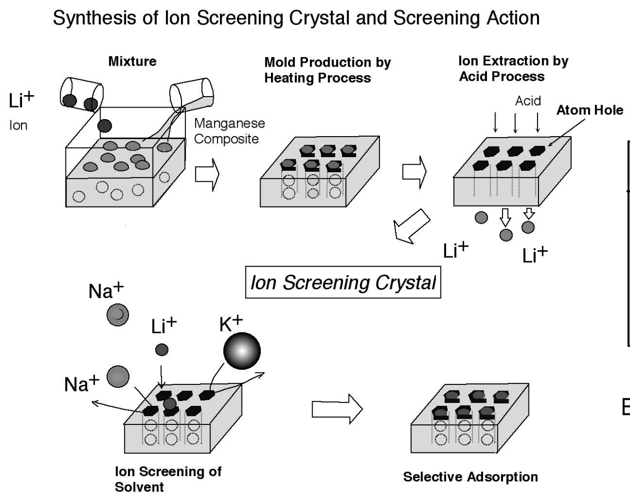
During the 30-year operation of a fusion reactor, the lithium that would be consumed is estimated to be 23 tons x 10 = 230 tons for the replaceable blankets and 74 tons for the permanent blankets, which is a total of 304 tons. The average consumption per year is estimated to be 10 tons. Thus the annual consumption of lithium becomes 15,000 tons and the resource life is estimated to be 600 years in the reserve base, ~50,000 years in the gross mineral resources and 1.5 million years in seawater, presuming that 1,500 fusion reactors, each with an output of 1 GW, are operational.

Present status of lithium extraction technology from seawater

As indicated by Driscoll's estimates in Fig. 1.3.1-2 and 1.3.1-3, the lithium concentration in seawater is noticeably high, suggesting that extraction from seawater will be economically feasible. The Shikoku National Industrial Research Institute has performed research on lithium extraction from seawater using an adsorbent form of manganese oxide. The results achieved a high enrichment, up to 10^5 cm^3/g, by the ion screening technique. Figure 1.3.1-6 depicts the conceptual picture and the characteristic enrichment of the adsorption technique. As illustrated, large ions such as sodium (Na) and potassium (K), are unable to traverse the atom holes in the manganese oxide adsorbent, while the lithium (Li) ions enter the holes and are adsorbed.

For lithium extraction from seawater, the assembly containing the adsorbent is submerged in flowing seawater and lithium is collected in the form of Li_2CO_3. The seawater flow can be obtained from the warm drain of a power station (230 tons of the lithium compound per year), by an extraction vessel operating in the ocean (125 tons per year), or by the outflow water of a wave-power generating station (50 tons per year). The cost for extraction is estimated to be 3,700 yen/kg-Li (700 yen/kg-Li_2CO_3) in the case of the extraction vessel [1.3.1-9], about twice today's market price (400 yen/kg- Li_2CO_3).

The radiation graft polymerization method of polyethylene/polypropylene polymer, which has been developed for the extraction of uranium from seawater, is considered applicable for lithium extraction. At any rate, the lithium extraction from seawater would be economical with further effort if the demand for lithium elevated as an important resource.



Characteristic Adsorption of Metal Ions in Seawater

Element	Adsorption(mg/g)	Enrichment(cm^3/g)
Li	17	100,000
Na	1.2	0.1
K	0.3	0.8
Mg	2.7	2.0

Enrichment = Adsorption / Concentration in Seawater

Fig. 1.3.1-6 Principle of lithium ion extraction by the ion screening method (developed by The Shikoku National Industrial Research Institute) and the characteristic adsorption of metal ions in seawater [1.3.1-8]

(3) Beryllium (Be)

Fast, 14 MeV, neutrons produced by the D-T fusion reaction can produce tritium (T) when they undergo a nuclear reaction with lithium in the blanket of a fusion reactor. But, some neutrons are lost by the interaction with structural material. It is then effective to locate neutron multiplier near the first wall in increasing tritium breeding ratio (TBR). It should enhance tritium production as explained by the tritium breeding ratio (TBR). Possible multiplier materials for neutrons are beryllium and lead (Pb). Lead is comparatively well explored and, consequently, the reserve base is as ample as 120 M tons (Table 1.3.1-1, 23% of the gross mineral resources). Here, beryllium resources will be discussed. As indicated in Table 1.3.1-1, the main producers of beryllium are the US and Russia and the resources are as follows.

Resources of beryllium (reserve base): 0.8 M tons (1985)
Gross mineral resources: 100 M tons (exploration rate* estimated about 1%)
Annual Production: 350 tons (1997)

If (1500 units of) fusion reactors produced the world's total supply of electricity, they world require 165 kilo tons of beryllium (20% of the reserve base). Presuming the reuse of beryllium, the consumption of beryllium is then measured by the burning rate. Since the burning rate is 1 ton/FPY (full power year) for an SSTR, the consumption of beryllium would be 1,500 tons per year. Thus, the resource life is estimated to be 430 years on the reserve base, seemingly short. Considering the extremely low exploration rate, beryllium could be supplied much longer if the demand for beryllium increased and thus exploration increased, since the resource life is estimated to be 70,000 years based on gross mineral resources.

When lead is used for the neutron multiplier, the supply is virtually inexhaustible, even on the reserve base. In addition, it is possible to design of a fusion reactor that does not require a neutron multiplier by using liquid lithium. The above results show that the beryllium resource does not limit the supply of fusion energy.

(4) Niobium (Nb)

Nb_3Sn or Nb_3Al strands are used in the superconductors of a fusion reactor at present. Thus, niobium could be an essential resource. On the other hand, it should be noted that the possibility of niobium being replaced by other materials is increasing as seen in the recent development of high temperature superconductor using bismuth (Bi). The main producers of niobium are Brazil and Canada. The resources are as follows.

Reserve base of niobium : 4.2 M tons
Gross mineral resources : 700 M tons (exploration rate *, less than 1%)
Annual Production : 16 kilo tons (1996)

When niobium is not reused for superconductors, the fusion reactors producing the entire world's supply of electricity require 310 kilo tons of niobium (7% of the reserve base); i.e. 207 tons for an SSTR. Presuming the construction rate of 1,500 reactors/30 years, i.e., 50 reactors/year, the resource life of niobium is 400 years on the resource base. If 90% of the used niobium is reusable, the resource of niobium meets the demand for several thousands of years, even on the resource base. On the gross mineral resource basis, niobium resources are available for ~70,000 years without reuse.

(5) Vanadium (V)

Besides low-activation ferritic steel and SiC/SiC composite material, a vanadium alloy is a proposed candidate for the blanket structural material in combination with a liquid lithium coolant. In fact, ARIES-RS, a fusion reactor designed by the United States adopts a vanadium alloy (V-4Cr-4Ti). A fusion reactor with a vanadium alloy has the advantage of not requiring beryllium as neutron multiplier.

Although vanadium resources are ample, 5,370 M tons in the gross mineral resources base, the reserve base remains as low as 27 M tons, 0.5% of the gross mineral resources. If selected for use, 3.6 M tons of the vanadium alloy would be required to operate 1,500 fusion reactors for 30 years (0.9 M tons + 2.7 M tons = 3.6 M tons). Here it is assumed that a fusion reactor uses 600 tons of vanadium alloys in the blanket and that 200 tons of the alloys are replaced every 3 years. This leads to the resource life of 225 years for vanadium in the reserve base. Resource life of vanadium for the gross mineral resources is 45,000 years. Furthermore, the ocean contains a huge amount of vanadium, 2,700 M tons. The Japan Atomic Energy Research Institute (JAERI) reports that the extraction of vanadium from seawater exceeds that of uranium using an amidoxime adsorbent [1.3.1-1].

(6) Material prices

Lithium, beryllium, and niobium are expensive (especially beryllium) because the production is as low as 350~21,000 tons/year. However, the price is inversely dependent on the quantity of production, as indicated in Fig. 1.3.1-7. Thus, an increase in the demand for these materials for fusion use will stimulate the exploration, probably resulting in a high production and a cost reduction.

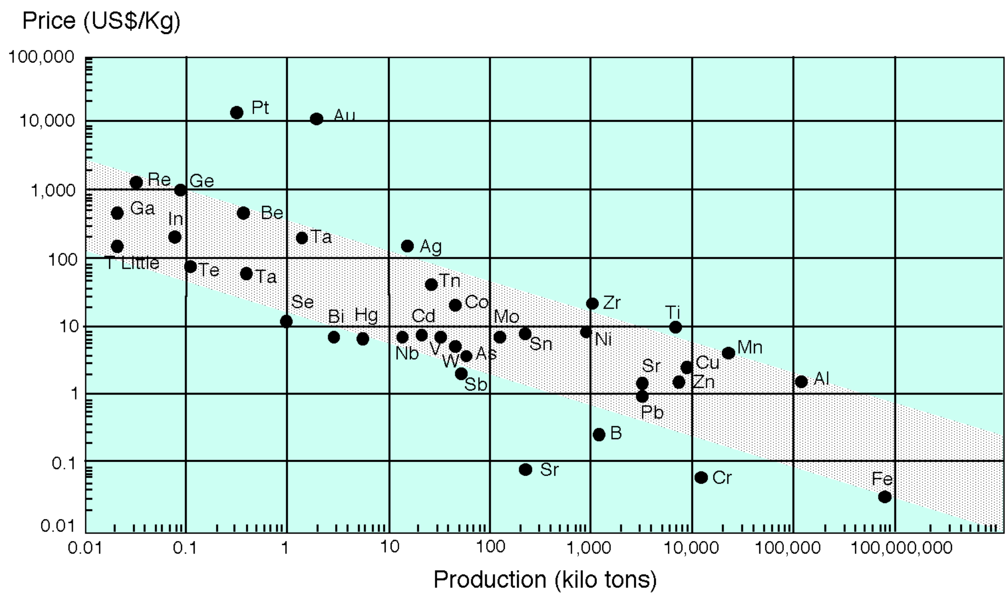


Fig. 1.3.1-7 Relation of price and production of mineral resources [1.3.1-3], [1.3.1-4]

1.3.1.2 Energy Resources

Resources of fossil fuels:

The world's energy consumption is about 8 G Tons in oil equivalent. Of this, fossil fuels comprise 90% of that total (petroleum 40%, coal 27%, and natural gas 23%).

Surveys in 1996 by the World Energy Conference (WEC), the International Energy Agency (IEA), and other international organizations report that the resource life based on reserves of coal, natural gas, and petroleum is 231, 63, and 44 years, respectively. This indicates that humankind will not face total exhaustion of fossil fuels for 200 years if the use of these fuels continues at the present consumption levels. As discussed in the previous section, the resource life based on reserves (so called reserve production ratio) is defined based on present technologies and prices and will change according to social and economic situations. For example, the reserve production ratio of petroleum evolved from 22 years in 1943, to 35 years in 1970, and to 42 years in 1995. That of natural gas also increased from 46 years in 1975 to 62 years in 1995.

In terms of petroleum and natural gas, Rogner reported the long-term demand and supply for these fuels under the collaboration of IIASA and WEC [1.3.1-11].

To illustrate natural resources, the McKelvey Box is often used, which is defined by the degree of geological assurance and the degree of economic feasibility (Fig. 1.3.1-8). He regards the hatched regions (Resource Base = Reserves + Resources) in the McKelvey Box as the usable resources in the 21st century by speculating on the technological and economical development in the 21st century. The definitions of categories are indicated in Table 1.3.1-3.

Non-conventional oil resources include oil shale (kerosene shale), tar sand (natural bitumen: mineral veins of heavy oil with high sulfur content in relatively shallow sand layers), heavy oil, and deep-sea oil fields.

Non-conventional natural gas resources are shale gas from the Devonian period, tar sand gas, underground aquifer, coal bed methane (gas in coal layers), methane hydrate, and deep layer gas.

Rogner estimated the resources of petroleum and natural gas in each category as shown in Table 1.3.1-4. He concluded that the resources in Category I - VI are usable technologically and economically in the 21st century without a significant price rise (based on international market prices) by assuming reasonable technological progress following previous trends. Considering the consumption of petroleum including non-conventional resources was 3.37 G tons in 1994 and that of natural gas was 1.87 G tons on the conventional resource basis, the resource lives in this Resource Base are 242 and 452 years for petroleum and natural gas, respectively. These resource lives are much larger than those based on reserves, 45 years for petroleum and 69 years for natural gas.

On the other hand, it is widely understood that considerations of the global environments will be the main restriction determining the demand and supply of fossil fuels as described in the following section 1.3.2.

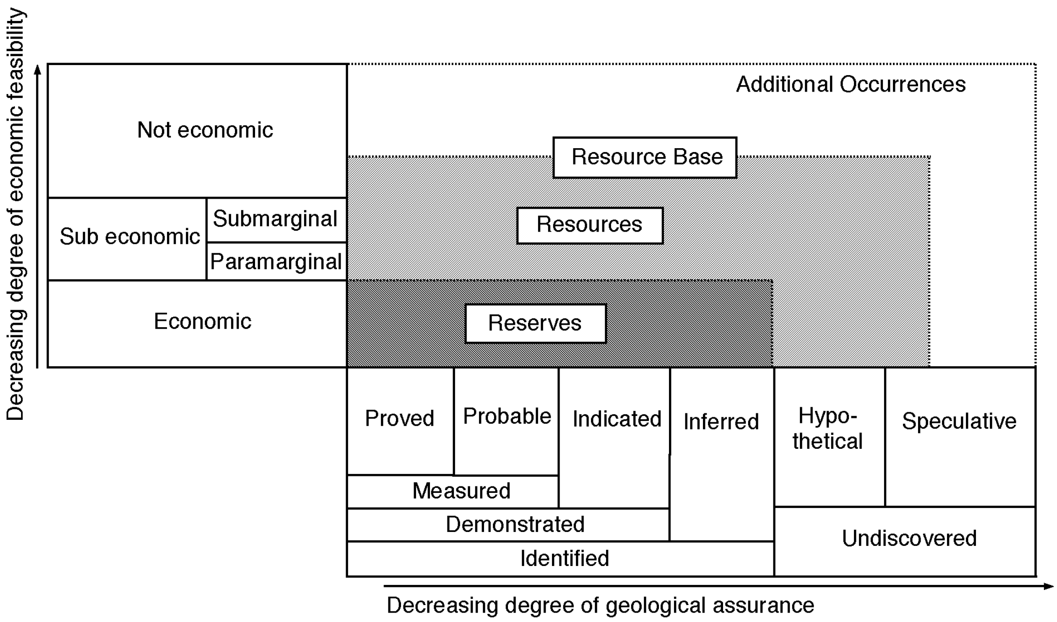


Fig. 1.3.1-8 McKelvey Box on mineral resources

Table 1.3.1-3 Categories of mineral oil and natural gas

	Definition in McKelvey Box	Rogner's Category	Definition of Category
Conventional	Reserves	CATEGORY I	Confirmed reserves. R/P ratio is usually estimated with these reserves.
	Resources	CATEGORY II	Undiscovered oil and gas that are likely to exist in a conventional sense. These resources will be included in CATEGORY I when exploration and mining technologies are developed in the future.
		CATEGORY III	The resources in this CATEGORY have higher uncertainly than CATEGORY II from the point of view of geology and technology.
Enhanced Recovery (Unconventional)	Reserves	CATEGORY IV	Potential reserves of enhanced recovery, corresponding to the reserves exceeding the recovery ratio for CATEGORY I – III (40%, actual result 34%).
Unconventional		CATEGORY V	Unconventional confirmed reserves that will be producible when economically feasible in future.
	Resources	CATEGORY VI	Estimates of unconventional resources of oil and gas.
	Additional Occurrences	CATEGORY VII	
		CATEGORY VIII	The other all oil and gas stored underground, that is considered to be difficult to extract before the end of 21st century

Table 1.3.1-4 Resources of petroleum and natural gas defined by Rogner

Categories		Petroleum (10^8 tons)	Natural Gas (10^8 tons oil equivalent)
Conventional	Category I	1,500	1,290
	Category II	610	1,120
	Category III	840	1,530
Unconventional	Category IV	1,380	560
	Category V	450	1,380
	Category VI	3,360	2,580
	Category VII	5,870	3,870
	Category VIII	12,370	187,590
Summation of I-VI		8,140	8,460
Summation of I-VIII		26,380	199,900

Nuclear energy resources:

The introduction of nuclear fission energy had been expected to be an alternative energy form for fossil fuels. In comparison with fossil fuel power generation, fuel consumption is small in nuclear power generation. A 1 GW power station annually consumes about 1 ton of uranium from the 30 tons of uranium contained in fuel assemblies. In contrast, a coal fired fossil fuel power plant burns about 2.4 M tons of fuel. The reserves for nuclear fission power generation are ~ 4.5 M tons, and the resource life is 73 years when the reserves are divided by 61.4 kilo tons, the demand in 1995. By including the unexplored (undiscovered) amount of uranium (U), the potential reserves become ~15 M tons, corresponding to 250 years of the resource life [1.3.1-1]. On the other hand, the oceans contain 3.3 mg of dissolved uranium per 1 m^3 of seawater, and the total U contained all oceans is estimated to be about 4.6 G Tons. If all uranium in seawater were extracted, the resource life would be 75,000 years. Thus, nuclear fission power generation will be considered an infinite energy source when an economical technology for extracting uranium from seawater [1.3.1-3] is established.

A promising technique for uranium extraction is an amidoxime adsorbent developed by the Japan Atomic Energy Research Institute, Takasaki Research Establishment. The development of the adsorbent produced by radiation graft polymerization was begun in 1981. The experiment performed at the offshore of Mutsu Sekinehama, from 1995-1998, resulted in the extraction of 16 g of yellow cake (that contained 10 g of uranium) and about 20 g of vanadium oxides. A follow-on, large-scale experiment is now under way.

The practical use of fast breeder reactors contributes to expanding the reserves needed for nuclear fission power generation by using uranium-238, which comprises 99.3% of natural uranium. (LWRs use uranium-235, rather than uranium-238, as their source of energy.) Moreover, thorium is a potential nuclear fission fuel, although it is not used at present.

Thus, it is widely accepted that nuclear fission power is virtually limitless when economical uranium extraction from seawater becomes feasible or when the use of fast breeder reactors becomes in use.

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1.3.2 CO_2 Emissions and Sustainability of Atmosphere

1.3.2.1 Global Warming

According to reports summarized by international organizations such as WEC and IEA in 1996, the reserve production ratio for fossil fuels is 231 years for coal, 63 years for natural gas, and 44 years for oil. On the basis of this report, humankind would not face an energy shortage for 200 years assuming our consumption rate remains unchanged, as indicated by the Central Research Institute of Electric Power Industry in Fig. 1.3.2-1. [1.3.2-1].

However, the world's annual energy consumption presently corresponds to about 8 G Tons of oil equivalent. This consumption emits the following air pollutants, 22 G Tons of carbon dioxide (CO_2), 0.13 G Tons of oxides of sulfur, and 0.09 G Tons of oxides of nitrogen. Among them, the CO_2 released to the atmosphere (6 G Tons of carbon equivalent) is regarded as a chief cause for global warming.

Dramatically reducing the present consumption of coal, which has the highest CO_2 emission rate, will result in an energy shortage in the middle of the 21st century (Fig. 1.3.2-1). Even if the reserves of oil and natural gas are increased by exploration in the future, CO_2 emission units (defined later) for oil and LNG (Liquefied Natural Gas) are much larger than those for other energy sources. Therefore, the use of fossil fuel might be restricted by the environmental constraint.

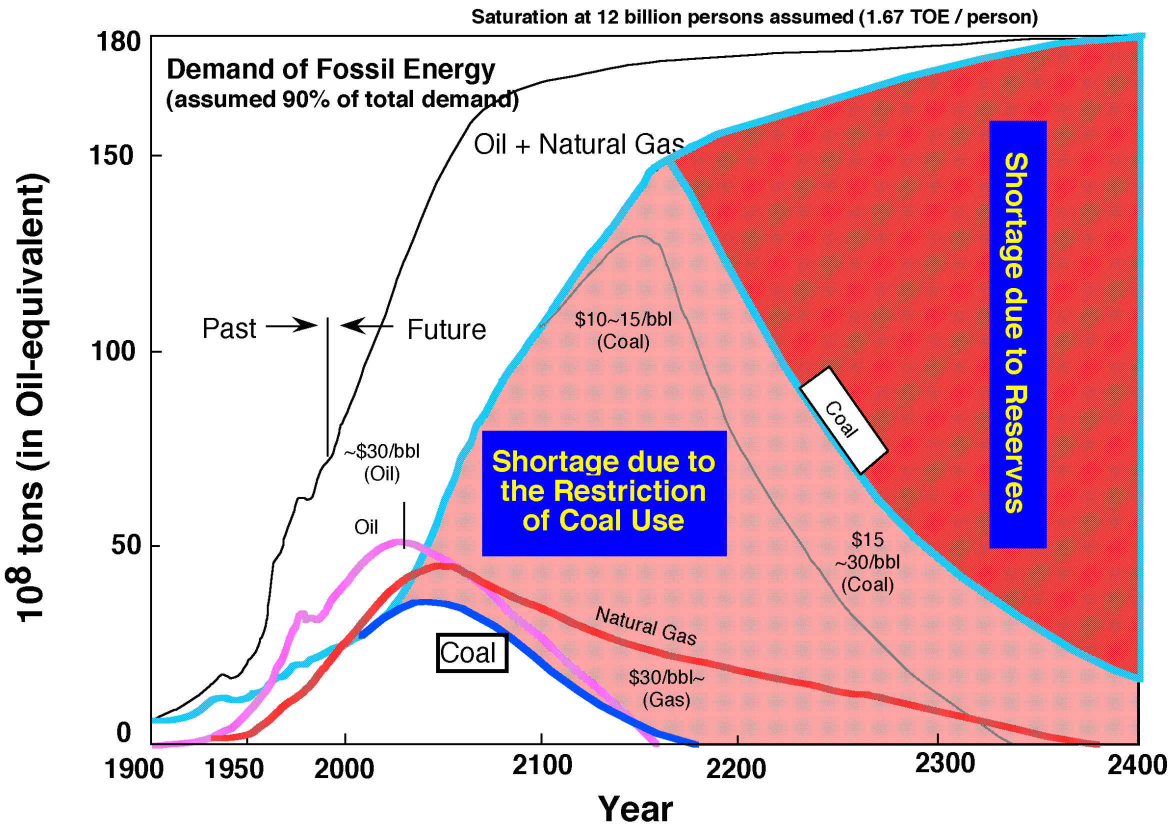

Fig. 1.3.2-1 Consumption curve of fossil fuels where 90% of the world's energy is supplied by fossil fuels. This projection assumes the world's population saturates at 12 billion and that each person consumes fossil fuel energy equivalent to 1.7 tons of oil a year (which is one-third the present consumption in developed countries) as a result of energy saving measures [1.3.2-1]. The energy supply shortage if coal use is restricted in consideration of the global environment is also shown.

Forewarned by the measurement of increasing concentrations of CO_2 in the air at the top of Mt. Mauna Loa in Hawaii (Fig. 1.3.2-2), a key global environment problem has been brought to the attention of the public—and one which the human race faces before the exhaustion of fossil fuels. Specifically, the increasing concentration of CO_2 in the atmosphere is causing significant undesirable changes to our earth.

Countermeasures for this environment problem have been enacted globally and are known as a framework treaty of climate change, which was established under the United Nations. The second assessment report of the Intergovernmental Panel on Climate Change (IPCC) issued in 1995 [1.3.2-2] estimates that the average surface temperature on the earth rose by 0.3-0.6ºC during the period from the latter half of the 19th century to the present. The IPCC also estimates from recent reliable data that the average surface temperature increased by 0.2-0.3ºC in the past 40 years. The temperature rise in the past 100 years is explained well by considering volcanic and solar activity as well as the greenhouse effect resulting from the increase in the concentration of atmospheric CO_2. The IPCC concluded that global warming is already happening--and they noted that the major cause is the increased atmospheric emission of greenhouse gasses, in particular, CO_2 [1.3.2-3].

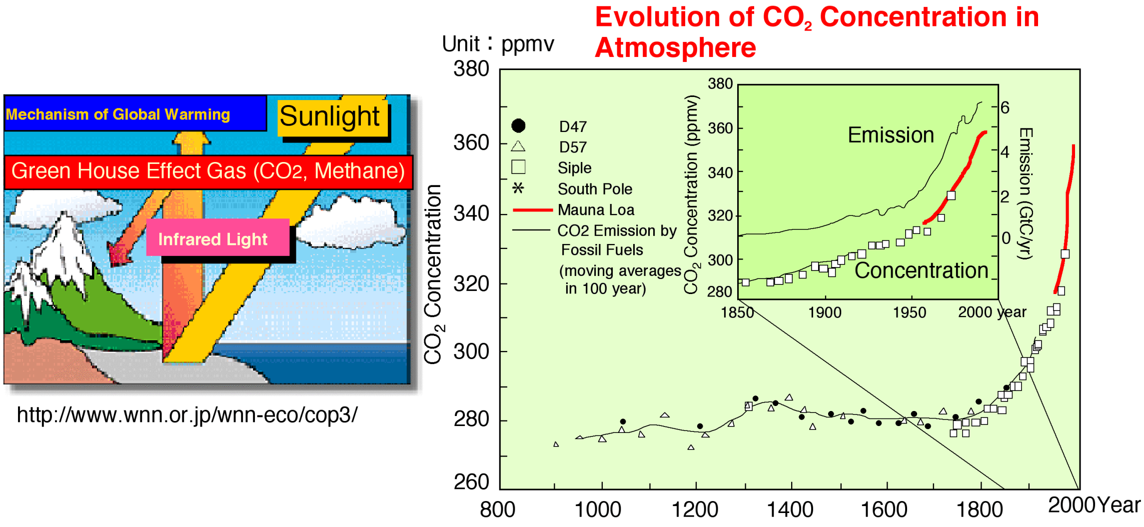

Fig. 1.3.2-2 Atmospheric CO_2 concentration for the past 1,000 years, which was measured with samples from ice sheets (D47, D57, and siple the South Pole). Also shown is that observed since 1958 at Mt. Mauna Loa in Hawaii (IPCC95) [1.3.2-2, 1.3.2-4].

The temperature rise of air is dependent on the atmospheric concentration of CO_2, which is determined from the integration of the CO_2, emitted in the past. The rising sea level is on a slower time scale than that for the temperature rise due to the combined influence of the atmosphere, the oceans, and the continents (Fig. 1.3.2-3). These global changes are gradual and, almost irreversible.

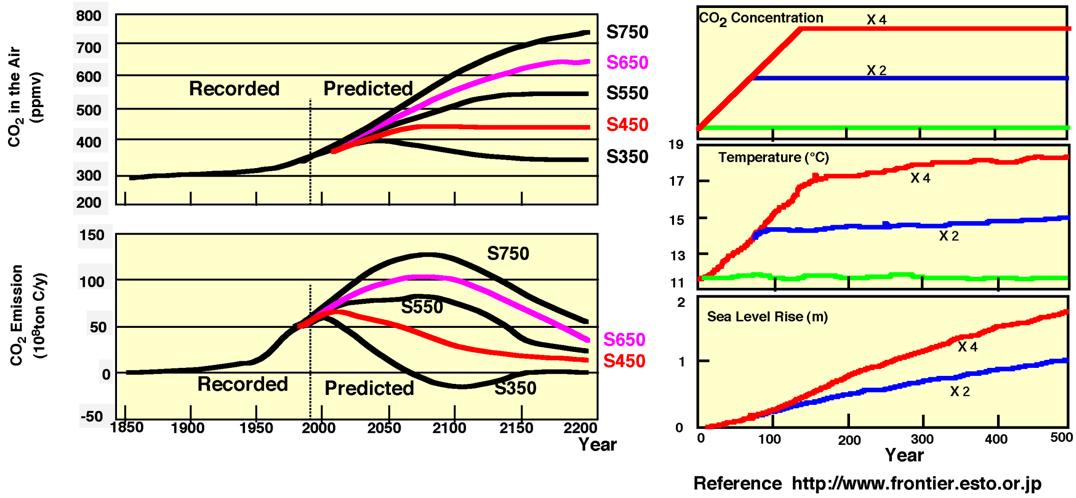

Fig. 1.3.2-3 Atmospheric CO_2 emissions and CO_2 concentration measured and predicted by IPCC. Also shown is the global warming prediction and the resulting sea level rise reported by Dr. Manabe (Earth Frontier Project)[1.3.2-2], [1.3.2-5].

Figure 1.3.2-3 shows the previous results since 1850 and predictions for the future for CO_2 emissions and its concentration in the air. If the atmospheric CO_2 concentration is to be stabilized at 55ppmv, which is about twice that at the inception of the industrial revolution, the CO_2 emissions must be curbed below about 23.8 G Tons (8 G Tons carbon equivalent) by the end of 21st century and then be reduced dramatically after that. If the target for atmospheric CO_2 concentration is 450 ppmv (to restrict the atmospheric temperature rise to below 1ºC), the reduction of CO_2 emissions should be started early in the 21st century. Further, the emissions should be reduced to half the present level, 11G Tons (3G Tons carbon equivalent) by the end of the 21st century. Consequently, energy consumption must dramatically be reduced because presently 90% of the energy consumed is produced from fossil fuels.

Furthermore, the impending rise in sea level should be a serious concern for Japan. Assuming a 1-m sea level rise, as indicated in Fig. 1.3.2-4, most of all coastal areas in Tokyo will be below sea level. The cost required for the countermeasures for all coastal areas in Japan is estimated to be 12 trillion yen [1.3.2-6].

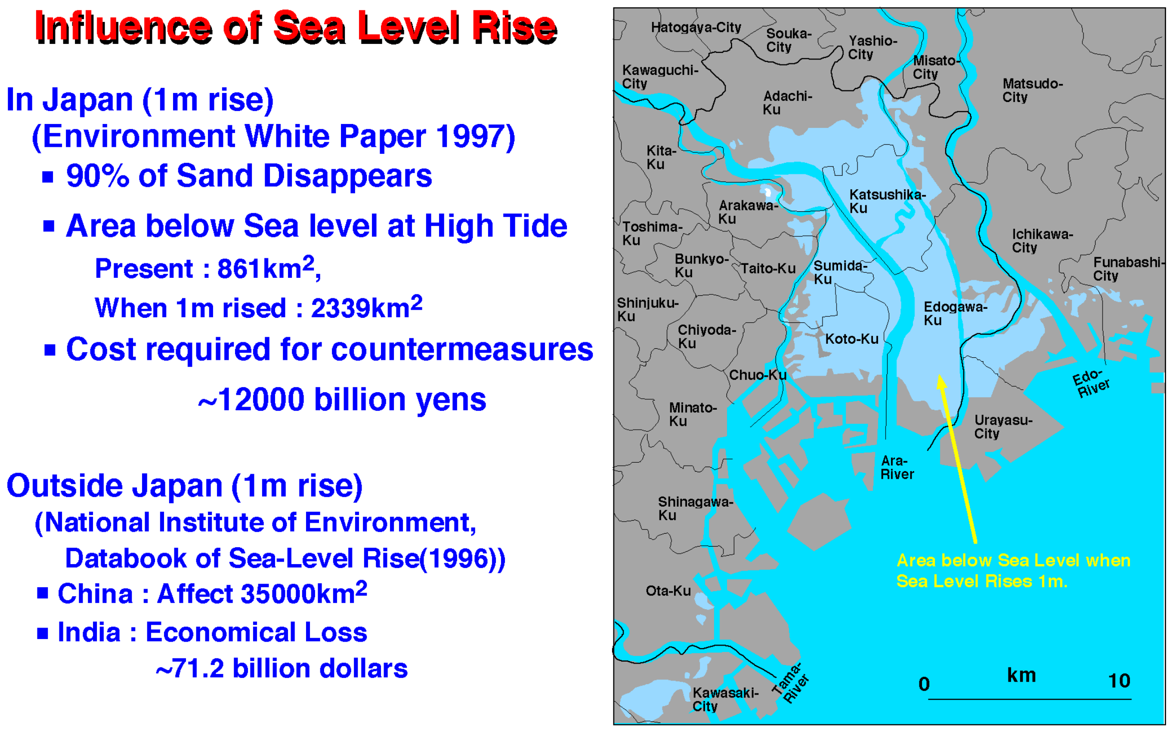

Fig. 1.3.2-4 Influence of a 1-m sea level rise resulting from global warming. The map shows the areas of Tokyo that will be below sea level if there is a 1-m sea level rise [1.3.2-6, 1.3.2-7].

1.3.2.2 CO_2 Emission Unit [1.3.2-8, 1.3.2-9]

The need for a reduction in CO_2 emissions to prevent global warming was discussed in the previous section. In Japan, the production of electricity by electric utilities produces 30% of the total emissions. Other industrial processes produce 40% of the total emissions. The global warming effect by electric power generation is now discussed on the basis of CO_2 emission units, which relate to CO_2 emissions that accompany the construction and operation and fuel consumption of power plant, leakage of methane in exploration, etc.

CO_2 emission unit = CO_2 emission during the life of the plant (construction + operation + fuel + methane leaks) / output to the power grid during the life of the plant

Figure 1.3.2-5 compares CO_2 emission units for fossil fuel power, fission power, etc. and fusion power generation estimated by the Central Research Institute of Electric Power Industry [1.3.2-8] and by Tokimatsu [1.3.2-9]. Coal, oil, and LNG (Liquefied Natural Gas) fired power generation expels significantly more CO_2 than the other power plants. Sequestration of CO_2 in LNG and coal fired power generation allows reduction of

CO_2 emissions to only a third. Thus, fossil fuel power generating plants are regarded as the main sources of CO_2 even after the sequestration. Since a rapid increase in energy consumption in developing countries will likely double the present world energy consumption in the 21st century, it is desirable to give higher priority on the development and construction of power generation systems with small CO_2 emission units, such as those driven by water power, fission power, and fusion power.

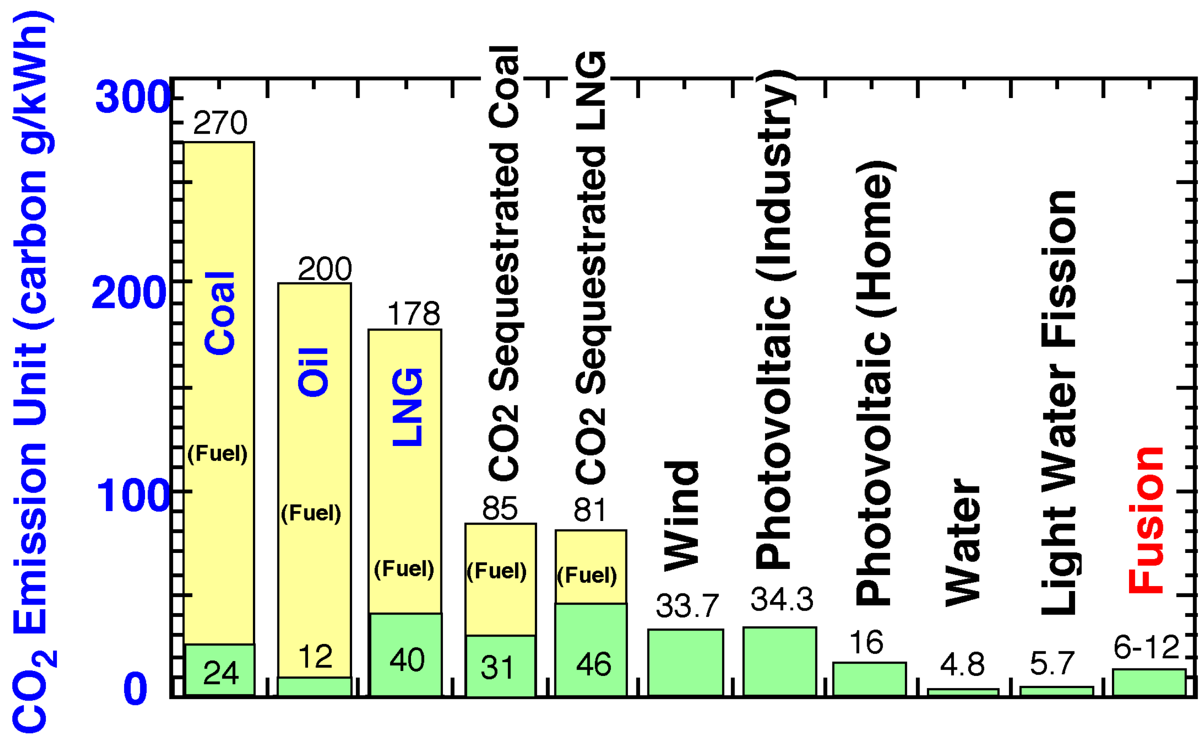

Fig. 1.3.2-5 CO_2 emission units for water power, fossil fuel power, photovoltaic power, wind power, nuclear fission, and nuclear fusion power generation. Values for nuclear fusion assessed by Tokimatsu et al [1.3.2-9] and for others by Uchiyama [1.3.2-8].

1.3.2.3 Suppression of Global Warming with CO_2 Sequestration in Fossil fuel Plants

The reduction of CO_2 emissions from fossil fuel power plants is an urgent issue to resolve the global warming problem. The CO_2 sequestration methods for fossil power plants are the PSA method (using an adsorbent made of zeolite etc.), an amine method (using a mono-ethanol amine solvent), etc.

Sequestration of CO_2 has been studied in detail by the Central Research Institute of Electric Power Industry. As indicated in Fig. 1.3.2-6, desulfurization and denitration processes are required in a coal-fired power plant, resulting in an estimated increased power generation cost of 10.7 yen/kWh because of the investment in the process facilities. In some case, sequestration of CO_2 lowers the thermal efficiency to 25% and raises the power generation cost to over 20 yen/kWh, as shown in Fig. 1.3.2-6. For coal-fired power plants, the pure oxygen PSA method is the most efficient. Yet, the power generation cost is estimated to increase to 17.5 yen/kWh, even when the liquefied CO_2 is simply deposited into the deep sea (below 3,000 m). Here, in the pure oxygen method, coal is burned in CO_2 and oxygen separated from the air, so the only gas produced is CO_2. For LNG-fired power plants, the estimated power generation cost including desulfurization and denitration is 9.7 yen/kWh, but the cost for CO_2 sequestration and deposition of liquefied CO_2 into the deep sea increases the total cost to 14.6 yen/kWh [1.3.2-8]. The cost assessment in the US in 1999 also points out that CO_2 sequestration leads to a COE (cost of electricity) increase of 1 ¢/kWh for LNG and 2 ¢/kWh for coal fired power generation [1.3.2-11].

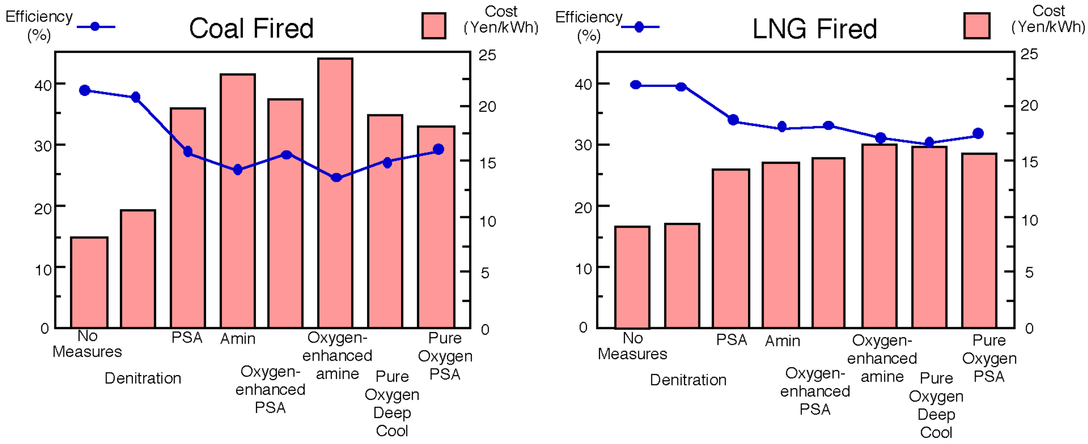

Fig. 1.3.2-6 Comparison of estimated power generation costs and efficiencies of 1 GW coal-fired and LNG-fired power plants for cases of no countermeasures to the environment, and denitration and various CO_2 sequestration methods for environmental protection (based on 1991 prices) [1.3.2-10].

In Europe and the US, CO_2 disposal into natural gas wells is possible. However, Japan does not have any appropriate sites for such disposal. For this reason, ocean retention (the specific gravity of CO_2 is greater than seawater at a depth of 3,000 m) may be the most promising method for CO_2 disposal. There are, however, concerns about the influence of the retained CO_2 on the ocean and the behavior of the CO_2 in the ocean over a long time. The cost estimates from the Central Research Institute of Electric Power Industry do not consider the secondary costs for countermeasures to minimize the influence on the oceans.

To summarize the problems on fossil fuel power generation with CO_2 sequestration,

- A large increase in COE (cost of electricity),
- A large drop in power generating efficiency,
- Re-emission of CO_2 accompanied by the deep-sea deposit and retention,
- Despite CO_2 sequestration, the CO_2 emission unit is larger than the other power sources, and
- Energy payback ratio is significantly lowered (see section 1.3.5.1).

Reference [1.3.2-10] also provides estimates for the total cost to introduce CO_2 sequestration systems to all coal and LNG fired power plants in Japan. Such an effort would require 3.7 trillion yen for the capital investment and 1.6 trillion yen for the facility operation and the resulting power loss. Since the present income in the power generation business is 6 trillion yen, it is dubious that CO_2 sequestration could be implemented without affecting the generating cost. Although shifting from coal fired to LNG fired power generation can curb CO_2 emissions, LNG will not meet energy demands in the long term because of finite reserves.

1.3.2.4 Suppression of Global Warming Using Nuclear Power Generation

Emissions of CO_2 accompanied by the mass consumption of fossil fuels have begun to affect the terrestrial environment. Therefore, replacing fossil fuels with environmentally friendly energy sources will be a prime issue in the 21st century. Nuclear power generation by light water reactors and fast breeder reactors has small CO_2 emission unit and LWRs are becoming more economical through reductions in construction costs in recent years. For example, the ABWRs recently introduced at Kashiwazaki Kariwa Units 6 & 7 of the Tokyo Electric Power Cooperation realized an average construction cost of 270 k yen/kW.

On the other hand, no nuclear power plants have recently been constructed in a suburban location in the light of public acceptance. To this end, nuclear power stations for the Kanto district are built in remote sites 200-300 km distant from Tokyo. It should be noted that the transmission cost of electric power is not negligible. The transmission cost from Aomori to Tokyo, for example, is comparable with the construction cost of a power plant.

1.3.2.5 Suppression of Global Warming Using Renewable Energy

Renewable energy (photovoltaic power generation, wind power generation, biomass, etc.) has received attention because of low CO_2 emissions. For photovoltaic (solar cell) and wind power generation, the output depends on weather and wind speed, respectively, and it is difficult to supply the required power for the required times.

Photovoltaic power generation is able to contribute to reduce peak power loads up to 10GW, and that peak load reduction is as valuable as 30 yen/kWh. However, the introduction of more than 10 GW in output does not contribute to peak reduction. In addition, the weather dependence of photovoltaic power generation (in combination with the daily and annual cycles) can reduce the annual capacity factor to as low as 12%. Thus, a 10 GW system could have an output only corresponding to the power supplied by a 1.5 GW nuclear power station with an availability of 80%.

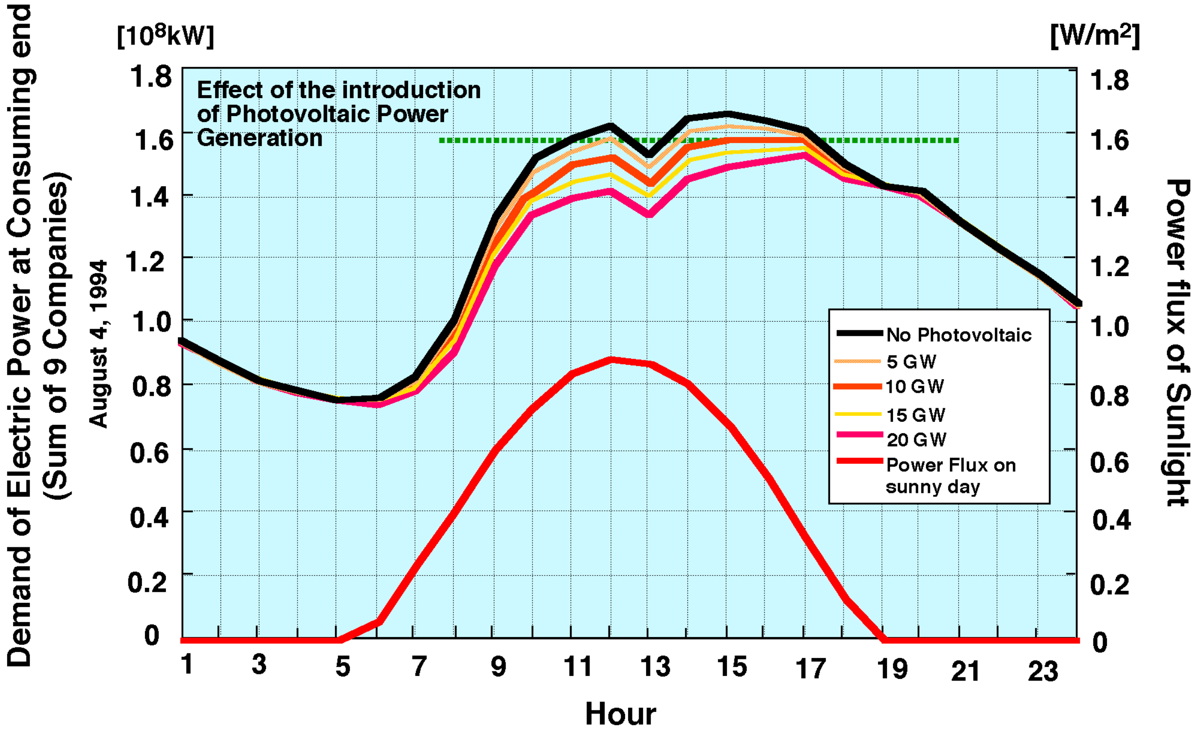

Fig. 1.3.2-7 Solar energy density of sunlight and the reduction of peak electric power demand by the introduction of 5-20 GW photovoltaic power generation system [1.3.2-12]

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1.3.3 Safety viewed from Radiological Toxic Hazard Potential

1.3.3.1 Radiological Toxic Hazard Potential

The radiological toxic hazard potential or biological hazard potential (BHP) is a quantity used to evaluate the influence caused by the exposure of a radioactive nuclide that enters the human body. This is defined as the value of the amount of radioactive nuclides (Ci) remaining in the reactor divided by the maximum permissible concentration (MPC: Bq/m^3-air) of the radioactive nuclide in the air. This value is the volume of the air necessary to dilute the radioactive nuclides remaining in the reactor to the MPC [1.3.3-1]. This can be said to be the quantity taken into consideration having an influence on the human body rather than just showing the activity of the radioactive nuclides (Bq). In terms of present radiation protection, derived air concentration (DAC) should be used instead of MPC, but the fundamental way of thinking is the same.

Tritium is a radioactive material from a fusion reactor. Its influence on the human body should taken into consideration. Tritium is a nuclide that can be handled with comparative ease. It emits beta rays with energies of 18.6 keV maximum and 5.7 keV average, and these rays can be shielded by one sheet of paper. Therefore, there is little danger from external exposure. If the tritium internally enters the human body, it does not remain in specific internal organs selectively. It is discharged from the body due to the metabolism at a rate with a 10-day half life if it is in the form of water and at a rate with a 40-day half life if it becomes an organic substance.

On the other hand, some radioactive materials produced in the light-water reactor are strontium-90, cesium-137, iodine-131, and so on. Iodine-131 is of most concern since it has the most influence on a human body. This nuclide is accumulated by thyroid gland and remains in the body for a long time. Therefore, it has a larger influence on the human body for other nuclides having the same radioactivity.

Table 1.3.3-1 shows a comparison of the radiological toxic hazard potential (=Radioactivity / Concentration limit in the air) considering the difference in such a biological influence as a concentration limit in the air.

Table 1.3.3-1 Comparison of the radiological toxic hazard potential between tritium and iodine-131 which are the typical volatile radioactive nuclides contained in a fusion reactor and a light-water reactor, respectively [1.3.3-2]

	Tritium in a fusion reactor generating 1 GW of electricity (T: 4.5 kg, in the form of HTO)	Iodine-131 in a light-water reactor plant generating 3 GW of thermal output
Radioactivity (Bq)	1.7×10^{18}	5.4×10^{18}
Concentration limit in the air (Bq/m^3)	5×10^{3}	10
Radiological toxic hazard potential (m^3)	3.5×10^{14}	5.4×10^{17}
Relative ratio	1	1500

The radioactivity of the tritium in a fusion reactor power plant generating 1GW of electricity is almost equal to that of the iodine-131 in a light-water reactor. However, the radiological toxic hazard potential of tritium is 1000 times less than that of the iodine-131 in the light-water reactor plant, because the concentration limit of tritium in the air is 1000 times higher than that of iodine-131. Namely, it can be said that the latent influence potential of the radioactive inventory to the human body in a fusion reactor is less than 1/1000 of that of a light-water reactor.

Figure 1.3.3-1 shows the low radiological toxic hazard potential and the low CO_2 emission unit of a fusion power reactor (described in Section 1.3.2 CO_2 Emissions and Sustainability of Atmosphere) when compared with those of a fission power reactor and a coal-fired power plant.

Global warming due to large production of CO_2 and latent risks radiation exposure are two major issues that must be minimized or eliminated the energy systems of the 21st century. This figure shows that a fusion reactor is capable of reducing both risks significantly and simultaneously.

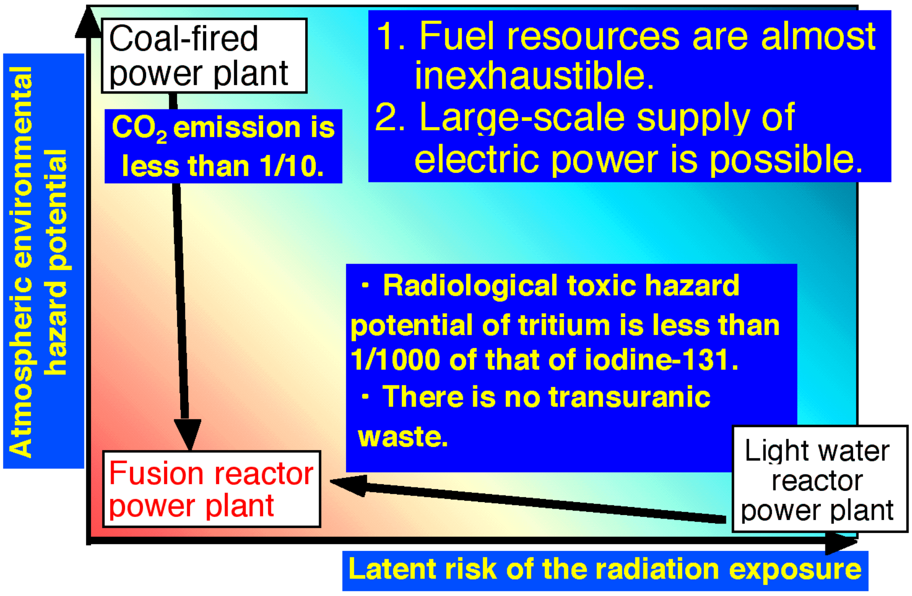

Fig. 1.3.3-1 Significance of fusion development from the viewpoint of the latent risk of radiation exposure and the atmosphere environmental hazard potential. This figure uses the radiological toxic hazard potential as the latent risk of radiation exposure and the CO_2 emission unit as the atmosphere environmental hazard potential [1.3.3-3].

1.3.3.2 Comparison of the Radiological Toxic Hazard Potential between Light-Water Reactor Plant, Fusion Reactor Plant, and Coal-Fired Power Plant [1.3.3-4]

In the previous section, the radiological toxic hazard potential of the tritium in the fusion reactor was compared with that of the iodine-131 (^{131}I), which is typical volatile radioactive nuclide in a fission reactor. In this section, the radiological toxic hazard potential by inhalation intake (taken into the body through the air by breathing) and ingestion intake (taken in the body through food and/or water) is compared for all the radioactive material contained in the system.

The SSTR designed in the Japan Atomic Energy Research Institute (JAERI) is selected as a typical fusion reactor, and a pressurized water reactor power plant generating 1 GW of electricity is selected as a typical light-water reactor. The coal ashes formed, that is the fly ash mixed with a solid, are evaluated in a coal-fired power plant having an output of 1 GW of electricity. Each radiological toxic hazard potential was evaluated for the radioactivity generated by operating for 30 years.

(1) Fusion reactor

The fusion demo reactor SSTR (1.08 GWe) is used as a reference. This fusion reactor uses the low-activation steel (F82H) as a fusion core structural material. The ANISN code [1.3.3-5] was used to calculate the types and amounts of radioactive nuclides in the fusion reactor based on the reactor materials and the configuration. The radiological toxic hazard potential was evaluated based on those quantities. As a precondition of the evaluation, the blanket is replaced every two years, other structures are in service for 30 years, and the assumed plant capacity factor is 100%. Tritium is almost completely burned by the fuel cycle, but the inventory of 4.5kg is considered in this evaluation.

(2) Light water reactor

A pressurized water reactor (PWR) plant generating 1 GW of electricity was selected. Only nuclear fuel was taken into consideration in this evaluation; the structural material was ignored. The nuclear fuel was assumed to be uranium dioxide (UO_2) having an initial enrichment of 4.1% used to a burn-up of 33,000 MWd/t

(MW day/ton, i.e., burn-up rate per one ton of uranium). There is a report that 22~30 tons of spent fuel are produced by a 1 GW power plant operated for one year. Here, the amount of radioactivity is assumed to be from 25 tons of spent fuel. The nuclide and radioactivity quantity due to nuclear burn-up, the attenuation throughout storage after operation, and the radiological toxic hazard potential were evaluated by using the ORIGEN2 code [1.3.3-6] based on the above data.

(3) Coal-fired power plant (one million kW electricity)

According to the literature [1.3.3-7], the amount of coal burned by operating a 1-GW coal-fired power plant for 30 years is 90 million tons. The radioactive nuclides and amounts included in the coal are summarized in Table 1.3.3-2. The nuclide amounts were obtained from logarithmic means of concentrations of the radioactive nuclides in the coal mined at each place in the world as shown in the literature. Assuming the coal-fired power plant operates for 30 years, the radioactivity and the radiological toxic hazard potential are evaluated and summarized in Table 1.3.3-2. The combustion of the coal creates 21 million tons of coal ash so the concentration of each radioactive nuclide is increased 4-5 times by the combustion. The time change of the radiological toxic hazard potential of coal ash discharged from the coal-fired power plant was obtained by using the ORIGEN2 code based on the quantities of these radioactive nuclides.

Table 1.3.3-2 The radioactive nuclide, half life, average concentration, radioactivity, concentration limit in air by the literature [1.3.3-9], and their radiological toxic hazard potential, which are included in the coal burned by operating a 1-GW coal-fired power plant for 30 years. The concentration limit was evaluated considering the inhalation intake and ingestion intake for the public.

Radioactive nuclide	Half life	Average concentration (Bq/kg)	Radioactivity (TBq)	Concentration limit in air (Bq/m^3)	Radiological toxic hazard potential (m^3)
^{40}K	1.28 billion years	127	11.4	30	3.8x10^{11}
^{238}U	4.468 billion years	27.8	2.50	4x10^{-3}	6.25x10^{14}
^{232}Th	14.0 billion years	18.4	1.66	3x10^{-4}	5.53x10^{15}
^{226}Ra	1,600 years	11.9	1.07	0.05	2.14x10^{13}
^{228}Ra	5.77 years	21.3	1.92	0.1	1.92x10^{13}
^{210}Po	138.38 days	23.1	2.07	0.05	4.14x10^{13}
^{210}Pb	22.26 years	16.7	1.5	0.03	5.0x10^{13}
Total					6.3x10^{15}

The radiological toxic hazard potentials concerning the inhalation and the ingestion intake are given in Fig. 1.3.3-2. The SSTR fusion reactor has smaller radiological toxic hazard potentials for inhalation intake than the PWR light water reactor. The difference is 100 times less just after cessation of operation, and 1,000 times less after 1 year. After that, the difference rapidly increases and reaches a difference of about 1,000,000 times less after 100 years. For the fusion reactor, the toxic hazard potential remains constant after 100 years. For the light water reactor, it gradually decreases for 200,000 years; following that, its change is extremely small. The value of the fusion reactor is 100 times more than that of the coal-fired power plant just after cessation of operation. About 20 years later, the value of the fusion reactor is equal to that of the coal-fired power plant because the value of the coal ash does not change much while that of the fusion reactor rapidly decreases. With the passing of more time, the value of the fusion reactor decreases further.

Considering the radiological toxic hazard potential of ingestion, that of the fusion reactor is about 10 times less than that of the light water reactor until 1 year after cessation of operation. At 100 years, the value of the fusion reactor becomes 100,000 times less than that of the light water reactor, since the former rapidly attenuates. In the light water reactor, the value decreases for 200,000 years; its change becomes extremely small after that. This is the same tendency as with inhalation. The radiological toxic hazard potential of the fusion reactor is over 1000 times larger than that of the coal-fired power plant from cessation of operation to 1 year, but rapidly attenuates afterwards. As is the trend for inhalation, the value of the fusion reactor becomes equal to that of coal ash in the coal-fired power plant at 20 years, and becomes smaller after that.

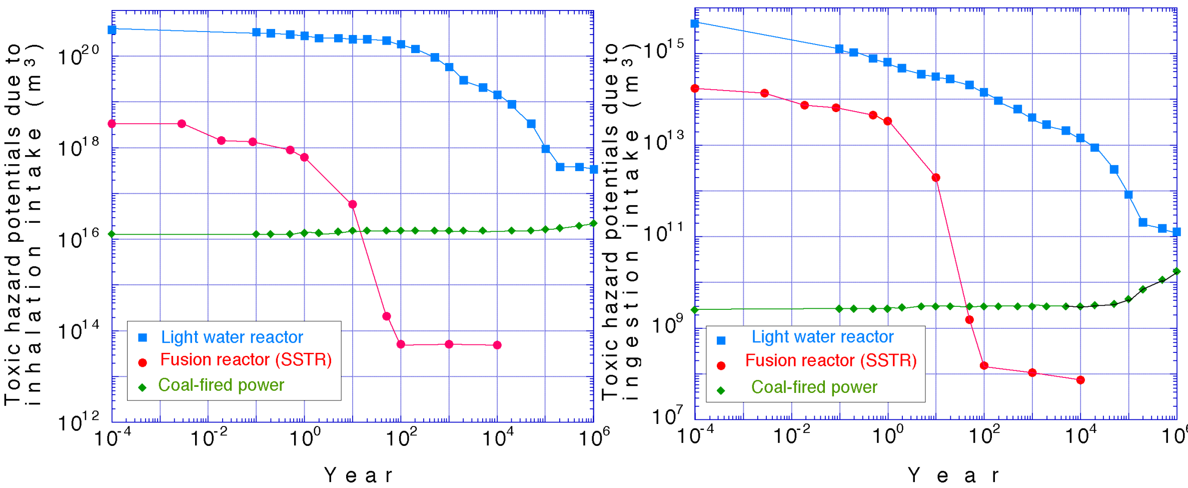

Fig. 1.3.3-2 Comparison of the radiological toxic hazard potentials due to inhalation intake and ingestion intake from a fusion reactor, a light water reactor, and a coal-fired power plant (Standards are U.S.A. 10 CFR Pt.20 App. B Table 2 (Effluent Concentration)), which are almost the same as those in reference [1.3.3-9].

Table 1.3.3-3 summarizes the nuclides having large radiological toxic hazard potentials 30 days after the shutdown of the SSTR fusion reactor. The six most hazardous nuclides are iron-55 (^{55}Fe), iron-59 (^{59}Fe), cobalt-60 (^{60}Co), tantalum-182 (^{182}Ta), manganese-54 (^{54}Mn), and tungsten-185 (^{185}W). These nuclides arise from the impurities and alloying elements in the low-activation ferritic steel. These nuclides occupy the largest fraction beginning 1 week after shutdown, but the radioactivity of these nuclides attenuates and beryllium-10 (^{10}Be) eventually becomes the dominant nuclide (after 100 years). The possibility of dispersing ^{55}Fe, ^{59}Fe, ^{60}Co, ^{182}Ta, ^{54}Mn, and ^{185}W into the atmosphere by accidents is drastically lower than that of tritium, since these are all metallic elements and are contained in metallic alloys. The radioactive cooling time should be about 100 years, after which waste disposal, such as blanket removal, and the decommissioning of reactor in the power plant becomes possible.

Table 1.3.3-3 Radiological toxic hazard potential of main radioactive nuclides 30 days after shutdown of the SSTR fusion reactor

Nuclide	Half life	Radioactive nuclide(Bq)	Radiological toxic hazard potential for inhalation intake (m^3)	Radiological toxic hazard potential for ingestion intake (m^3)
^{55}Fe	2.685 years	1.28×10^{19}	3.33×10^{17}	2.22×10^{13}
^{54}Mn	312.2 days	3.73×10^{18}	3.99×10^{17}	2.79×10^{13}
^{185}W	75.1 days	1.04×10^{18}	1.74×10^{16}	3.50×10^{12}
^{182}Ta	115 days	6.95×10^{17}	4.58×10^{17}	7.62×10^{12}
^{59}Fe	44.56 days	6.69×10^{17}	3.35×10^{16}	2.01×10^{12}
^{60}Co	5.27 years	6.51×10^{16}	1.14×10^{17}	8.52×10^{11}

The environment safety assessment project of the fusion reactor, SEAFP, published a report in Europe in 1995 [1.3.3-10]. In this report, the latent dosages of radioactive materials accumulated during the plant operation through inhalation intake and ingestion intake into the human body are defined as the radio-toxicity indexes through the inhalation intake and the ingestion intake, respectively. Figure 1.3.3-3 shows the comparison of the radio-toxicity indexes by the inhalation intake and the ingestion intake from two kinds of fusion reactors having different structural materials, two kinds of fast neutron reactors, a light-water reactor, and a coal-fired power plant. In this case, spent nuclear fuels were used as the radioactive materials in the fission reactor, and uranium and thorium in the solid cinders were used as radioactive materials in the coal-fired plant. The reuse of the radioactive waste was not considered. These results reinforce the conclusions reached earlier in this section for the radiological toxic hazard potentials.

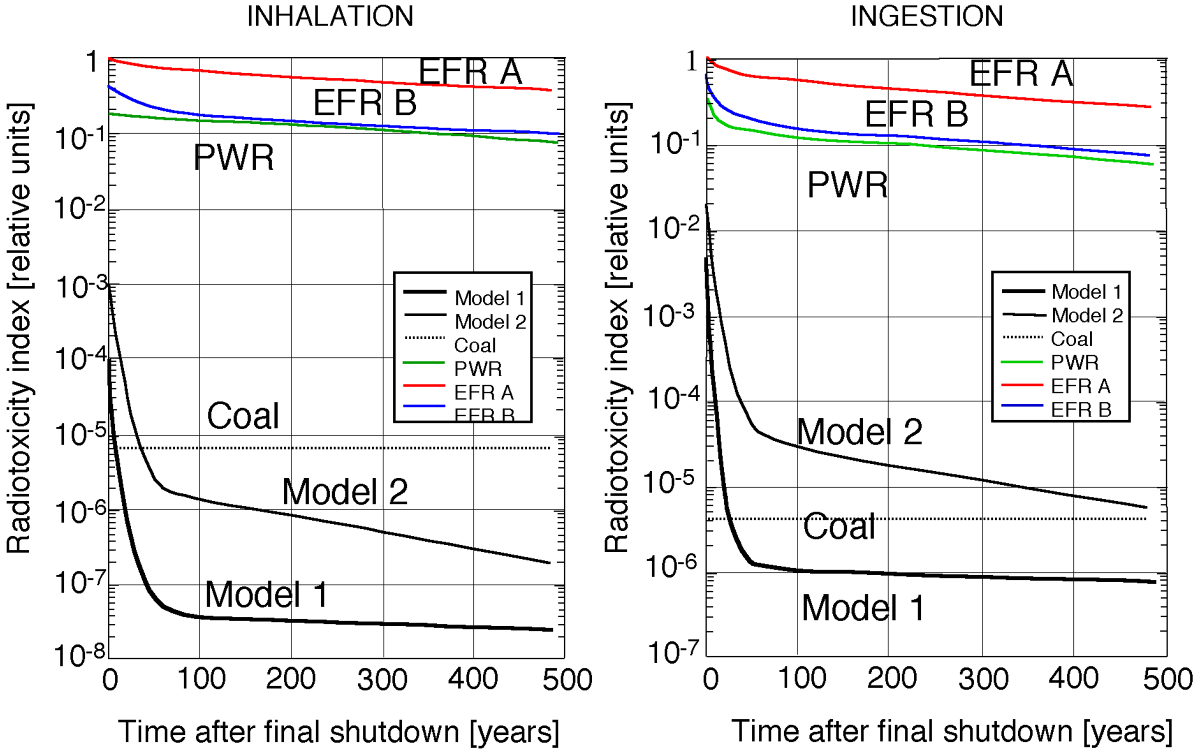

Fig. 1.3.3-3 Radiotoxicity index by inhalation intake (left) and ingestion intake (right) in fast reactors (EFR A, EFR B), a light-water reactor, a coal-fired plant, and a fusion reactor using vanadium alloy (Model 1) and a fusion reactor using low-activation steel (Model 2) [1.3.4-10].

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1.3.4 Radioactive Waste and Environmental Adaptability

As described in the previous section, a fusion reactor has the characteristic that high-level waste, such as occurs in a fission reactor, is not produced. This is because radioactive waste does not result from fusion reactions. Only materials irradiated by the neutrons resulting from the reaction may be considered radioactive waste in fusion reactor. The radiological toxic hazard potential is much lower than for the fission reactor, and it can be minimized by careful material selection. A fusion reactor has the excellent advantage that its radiological toxic hazard potential is less than that of the radioactive materials in the combustion ashes of a coal-fired plant, i.e., thorium-232 (^{232}Th). However, the amount of radioactive waste from the fusion reactor is significant. In this section, the type and quantity of radioactive wastes generated by a fusion power reactor will be described. Also, the exposure doses from radioactive wastes of a fusion reactor, a fission reactor, and the combustion ash of a coal-fired plant will be compared.

1.3.4.1 Classification of the Radioactive Waste

The wastes generated from a nuclear power plant are classified as “high-level radioactive waste,” which is generated primarily from the operation of a reprocessing facility, “low-level radioactive waste,” and “waste that does not need to be treated as a radioactive material.” The Nuclear Safety Commission defined the radioactive concentration limit below which shallow land burial of the radioactive waste is possible. This is defined in a government regulation. The radioactive wastes below and above this concentration limit are called “low-level waste” and “high βγ waste,” respectively. Radioactive waste with an extremely low concentration is called “ultra-low-level waste,” and its criterion is not decided yet. Other wastes including transuranic (TRU) waste, uranium waste, RI and laboratory waste are classified as low-level radioactive waste.

Table 1.3.4-1 Nuclear waste classifications and status of government examination and regulation [1.3.4-1]

Waste level			Atomic Energy Commission: Disposal plan	Atomic Energy Commission: Execution system and responsibility assignment	Nuclear Safety Commission: Fundamental thinking for Safety regulation.	Nuclear Safety Commission: Actual standard such as radioactive concentration limit etc.	Related laws and regulations
High-level radioactive waste			Completed in June 1998.		Under investigation since June 1998.	T.B.D.	T.B.D.
Low-level radioactive waste	Generated from nuclear facility	High βγ waste (Radioactive level is high. In-vessel structure, etc.)	Completed in June 1998.	Completed in October 1985.	T.B.D.		T.B.D.
		Low-level waste (Radioactive waste below concentration limit. This can be buried in a concrete pit.)	Completed in August 1984.		Completed in October 1985.	Completed from Feb. 1987 – Jan. 1993.	Completed from March 1987 to Sept. 1994.
		Ultra low level waste (Radioactive concentration is very low. This can be laid in a trench.)				(metals with very low radioactive concentration are left for future examination)	
	Transuranic waste, Uranium waste		Under investigation		T.B.D.		T.B.D.
	RI, laboratory waste		Completed in June 1998.		Being examined since June 1998.	T.B.D.	T.B.D.
Waste does not need to be dealt with as a radioactive material.	Waste under the clearance level.		Completed in August 1984.		Under investigation since May 1997.		T.B.D.
	No possibility at all of being contaminated by radioactive material.				Completed in June 1992.		

As shown in Table 1.3.4-1, a determination on the disposal of radioactive waste of a fission reactor is progressing through examination by the Atomic Energy Commission and Nuclear Safety Commission. Related laws and regulations are being advanced, driven in part by the decommissioning of a commercial plant, the Tokai nuclear power plant. Shallow land burial in a concrete pit 5 m underground would be possible for the low-level waste under the government regulation. On the other hand, the high βγ waste, waste that is above the government regulation limit, should be buried in an underground cavity, 50 – 100 m below the surface, in a designated disposal facility, where safety assurance is thought to be possible by proper safety management for a few hundred years. Furthermore, disposal of high-level radioactive waste in a stable rock formation several hundred meters (500 - 1,000 m) underground is being considered. An interim report has been summarized in the atomic energy subcommittee of the Advisory Committee for Energy [1.3.4-3].

(1) Radioactive concentration upper limit of representative nuclides generated in fission and fusion reactors for shallow land burial

The radioactive concentration upper limit was evaluated so that annual individual exposure dosage from one radiation source after the management period (300 years) is 0.01 mSv/year for the exposure route in the IAEA-TECDOC-401. Here 0.01 mSv/year is 1/10 of 0.1 mSv/year corresponding to the regulation exemption of ICRP (fatality probability below 10^{-6} can be negligible as a personal risk) [1.3.4-4]. The Nuclear Safety Commission has evaluated this shallow land burial radioactive concentration upper limit on representative nuclides from a fission reactor such as carbon-14 (^{14}C), calcium-41 (^{41}Ca), cobalt-60 (^{60}Co), nickel-63 (^{63}Ni), strontium-90 (^{90}Sr), cesium-137 (^{137}Cs), and alpha (α)nuclides. T. Tabara, et al. [1.3.4-5] has evaluated these values for representative nuclides from a fusion reactor, such as tritium (T=^{3}H), beryllium-10 (^{10}Be), carbon-14 (^{14}C), aluminum-26 (^{26}Al), chlorine-36 (^{36}Cl), cobalt-60 (^{60}Co), nickel-59 (^{59}Ni), nickel-63 (^{63}Ni), niobium-91 (^{91}Nb), niobium-94 (^{94}Nb), molybdenum-93 (^{93}Mo), rhenium-186 (^{186}Re), iridium-192 (^{192}Ir), and platinum-193 (^{193}Pt).

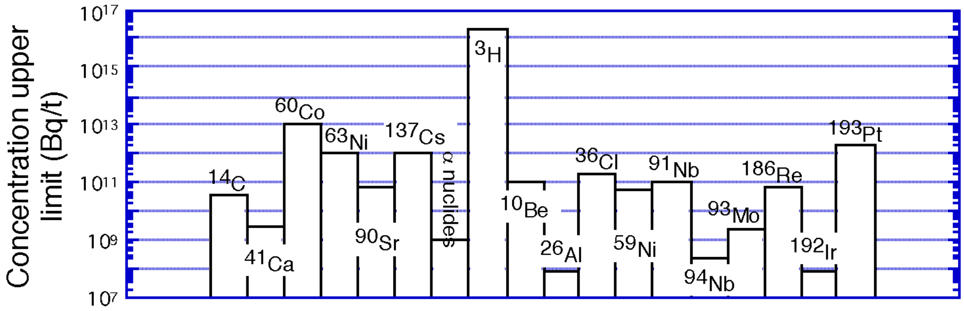

Fig.1.3.4-1 The concentration upper limits, listed by nuclide, in which the shallow land burial for these representative nuclides of fusion reactor and fission reactor becomes possible. ^{14}C, ^{41}Ca, ^{60}Co, ^{63}Ni, ^{90}Sr, ^{137}Cs,α nuclides use the government ordinance values, others use the values evaluated by T. Tabara, etc.

(2) Radioactive waste arising during disassembly of a commercial nuclear power plant

On the basis of the above government ordinance on the concentration upper limit, the classification is determined as high βγ waste surpassing concentration upper limit, low-level waste under the concentration upper limit, ultra-low-level waste, and waste under the clearance level.

The gas cooled reactor plant produces largest amount of high βγ waste and low-level waste under the concentration upper limit. The amount of high βγ waste is considerably smaller in the light water reactor plants.

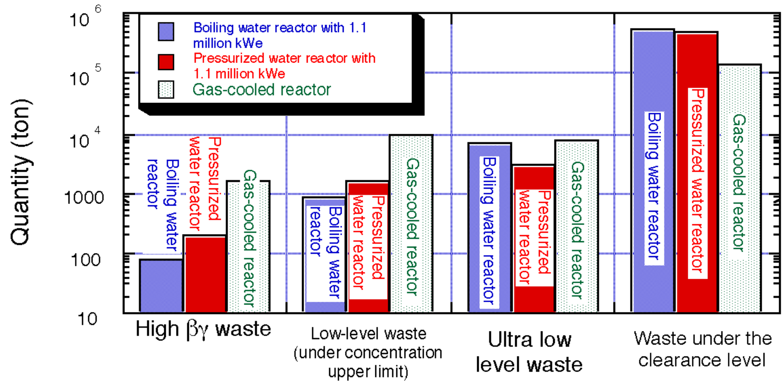

Fig. 1.3.4-2 Disassembly waste amounts from the commercial fission nuclear power plants (boiling water reactor, pressurized water reactor, and gas-cooled reactor). Numerical values are those after system decontamination (DF=30) and post-disassembly decontamination (DF=100). Secondary wastes are excepted. The values under the clearance level are estimated based on the numerical values of the clearance level in the IAEA-TECDOC-855 [1.3.4-6]. Here DF (Decontamination Factor) is ratio of concentration before and after the surface decontamination.

(3) Comparison of waste amounts between BWR and PWR light water reactors, a fusion reactor (SSTR), and a coal-fired plant

Figure 1.3.4-3 shows a comparison of the waste volume (m^3) between a BWR, a PWR, and a fusion reactor (SSTR). The most important characteristic of the fusion reactor is that no high-level radioactive waste is produced. However, the amount of low-level radioactive waste from the fusion reactor is a few times more than that from fission reactors. Low-activation ferritic steel is used as a probable material in SSTR for this evaluation. This fusion reactor produces a comparatively large amount of high βγ waste because of the multistage nuclear transformation reaction of tungsten (2 wt%), which is used as an alternative alloying element to molybdenum in low activation ferritic steel.

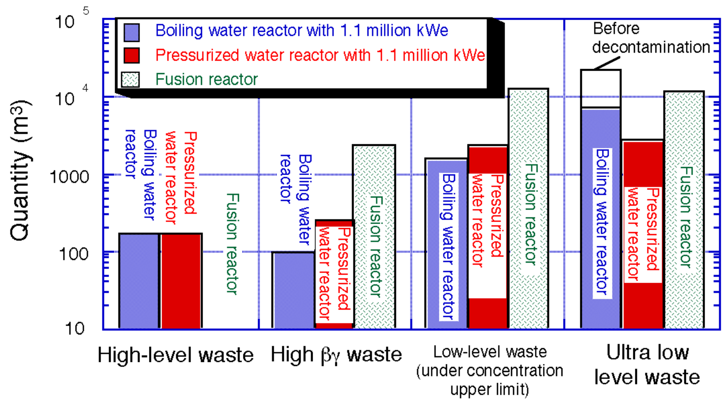

Fig. 1.3.4-3 Quantity of radioactive waste after the decontamination of a BWR, a PWR, and a fusion reactor. The amount of high-level radioactive waste is from a 1 GW nuclear reactor after 30-years of operation. The amount of ultra-low-level waste from the boiling water reactor is the value before post-disassembly decontamination. The numerical value for the fusion reactor (SSTR) is only from the reactor and not from building components and peripheral equipment such as the biological shield concrete. Evaluation of the ultra-low-level waste from fusion reactor used a radioactivity level of 74 Bq/g, which is the lowest value of concentration defined as a radioisotope in the Radiation Hazard Prevention Act and it includes waste under the clearance level. [1.3.4-1], [1.3.4-7]

The combustion ash of a coal-fired power plant is outside the objects managed as radioactive materials at present. There is some indication that the radiation exposure should not be neglected due to the concentration of the natural radioactive materials by the combustion. As shown in Table 1.3.3-2, the average concentration of thorium-232 (^{232}Th) in coal is 18.4 Bq/kg. Combustion increases the radioactive concentration in coal ash to about 80 Bq/kg. The 1988 report of the United Nations scientific committee described 70 Bq/kg as the average concentration of thorium-232 (^{232}Th) in airborne fly ash. This value exceeds lowest value 20 Bq/kg (Table I-6) of the clearance level evaluated in IAEA-TECDOC-855.

Especially troubling is the fact that 280 M tons of coal ash produced every year is used for manufacturing cement and concrete, road stabilization material, road filler, asphalt admixture, and fertilizer. As described in a report of the United Nations scientific committee, there is some indication from the viewpoint of radiation that attention should be directed to the use of coal ashes for building materials. The report stated that the annual equivalent effective dose from a concrete building and a wooden house was estimated to increase by 70 μSv/year and 30 μSv/year, respectively, by using the fly ash in the concrete for the house construction. The group equivalent effective dose by the use of 14 million tons of coal ash for building materials has been estimated as 50,000 man Sv.

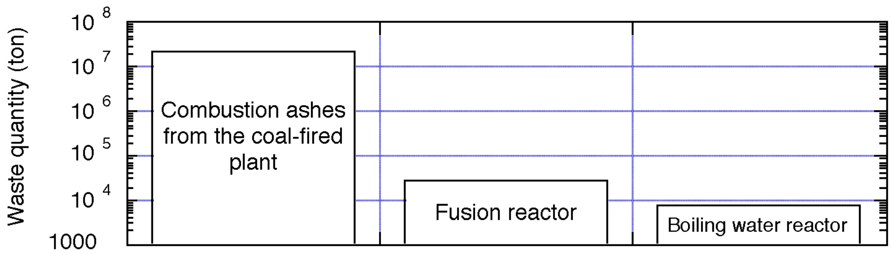

Fig.1.3.4-4 Comparison of the amount of material from a coal-fired plant (combustion ash), fusion reactor waste, and fission reactor waste. The weight of coal ash is extraordinarily larger than that of the waste from a fusion reactor or a light water reactor.

(4) Cost comparison of radioactive waste disposal for a fission reactor and a fusion reactor

The disposal cost of the radioactive waste (high-level radioactive waste is excluded) associated with the decommissioning of nuclear reactors has been evaluated as 17.8 billion-yen for a BWR and 19.2 billion-yen for a PWR (both being 1.1 GW reactors) [1.3.4-1]. In a research report of the atomic energy society, the waste disposal cost of a fusion reactor is compared with that of a fission reactor [1.3.4-7]. The disposal cost of the radioactive waste arising from the decommissioning of a 1 GW fission nuclear reactor is estimated to be 15.54 billion-yen. When the disposal cost includes high-level waste, the cost is estimated to be above 100 billion-yen. However, since there is no high-level radioactive waste, the total waste disposal cost for a fusion reactor is estimated to be about 36 billion-yen.

	High-level waste	High βγ waste	Low-level waste	Total
Fission reactor (1 GW of electricity)	90 B yen / 180 m^3	3.84 B yen / 1,600 m^3	11.7 B yen / 9,750 m^3	105.54 B yen
Fusion reactor (SSTR, 1.08 GW of electricity)	-	6 B yen / 2,500 m^3	30.12 B yen / 25,100 m^3	36.12 B yen

Table1.3.4-2 Comparison of waste disposal cost between a fission reactor and a fusion reactor [1.3.4-5]. Disposal unit prices are ¥1,200,000/m^3 for low-level waste, ¥2,400,000/m^3 for high βγ waste, and, ¥500 x 10^6/m^3 for high-level radioactive waste [1.3.4-7].

(5) Radioactive waste from ITER

Considering the radioactive waste from ITER as it is presently being designed, the details are dependent on the future design. The amount of waste having significant activation, and thus being classified as controlled waste, has been estimated to be about 39,000 tons from the main body. Low-level radioactive waste dominated by the surface contamination is excluded from consideration in the final disposal by decontamination and incineration. Most of the radioactive wastes included in the 39,000 tons generated during the disassembly will be below the clearance level about 100 years after the disassembly, although about 12,000 tons will remain controlled waste.

1.3.4.2 Long term risks of radioactive waste disposal from a fusion reactor and a light water reactor

Figure 1.3.4-4 (left) shows the evaluation of the individual exposure dosage in the environment by radioactive waste considering the permeability rate of the radioactive nuclides through artificial and natural barriers when the radioactive waste from a fusion reactor using various structural materials is buried in a deeper stratum (more than 50 m underground) [1.3.4-5], [1.3.4-7]. Here, drums are filled with the radioactive waste and then are arranged in an orderly manner in a concrete pit. Voids around the drums are then filled with bentonite, and the pit is covered by a concrete lid. This is similar to shallow-land burial treatment in Rokkasho village. By burial more deeply than 50-m underground, it is regarded that the buried radioactive waste will not be excavated in the future or no other underground utilization is required at designated radioactive waste disposal facilities. Thus, only the underground water path is considered for the radioactive nuclides to reach the biosphere. As a result, after one hundred years, the annual individual dose from one *radiation source in a* fusion reactor is less than 10 µSv, which is 1/10 of 100 µSv corresponding to the fatality probability for 10^{-6}/year being disregarded as a risk to an individual. It is important to establish the technology to decontaminate and recover tritium before burying the structural materials since the exposure risk for the first 100 years occurs from the tritium mainly included in the structural materials.

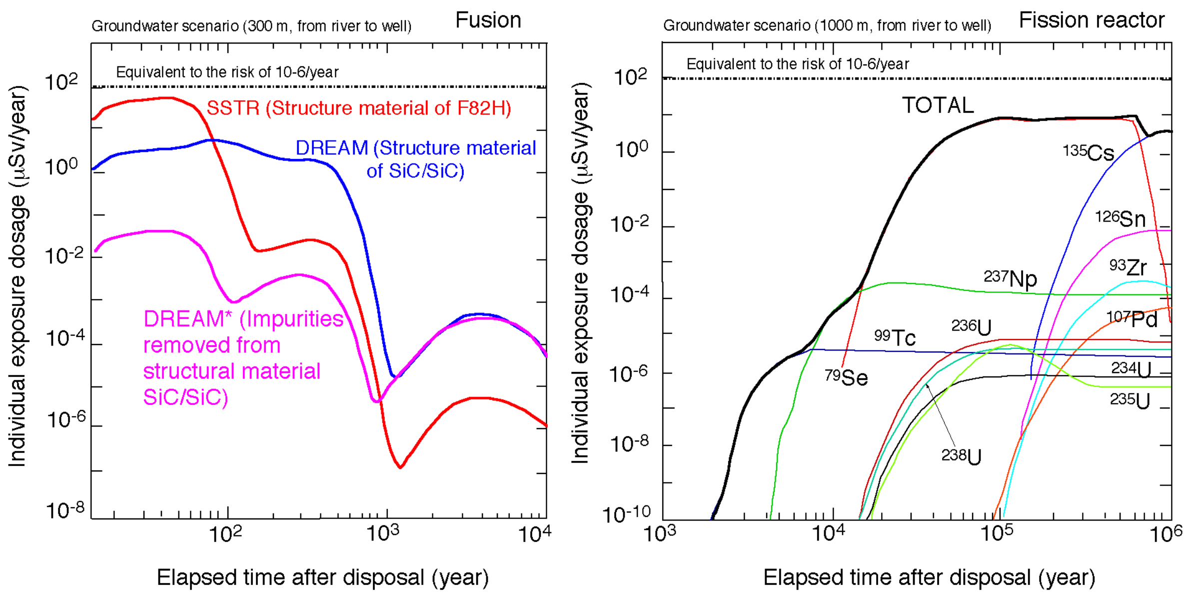

Fig. 1.3.4-5 The left figure shows the individual exposure dosage in the general environment for the disposition of high βγ waste from a fusion reactor in a stratum concrete pit about 50-100 m underground. Disposal volume is 20,000 m^3 that is 15 units in SSTR, 2.2 units in DREAM, and 77 units in DREAM*. The right figure shows the individual exposure dosage in the general environment for the disposition of high-level radioactive waste from a fission reactor deep underground (500-1,000 m). Disposal volume is 2,240 m^3. [1.3.4-5], [1.3.4-7]

Figure 1.3.4-5 (right) shows the environmental effects through the groundwater when high-level radioactive wastes from a fission reactor are disposed in a stable rock mass of 500~1,000 m underground. Here, high-level radioactive wastes are disposed inside the artificial barrier of a waste package (a stainless steel canister containing the solidified glass of the high-level radioactive waste , the outside of the canister is wrapped by oversized packing materials such as about 10~30-cm thick mild steel, titanium, or copper), which is then covered with a buffer clay material, such as bentonite. The individual exposure dosage after several tens of thousands years to several hundreds of thousands years is equivalent to the 10 μSv standard acceptable level, even when it is disposed of in this manner. This comparison shows that the annual individual dose from the radioactive waste disposal facilities for a fusion reactor is less than 10 μSv after hundred years, a value that the high-level radioactive waste of fission reactor only reaches after several tens of thousands to several hundreds of thousands of years. This means that the risk of radioactive waste given to posterity by a fusion reactor is far less than that by a light water reactor.

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1.3.5 Plant Characteristics

In this section, the characteristic of the tokamak power plant is described from viewpoints of basic performances of an ordinary power plant, such as capacity factor, energy payback ratio, recirculation power ratio, output stability, thermal efficiency, power controllability, and so on.

1.3.5.1 Comparison of General Characteristic as Power Sources

(1) Energy Payback Ratio

The energy payback ratio is defined as the ratio of “the total generated energy during the plant lifetime (e.g., 30 years)” to “the total required energy for the plant construction, the operation and maintenance, and the fuel mining, refining, and transportation). As a rule, each type of energy is converted into primary energy since the generated energy is electricity (secondary energy) and the required (consumed) energy is a mixture of primary and secondary energy.

$$R = \eta P/(C1+ \eta C2)$$

Here, R, η, P, C1, and C2 are the energy payback ratio, the conversion coefficient (2250 kcal/kWh), the generated energy, the primary consumed energy, and the secondary consumed energy, respectively.

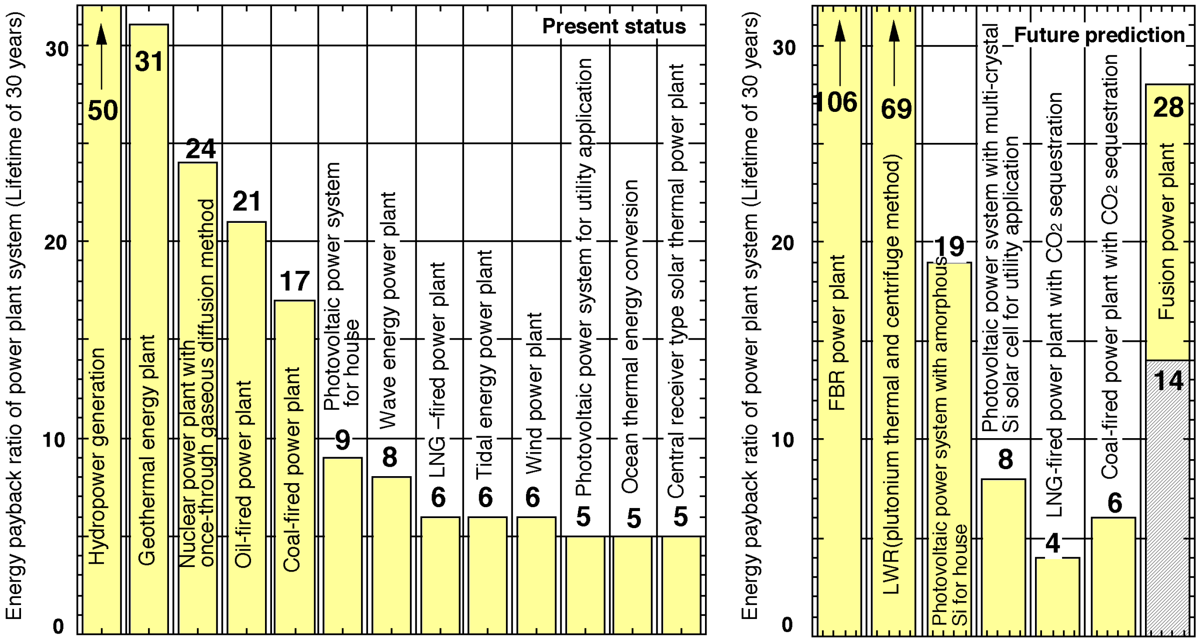

Fig. 1.3.5.1-1 Present status and future prediction of energy payback ratio for various energy systems.

The present and future predicted energy payback ratio for various kinds of power plants is shown in Fig. 1.3.5.1-1. According to the present assessment, the energy payback ratio of hydropower generation is the highest. For a fission nuclear power plant, assuming a fuel burn-up ratio of 30,000 MWD and fuel enrichment using the gaseous diffusion process that consumes much energy, the energy payback ratio is rather high, 24. However, the energy payback ratio is estimated to be 69 if the centrifugal process is applied where the consumed energy is only a few tenth energy of the gaseous diffusion method for enrichment of uranium [1.3.5.1-1]. Even if the laser uranium enrichment method (AVLIS) can be realized, the value will be about the same as the centrifugal method. The household photovoltaic power generation will be improved in the future from 9 to 19. The industrial use of photovoltaic power generation will also be improved from 5 to 8. For a fusion power plant, the energy payback ratio was evaluated as 14~28 [1.3.5.1-2] by the same method as method in [1.3.5.1-1]. The evaluation for fusion was also made in the US. It must be noted that the superiority of a power generation system is not decided by only the energy payback ratio. For example, the energy payback ratio of an LNG plant is low, however it produces very low cost electricity, and has a very low

CO_2 emission rate as described in the Section 1.3.2. The reason the energy payback ratio of the LNG plant is low stems from the fact that a large amount of energy is consumed in liquefying natural gas to make the LNG.

(2) Capacity Factor

The capacity factor is an important element to judge the value of power system. The power should be generated in accordance with the demand situation since there is no efficient energy storage systems. From this requirement, the natural energy systems such as photovoltaic power generation, wind power generation, or wave power generation, are not suitable for a large-scale, main power plant. Figure 1.3.5.1-2 shows the capacity factors of the present power plants. Here, the value for thermal power is the availability factor, not the capacity factor.

The capacity factor of the fusion power plant should be at least same level as the (fission) nuclear power plant, although its prediction is difficult without any commercial fusion plants. The attainability of this requirement is discussed in reference [1.3.5.1-4]. The achieved capacity factor of domestic nuclear plants was 70~80% on average from 1983 to 1995 [1.3.5.1-3]. In 1998, a capacity factor of 85% was realized. Scheduled inspection is required every 13 months. If unexpected troubles do not occur and the inspection time is the planned 60 days, the maximum achievable capacity factor will be 87%. The actual capacity factor of 85% was slightly reduced from the ideal value due to various situations.

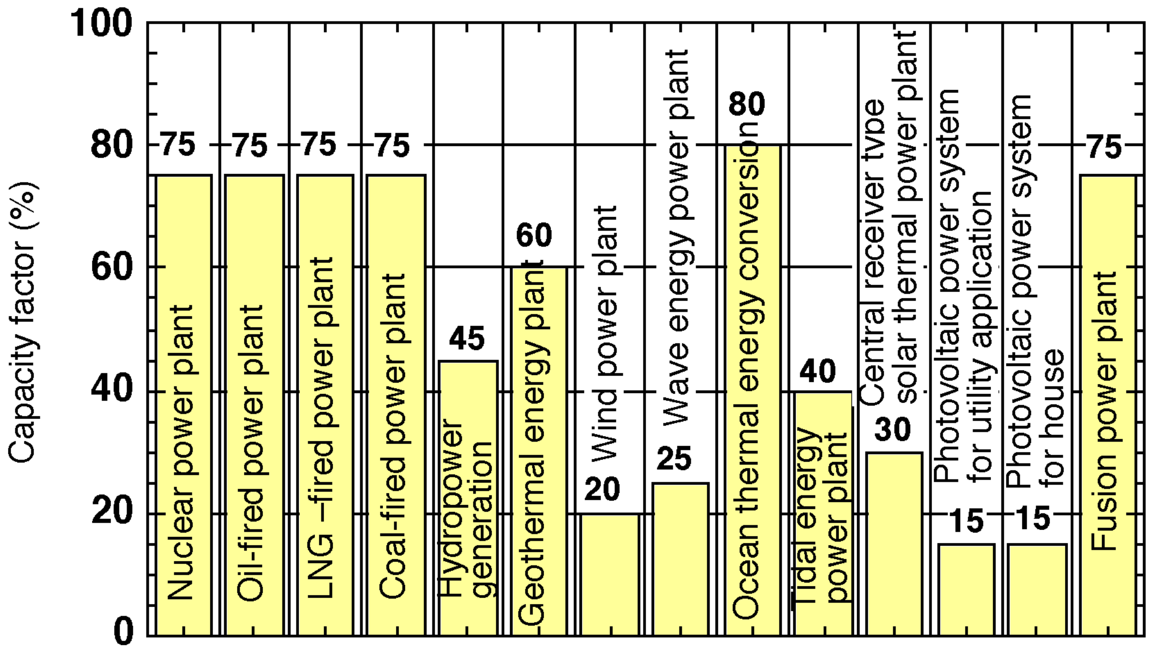

Fig. 1.3.5.1-2 Capacity factor of various kinds of power plants.
(Availability factor is shown for thermal power, value for fusion is prediction)

For a fusion power plant, a capacity factor similar to a (fission) nuclear plant should be possible for actual operation if the maintenance scenario can be planned within a similar periodic inspection and downtime schedule as that for the nuclear plant. In the CREST reactor where the blanket is replaced in the sector unit using a collective method, replacement time for all blankets is estimated to be 65 days [1.3.5.1-4]. Therefore, a capacity factor of 75% can be achievable. As will be described in Section 3.5.2, even if all blanket modules are replaced as in the ITER method, the maintenance time is estimated to be 60~70 days including replacement of half module required for one maintenance period in 28 days and the working days before and after the replacement. The same capacity factor as that of the nuclear plant is also achievable for this maintenance scheme.

(3) Re-circulation Power Ratio

A power plant that requires high circulation power is not desirable as an energy generating system. The re-circulation power ratios of various power plants are shown in Fig. 1.3.5.1-3. The wave energy power, the ocean thermal energy conversion, and the tidal energy power systems have rather high values, 30, 50, and 30%, respectively. For the fusion power plant, the SSTR's value is calculated to be 16.3%, which is a little high. More improvement, e.g., down to ~10% such as the A-SSTR's value, is needed.

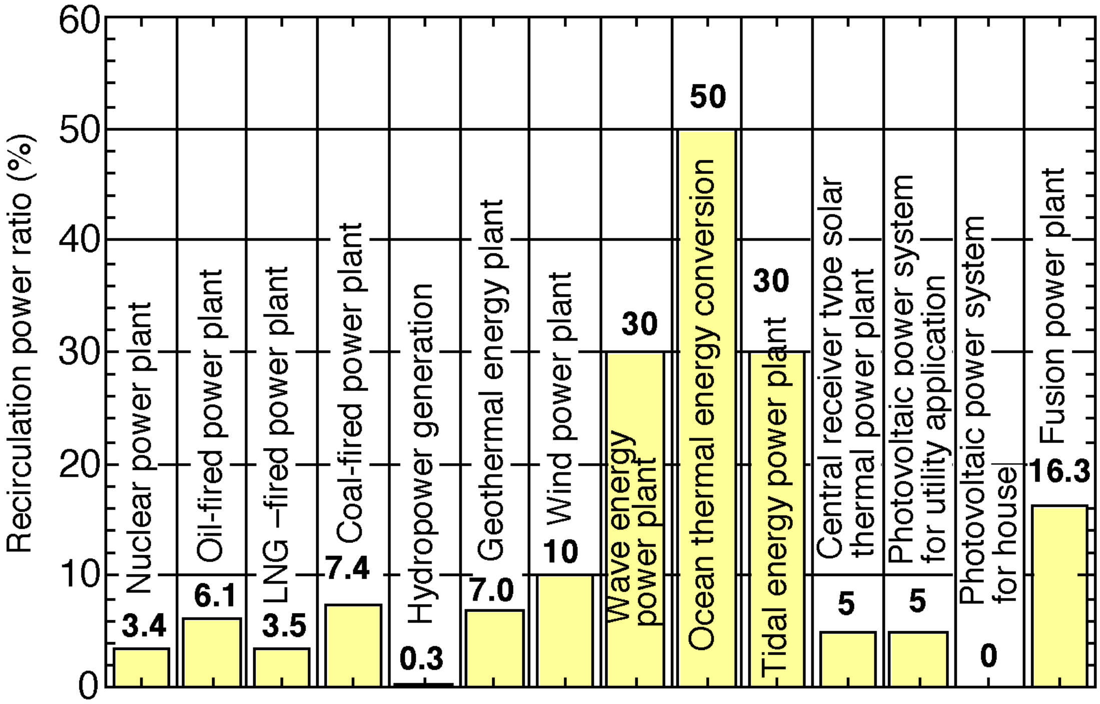

Fig. 1.3.5.1-3 Re-circulation power ratio of various kinds of power plants.

(4) Thermal Efficiency

A high thermal efficiency (thermoelectric conversion efficiency) will enable realization of the compact reactor size, reduction in the COE (cost of electricity), fuel saving, reduction in thermal pollution, and so on. Therefore, high thermal efficiency is desirable for every kinds of power plant.

In a fusion reactor, the nuclear fusion reaction performance is decided almost independently of the coolant situation. This is a remarkable difference from a nuclear fission reactor where the nuclear reaction performance is inseparably related to the coolant characteristics. *For a fission reactor using temperature, superheated steam, there is some possibility of the reactivity accident by the introduction of positive vessel reactivity resulting from the possible condensation of steam to water.* On the other hand, such a phenomenon can never occur in fusion reactor; thus, a fusion reactor can have no reactivity accident. There is a lot of freedom in choosing the coolant in a fusion power plant. In Table 1.3.5.1-1, various coolant types and thermal efficiencies are indicated. Actual selection depends largely on the combination of the coolant and the structural material.

Table 1.3.5.1-1 Cooling condition and thermal efficiency for some tokamak power reactors

	SSTR	DREAM	CREST	ARIES-RS
Structural Material	Ferritic Steel	SiC/SiC	ODS- Ferritic Steel	Vanadium Alloy
Coolant	Water	Helium gas	Superheated Steam	Liquid Lithium
Maximum Coolant Temperature	325°C	900°C	480°C	610°C
Thermal Efficiency	34.5%	45%	41%	46%

(5) Power Controllability including Load-Following Operation

In a magnetic fusion system such as the tokamak device, power control is performed indirectly, e.g., plasma density control or plasma temperature control. Therefore, it seems more difficult in a magnetic fusion power plant than in a fossil fuel power plant or in a nuclear fission plant that have self-regulating control of their output power. It is desired that the fusion power plant also have some possibility of self-control (negative feedback) mechanism like a nuclear fission reactor. A weak instability nearby the beta limit or the temperature dependence of synchrotron radiation may allow such control but the effectiveness of either phenomenon is not understood at present. The feasibility of the burning plasma power control will be clarified by the ITER operation. Load-following operation is performed by simultaneously controlling the plasma power and the particle balances. Thus, load-following operation may be more difficult than constant power operation. For steady-state tokamak operation, power control may be rather easy, since the current driving power is continuously injected into the plasma. It has been demonstrated by computer simulation that controlling the injection power (see Section 3.1.6) performs the load-following operation (power variation range from 50% to 100%).

In a commercial fusion reactor, its shutdown due to a plasma disruption is of course very undesirable. Even if the disruption of plasma cannot be avoided entirely, its impact should be reduced to the same or lower level of the shutdown due to an external accident like lightning.

1.3.5.2 Fusion-Specific Plant Characteristics

A fusion power plant has some unique features that are not common to other existing power plants. Those can be categorized into the four groups below. The first category (1) is devoted to advantages while the second (2) includes both advantages and disadvantages. The third (3) and fourth (4) describe disadvantages.

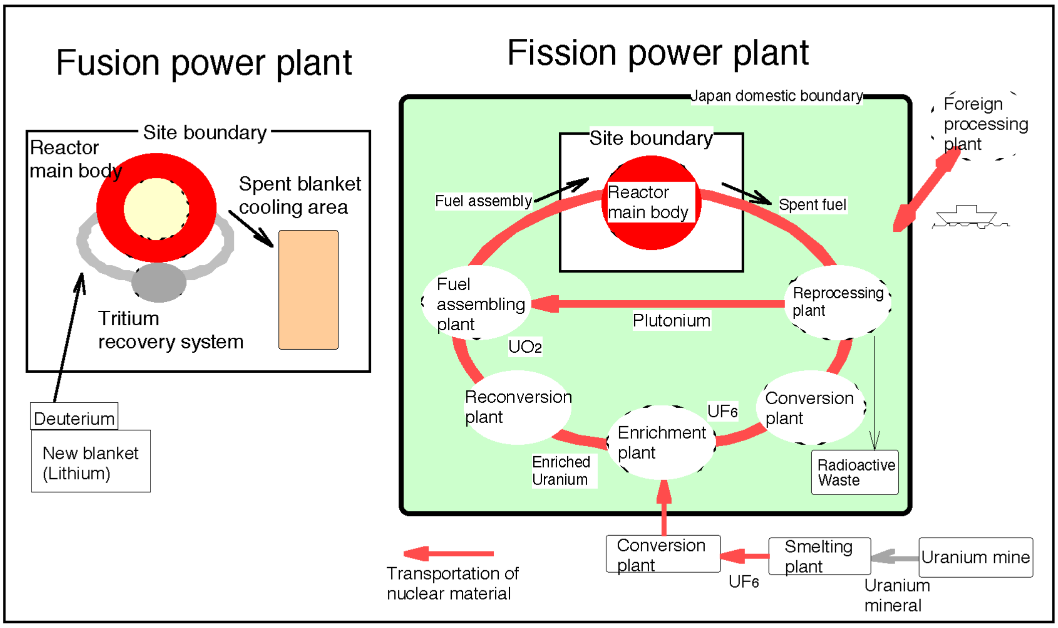

Fig. 1.3.5.2-1 Fuel cycle comparison between fusion and fission power plants

(1) Closed fuel cycle within the power plant

In a fusion power plant, the tritium breeding process, the tritium recovery process, and the tritium separation and purification processes are all performed within the plant site; i.e., the fuel cycle is closed. Furthermore, assuming that the used blanket modules are retained onsite for safekeeping during the plant lifetime, transportation of radioactive substances to and from the site can be avoided after the initial tritium charge. Since this closed fuel cycle should be realizable, it would be a remarkable feature in comparison with a light water fission reactor or an FBR (fast breeder reactor). Further, this would be an advantage from the viewpoint of safety as well as for public acceptance. The fusion plant fuel cycle and the fission plant fuel cycle are schematically shown in Fig. 1.3.5.2-1. The fusion plant has a simple, closed fuel cycle, which should provide high safety and thus would be easy to gain public acceptance.

(2) Scheduled replacement of the blanket structure

The blanket of the fusion reactor is a large, heavy, complicated structure that has many attached piping systems. In the present and usual reactor designs, the blanket structure is to be replaced every few years. The scheduled replacement design concept brings the following merits and demerits.

• Demerits associated with replacement:

It is difficult to obtain high plant capacity factor (higher than 75%), even with various maintenance scenarios. However, this would not be serious as discussed in Section 1.3.5.1. The total amount of radioactive waste is massive at the end of the plant lifetime.

• Merits associated with replacement:

Using this blanket replacement concept, a fusion reactor can be a very flexible energy system. It is possible to test several kinds of blanket designs while keeping the same plasma conditions. For instance, it may be possible to replace an electricity-generating blanket with a new hydrogen-generating blanket to meet future energy demands. Upgrading to a new high-efficiency blanket based on future advanced technologies is also a possibility. Looking at another field, the ongoing re-powering technology applied to convert an old coal-fired power plant to an LNG gas turbine plant is a good illustration. When the coal-fired boiler (which may be the component having the shortest lifetime) must be replaced, a gas turbine system and a new boiler using the exhaust heat from the gas turbine are simultaneously introduced. Called a "gas turbine combined system," this upgrade brings a significant increase in power output and a mitigation of the environmental loading. The outdated coal power plant is reborn as a new gas turbine combined system, where the existing equipment and facility are reused as much as possible. Figure 1.3.5.2-2 shows the concept of the GT26 gas turbine combined system, which was constructed by the ABB Company (Germany). This kind of flexibility is a great advantage especially to provide a long lifetime to a large-scale system, such as a power plant. In a fusion reactor, the nuclear fusion reaction performance is independently decided from the blanket configuration. This is remarkably different from a nuclear fission reactor where the nuclear reaction performance is inseparably related to the coolant characteristics. The flexibility of use for a fusion reactor is schematically shown in Fig. 1.3.5.2-3. The fusion reactor can be designed as a power station, however it can be adapted to meet new socioeconomic demands by changing the type of blanket system during scheduled blanket replacement activities. Therefore, this flexibility might avoid the shortening of the plant lifetime by applying technological evolutions

A fusion reactor itself is characterized as a large, central energy source. However, it can respond relatively quickly, through blanket replacement, to needs other than the demand for electricity. If distributed power sources such as fuel cells fueled by hydrogen or methanol become common, a fusion reactor would be capable of taking the role of an energy station to produce such fuels.

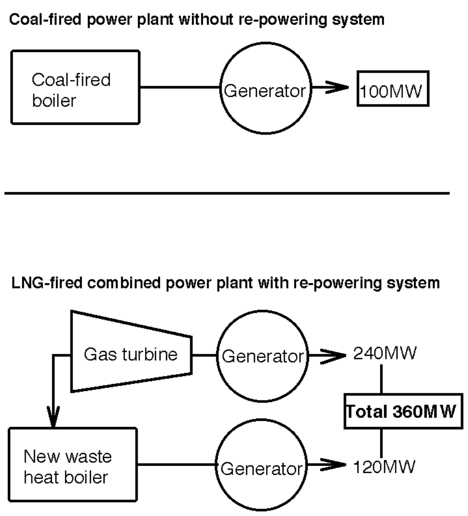

Fig. 1.3.5.2-2 Re-powering system for coal-fired plant. Example of GT26, ABB Company, Germany.

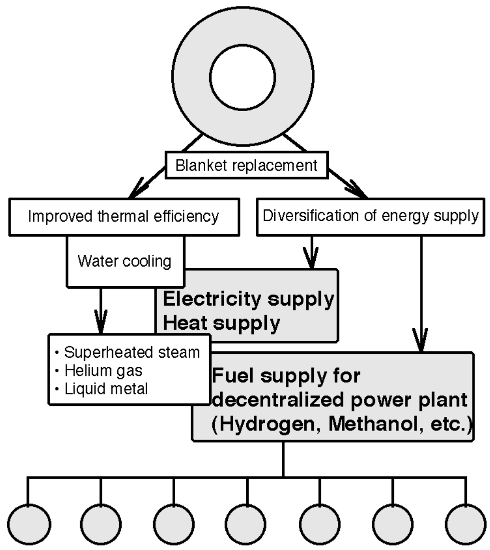

Fig. 1.3.5.2-3 Flexible use system of fusion power

(3) Power required for plant startup

Not only the tokamak plant but also any other type of fusion plant requires a large power supply for plant startup. Future magnetic fusion reactors will require a total starting power of a few hundred MWe for the PF coil power supply and the plasma heating power supply. For example, ITER requires a pulse power of 500 MWe during the plasma ramp-up phase. However, since future power plants with steady-state operation will allow a slow plasma ramp-up scenario, the start-up power will be much less. The power required for SSTR plant startup is estimated to be ~200 MWe.

This startup power must be provided by a start-up power generator, an energy storage device, or directly from the existing power distribution grid. The first two methods require a significant additional capital outlay, and the last imposes a restriction on plant operation. Different from emergency shutdown, the start-up operation will have to be made according to a schedule. Start-up power for a fusion reactor does not involve insurmountable problems if startup is done based on a schedule and with attention to the power grid flow. For a power grid having a small capacity, such as in an isolated local area, siting the fusion plant in combination with another type of power generation system should be considered.

(4) Uniqueness of tritium fuel

Tritium, which is the fusion reactor fuel, does not exist in nature. The required tritium is to be produced within the reactor itself in a working fusion plant. The initial tritium charge for a new plant will be supplied by a preceding fusion plant. Since the tritium, a radioisotope of hydrogen, has a rather short half-life, 12.3 years, the loss of tritium in storage over time cannot be ignored. In such a sense, tritium is a unique fuel for a large-scale power plant.

Fuel stock:

When fusion power comes into widespread use, existing fusion plants will supply the initial tritium charge for a newly built fusion plant and the fuel stock will not be a serious issue. For the first fusion plant, sufficient fuel stock would be needed for a stable operation. The required fuel stock is estimated to be the quantity required for 50 days of operation [1.3.5.1-5].

Plant doubling time:

The propagation speed of new fusion plants might be restricted by the tritium-breeding ratio since the fusion fuel is bred by a fusion plant itself. Therefore, it is necessary to analyze what propagation speed can be realized. The dependence of power capacity growth on the tritium doubling time is shown in Fig. 1.5.3.2-4.

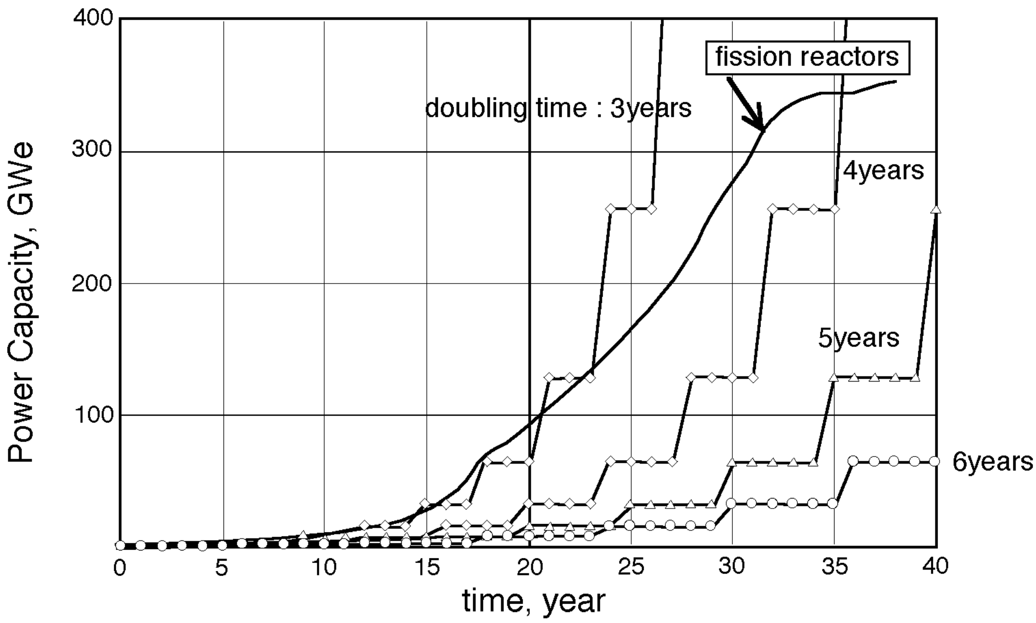

Fig. 1.3.5.2-4 Growth curve of nuclear power plants (solid thick line) and fusion power plants. For fusion, the dependence of the power capacity growth on fuel breeding doubling times is illustrated.

Here, the doubling time is the time required to accumulate an initial tritium charge. For comparison, the achievements of fission power plants are also shown. A fusion plant propagation speed with a tritium doubling time of 4 years corresponds to the past fission achievements of 15 GWe/year during a 20-year period (from 10 to 30 years in the figure). Here, the availability of the fusion plant is assumed to be 100%. When the plant capacity factor is 75%, a doubling time of 3 years is needed for the same achievement.

The tritium doubling time depends on the amount of initial charge (loading). Smaller initial loading results in a shorter doubling time, although the initial loading and doubling time are related to the fuel stock for each plant. The required tritium-breeding ratio has been evaluated under the following assumptions [1.3.5.1-5].

• Tritium stock is equivalent quantity required for 50 days operation.
• Doubling time is 3 years.
• Recovery time is 1 year to compensate for the adsorbed tritium.

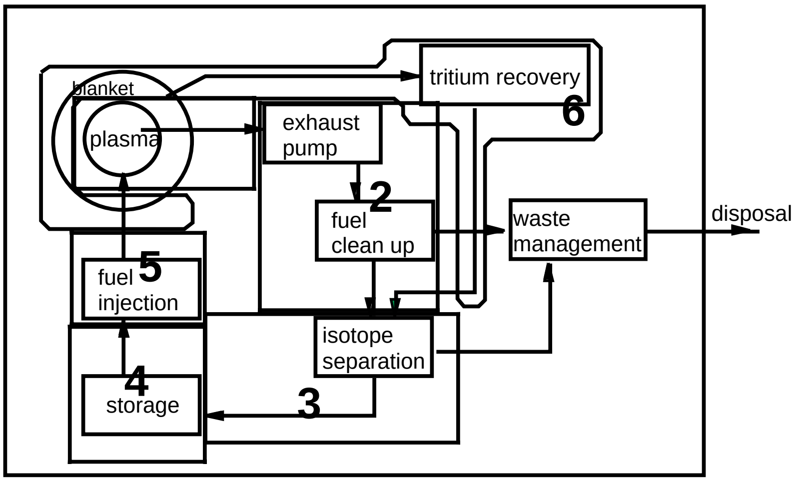

Fig. 1.3.5.2-5 Schematic of tritium inventory model Sojourn time where absorption, decay loss, etc., are considered

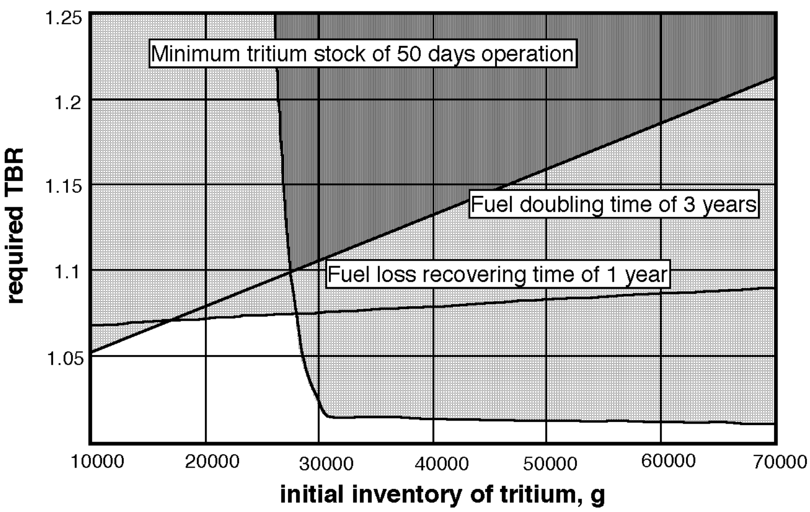

Fig. 1.3.5.2-6 Required tritium breeding ratio of newly constructed fusion power plant

The calculation model and the results are shown in Fig. 1.3.5.2-5 and Fig. 1.3.5.2-6, respectively. The required breeding ratio shown in the figure is 1.1. This ratio is decided by the stock and doubling time requirements. According to current design reports on the fusion blanket, the breeding ratio of 1.1 is an achievable value. This calculation assumed the above stock requirements for each reactor. When a number of plants are constructed in a same region or the fuel stock is secured outside the plant site, the stock requirement can be relaxed. Thus, the plant propagation speed can be expedited and/or the TBR (tritium breeding ratio) requirement can be mitigated.

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1.3.6 Economic Efficiency

It is quite difficult to predict the worldwide energy situation in the latter half of the 21st century. However, it may be possible to enumerate the minimum requirements for fusion energy that would make it acceptable as a commercial energy source. Especially focusing on the economic aspects, the allowable band of the COE (cost of electricity) can be predicted in comparison with other exiting and future power plants.

1.3.6.1 Fusion Competitors and COE

Provided only the electricity power supply system is considered, fusion power plants and their competitors will be selected as viable power plants if they satisfy the following four conditions.

1) Resource (fuel) availability for several hundred years.
2) No environmental impact (or minimal unavoidable impact).
3) Competitive COE from the power source.
4) Large and stable power supply to become an essential power source.

Oil-fired power plants do not satisfy above-mentioned 1) resource issue, judging from the estimated available reserves. Fission power plants using natural uranium from mines have the same problem without using plutonium fuel or recovered uranium from seawater. Either FBRs (fast breeder reactors) or uranium recovery technology from seawater should be put to practical use to satisfy condition 1).

The COE and the unit cost of construction for the various power generation systems including natural energy sources have been evaluated based on reference [1.3.6.1-1]. The results are shown in Fig. 1.3.6.1-1. The COE is normalized (expressed as COEn) by the present COE (~10 yen/kWh) of a coal power plant without a CO_2 sequestration system to eliminate changes in the value of currency. The future costs of fossil fuel and uranium fuel are assumed to be same as the present costs, since fuel costs depend not only on supply-demand factors but also on international political factors, the progress of mining technology, currency exchange rates, and so on. It is too optimistic to assume that price of the fossil fuel will elevate significantly in the future. Further, the cost of uranium will not increase above that of uranium extracted from seawater.

Nuclear power plants, fossil fuel power plants, and hydroelectric power plants that are now in use are categorized as "Power plants now in use." Photovoltaic and wind power plants, both with the present value and with the future value reduced by expected technical progress, are shown. The photovoltaic system shown in Fig. 1.3.6.1-1 is a large-scale plant for industrial use. Considering a household solar generator, since the cost of the house roof is not taken into account, the cost is half that for industrial use. Except for the geothermal power, natural energy systems are subject to low capacity factor and large plant size. Therefore, it may be difficult for the natural energy systems such as the photovoltaic plant to satisfy the above mentioned 3) COE issue and the 4) supply issue (the cost here does not include costs for the power output smoothing equipment such as the energy storage system). For the wind power plant, the 1994 presumed value is shown in this figure. Based on a more recent, 1996, COE of 17.9 yen/kWh for a small-scale 100 kW-level plant, the future COE of wind power can be reduced. A wind power plant with a market price of less than 10 yen/kWh is reported, but 1/3 to 1/2 of the construction cost is financially assisted by the Government in this case. Suitable domestic locations for wind power plants are limited and localized. Therefore, a wind power plant does not satisfy the above-mentioned 1) resource issue. A geothermal power plant, which is sufficiently competitive, is now in the pilot plant phase. Nevertheless, its supply capacity in Japan is only 1% of the total domestic generation capacity. The use of geothermal power will continue, but it cannot be a major energy source due to its site issue. The location of the fusion power target position shown in Fig. 1.3.6.1-1 will be described later.

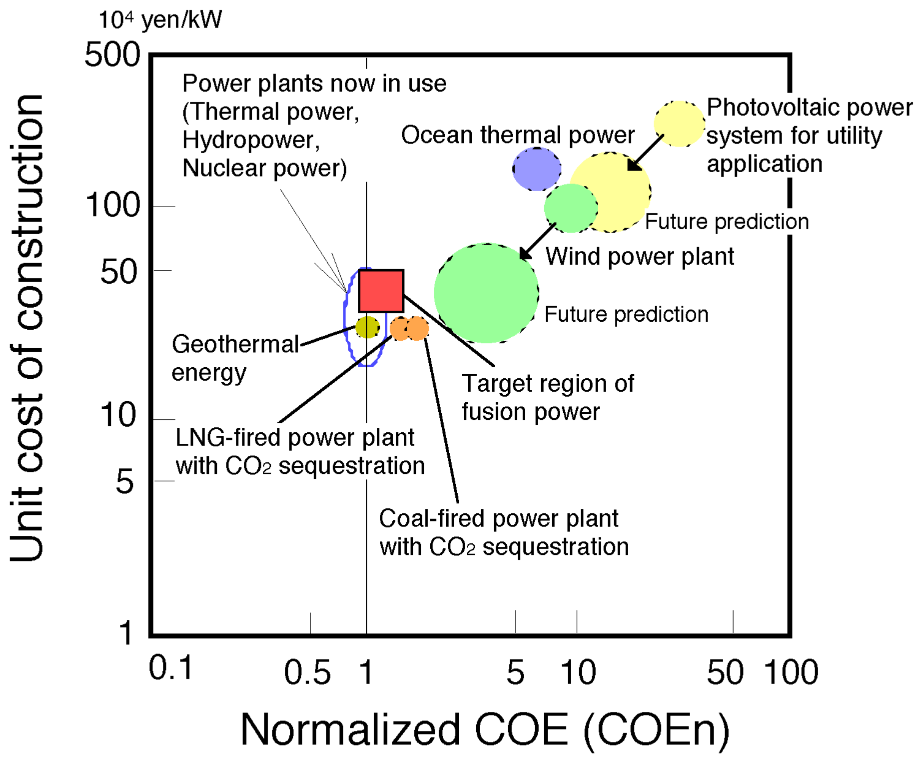

Fig. 1.3.6.1-1 Positioning of various kinds of power plants in a COE (cost of electricity) - Construction cost diagram. Values of photovoltaic and wind are 1994 predicted values from reference [1.3.6.1-1]. There is another optimistic prediction, an existing small-scale wind power plant achieved a COEn < 2.0.

As a conclusion to above discussion on resources and cost competitiveness, the following two types of plants might be strong competitors with fusion power plants.
1) Fission power plant (FBR, LWR-SW with seawater uranium recovery).
2) Natural gas or coal plant with CO_2 sequestration system.

Fossil fuel is exhaustible. However, the estimated amount of coal reserves corresponds to a few hundred years of use. For natural gas, the confirmed reserves are 144 Tm^3, which corresponds to 62 years of use. However, the reserve numbers increase with each annual investigation. Furthermore, non-customary natural gas such as methane hydrate, free gas (methane), or coal-bed methane can be used depending on their future extraction costs. Although the amount of reserves for non-customary natural gas are not well confirmed, the expected amounts are at least equivalent to that of customary natural gas. Although the extraction costs of the non-customary natural gas is not well known, there should be no serious obstacle for the development of one or more of these sources. Hence, the depletion of natural gas will not occur within several tens of years. Therefore, natural gas may be a serious competitor of the fusion plant if its commercialization is addressed in the mid-21st century.

Figure 1.3.6.1-2 shows the future COE predicted by an extrapolation of the present technology database. The FBR cost is based on the high performance concept ARES [1.3.6.1-2] where the secondary cooling system is excluded and metallic fuel is introduced. Though the uranium recovery cost from seawater is predicted to be 5.7 times higher than the present mining cost, the calculated COE in Fig. 1.3.6.1-2 corresponds to uranium costs 3 to 10 times higher than present costs [1.3.6.1-3, 1.3.6.1-4]. For the LWR, two kinds of construction costs are considered. One is the present database (COEn=0.9). The other is for the large scale ABWR with an output of 1.35 GWe, where a 30% cost reduction can be realized. Two points are concluded from Fig. 1.3.6.1-2.

1) If fusion power plants are forced to be competitive only for the COE issue, a COEn of 0.5~0.7 must be realized in future.

2) If fusion COEn will be much more than 1.5, fusion will be noncompetitive. Even if fission plants will be unavailable for one reason or other, the fossil power plants with CO_2 sequestration systems will need lower cost than the fusion plants. Furthermore, the cost of CO_2 sequestration will be reduced in future.

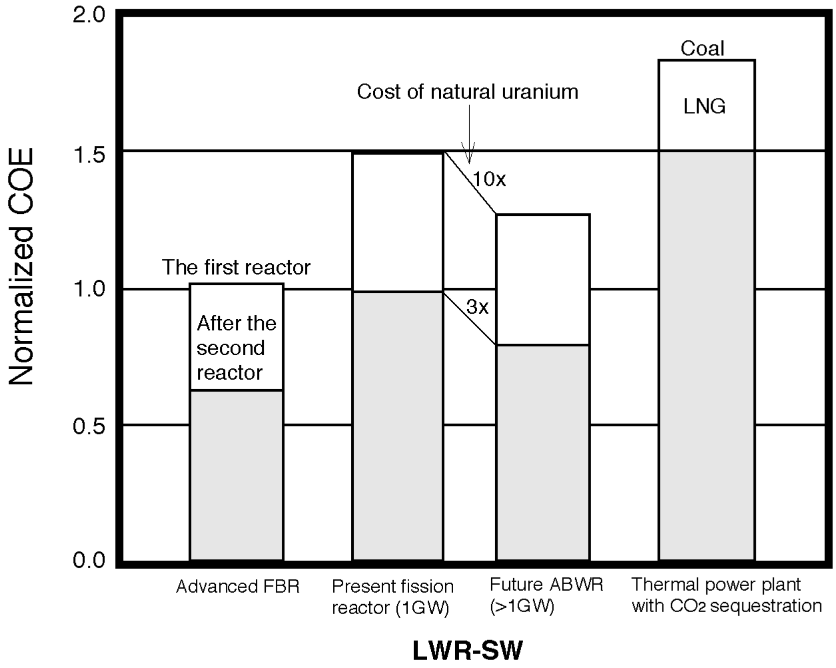

Fig. 1.3.6.1-2 Predicted COEs of future power plants competing with fusion power plants

As a conclusion from above results, the fusion cost target should be much less than a COEn of 1.5. Otherwise, the commercialization of fusion will be quite difficult for at least a few hundred years. For the long term, the cost target should be around a COEn of 0.7.

As seen from the figure, direct COE competition with LWRs and FBRs is difficult for COEn<1.5. From the safety and environmental points of view, the fusion power plant is potentially superior to the LWR and FBR power plants. Therefore, the siting conditions can be alleviated and the decommissioning costs can be reduced in comparison to the LWR and FBR power plants. In Japan, LWR plants are not sited near big cities from a public acceptance point of view(Near-city construction of LWR is technically possible if some legal issues are solved. But such an application has never happened in Japan). Instead, the plants are located some 300-km from the major consuming areas. If the power plant construction is allowed in the big city environs, the power transmission costs will be reduced somewhat. In Japan, the power transmission cost is estimated to be 1.17 yen/kWh based on the securities report of 1995 (average of 10 electric power industries including suburban fossil fuel stations). The power transmission cost for 10 GWe of power over a distance of 600 km is almost same as the construction cost of the 10 GWe nuclear power plant producing the power. Thus, the present COE of the LWRs may have been increased more than 1.5 times the intrinsic COE. Therefore, if fusion power plants are constructed with COEn values less than 1.5, there is a chance for fusion plants to compete with the LWR and FBR power plants when total costs are considered.

The disposal cost of the recovered CO_2 is evaluated as about 1 yen/kWh based on a sea dump. However, since there is some possibility that a new environmental problem will be caused by the sea dump, the sea dump sequestration method may not be feasible.

1.3.6.2 Economical Prospects of the Fusion power plant

The construction cost and the COEn are evaluated for three types of tokamak power plants that have been designed in our country. The results are listed in Table 1.3.6.2-1 [1.3.6.2-1, 1.3.6.2-2, 1.3.6.2-3]. The type A is the conventional model based on the experimental reactor (such as ITER) database. The type B is the moderate model based on a somewhat advanced database. The type C is the advanced model aiming for a high economic superiority. As a common parameter among them, the electricity output is 1 GWe (fusion power of 3 GW and thermal efficiency of 34.5%).

Table 1.3.6.2-1 Cost comparison between typical three types of tokamak reactors

	A type [1.3.6.2-1]	B type [1.3.6.2-3]	C type [1.3.6.2-2]
	Reactor based on ITER physics Conservative model	Demonstration reactor (SSTR) Moderate model	Economical reactor with reversed shear configuration Advanced model
Major radius	7.46 m	7.0 m	5.4 m
Electricity output (net)	1000 MW	1080 MW	1000 MW
Maximum toroidal field	12 Tesla	16.5 Tesla	12.5 Tesla
Normalized beta (β_n)	3.0	3.5	5.5 (reversed shear)
Construction cost	980 BYen	720 BYen	480 BYen
Unit cost of construction	980 kYen/kW	660 kYen/kW	480 kYen/kW
COEn (lower value ~ upper value)	2.16~2.70	1.6~1.8	1.04~1.30
Unit cost of construction / COEn	445~363 kYen/kW	413~367 kYen/kW	462~369 kYen/kW

In these calculations, the construction (total capital) costs, the operation and maintenance costs, the fuel costs, the taxes, etc., are considered. The COEn depends on the operation and maintenance costs that are decided as a certain percentage of the construction cost. For the lower and higher cases, 2% and 4% are used, respectively. Since the COE of a fusion power plant is mainly decided by the construction costs and the operation and maintenance costs, the unit cost of construction (construction cost / electricity output) per COEn of the unit becomes almost same value (0.36 ~ 0.46 MYen/kW) among them. Based on this empirical scaling, the relationship between the unit cost of construction (or COEn) and the construction cost can be shown as such in Fig. 1.3.6.2-1. Here, the target region is also shown. The lower and upper lines denote the electricity output of 1 GWe and 2 GWe, respectively. The A-SSTR with an output of 1.7 GWe and a construction cost of 51 BYen [1.3.6.2-4] and the CREST with an output of 1.16 GWe and a construction cost of 49 BYen will plot inside the target region. The above-mentioned types A, B, and C are also plotted. The difference between the CREST and the type C reactor is caused by the thermal efficiency improvement (increase from 34.5% to 41%) and the electricity output increase (1 GWe to 1.16 GWe). The LWR is also plotted for reference. If the LWR is able to win public acceptance, it will be the most promising energy source from the viewpoints of the CO_2 and cost issues. The target region in Fig. 1.3.6.2-1 corresponds to the target region in Fig. 1.3.6.1-1.

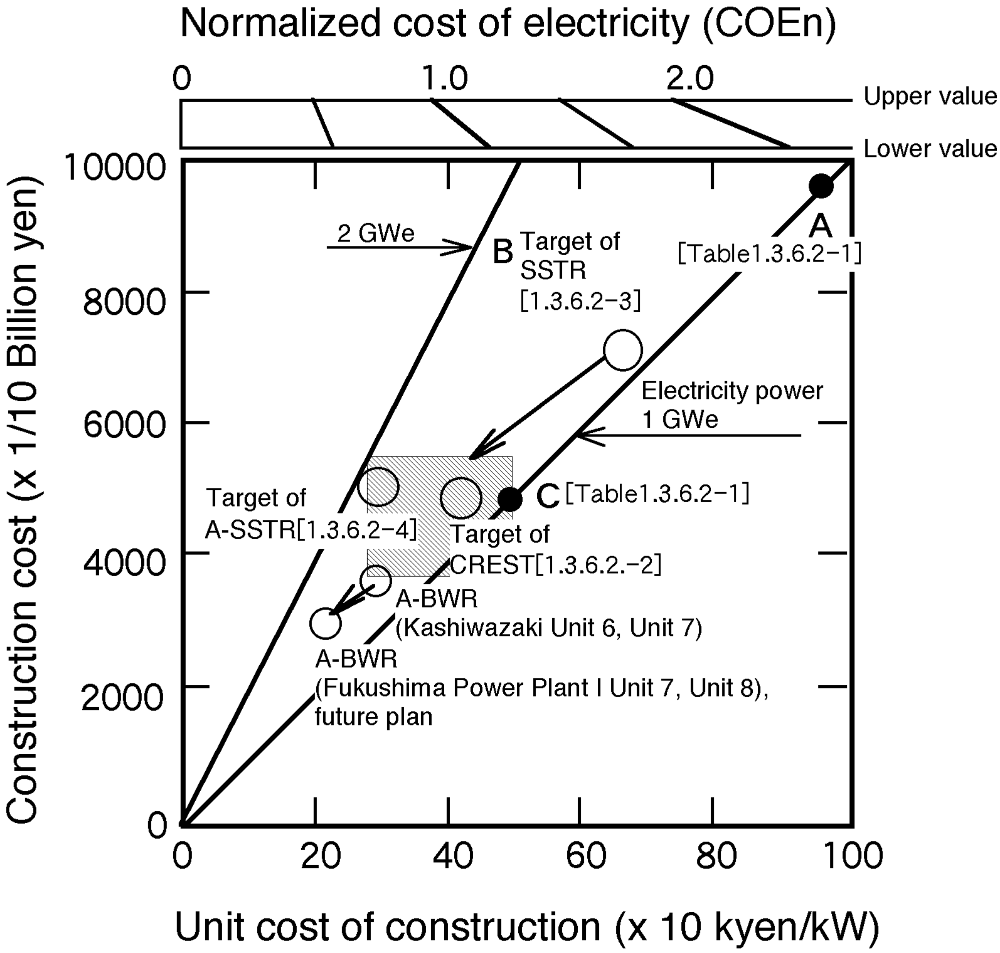

Fig. 1.3.6.2-1 Cost related target region for fusion power plants

The construction cost items for the A-SSTR and the CREST are listed in Table 1.3.6.2-2. The reactor equipment cost is about half of the total cost. Since most of the reactor components such as the super-conducting coil and the neutral beam injector require high technology, the maturation of technology will reduce the construction cost.

Table 1.3.6.2-2 Construction cost breakdown of tokamak power reactor A-SSTR and CREST

	A-SSTR (1,700 MW)	CREST (1,160 MW)
Total construction cost	510 BYen	490 BYen
Reactor equipment and Auxiliary system	49%	44%
Land and Buildings	8%	11%
Turbine and Electricity system	29%	32%
Indirect costs and Interest	14%	13%

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1.3.7 Use of Fusion Energy in Forms other than Electricity

Because fusion reactors have a low radiological toxic hazard potential, the following unique features are expected; 1) it may be possible to site plants closer to the consumer, 2) the temperature of the product heat is selectable by blanket design, material, and coolant media regardless of the design of the reactor core plasma, and 3) the reactor can be used as a large-scale fast-neutron source. However the development of the fusion reactor will require a long R&D period, 20 years or more for the experimental reactor ITER, and 40 years or more for the DEMO reactor *and the first commercial reactor*, even if the program proceeds according to the expected schedule. The target of the development of fusion energy during this period must be flexible to accommodate worldwide changes in energy consumption, and innovations in technology related to existing/new energy (moving target).

For this purpose, it is desirable to pursue fusion reactor concepts as more attractive electricity generation systems and to investigate the various uses of fusion energy other than electricity and neutron applications. Fusion energy can be used mainly as a heat source and a neutron source when considering applications other than production of electricity. Because a fusion reactor has a very small radiological toxic hazard potential, which may enable a fusion energy source to be located in a suburban site, such applications in these areas should be promoted. This is an advantage of fusion energy, and makes a fusion reactor even more attractive. However, the research of application of fusion energy for other than electricity is still in a very early stage, and development of various blankets will be needed.

1.3.7.1 Use of Fusion Reactor as a Heat Source

Since the fusion reactor blanket is an essentially replaceable component, changing the temperature range of operation or changing the application between a heat source and electricity generation are possible even after construction to respond improvements in technology or changes in demand. In addition, different types of blankets can be installed in a reactor for multiple uses at the same time. This is one of the major advantages of fusion reactor.

(1) Use of heat at low temperature (~250ºC) region

In this temperature region, heat is mainly used for community heating in the form of high temperature water.

(2) Use of heat at medium temperature (~600ºC) region

Typical industrial uses for heat in this temperature region are petroleum distillation, petrochemical synthesis, liquefaction of coal, gasification of oil, paper/pulp processing, desalination of seawater and chemical synthesis, such as the manufacture of fertilizer. Heat can be used in cascade, and electricity generation is possible as a desired fraction. From the viewpoint of electricity generation, the use of heat can effectively work as a buffer for changing electrical loads. Thus, multiple purpose use of heat may make the fusion reactor more attractive. The present-day heat source for industry is from fossil fuels as the use of nuclear heat is not advantageous from a cost point of view. However, to reduce the greenhouse effect gas from sources other than the generation of electricity and to preserve fossil fuels, it is desirable to use this form of nuclear heat for industry applications. If a high carbon tax were applied, applications of this source of heat will be advantageous from a cost perspective.

(3) Use of heat at high temperature (>800ºC) region

• Hydrogen production

Electrolysis of water vapor at a temperature as high as possible for hydrogen production is an excellent application. Electrolysis at 1,400ºC is 30~40% more efficient than electrolysis at lower temperature. The hydrogen product can be used for making iron by direct reduction, and for the production of ammonia, methanol, or methane.

• Gas production from fossil fuel

The reaction between 900~1,100ºC water vapor and the carbon in charcoal is called coal gasification. The gas product becomes the raw material methane or another reactant gas for a reforming reaction with oil. Since the efficiency for gas production strongly depends of the gas temperature, the fusion reactor that generates a high temperature is advantageous.

• Steam reforming with natural gas

Lower hydrocarbons such as natural gas can react with vapor at temperatures of 700~900ºC to form hydrogen. The gas product can be used for synthesis of ammonia, methanol, methane, or other liquid synthetic fuels.

Fusion energy can contibute to solve some global environmental problems by using fusion energy as a multiple purpose heat source. Fusion energy itself is a clean energy source for the production of electricity without generating carbon dioxide. Also, by the use of heat as explained above, fusion energy could reduce the consumption of fossil fuels, and synthesize fossil fuel materials to form relatively low carbon fuel materials. Although it may be difficult from the aspect of cost, using fusion energy it is technically possible to provide a low carbon fuel for industries that require very high temperatures (usually using fossil fuels) such as the manufacture or processing of metals, ceramics, or cement. Moreover, desalination of seawater using fusion energy can provide large quantities of fresh water, which could be used for irrigation of a plantation in the desert, or producing forests that generate a "sink" for carbon dioxide. Thus, fusion energy as a multi-purpose energy source is expected to contribute to the conservation and regeneration of the global environment in various and manifold ways. Examples of the above multiple uses of fusion energy are illustrated in the Fig.1.3.7-1.

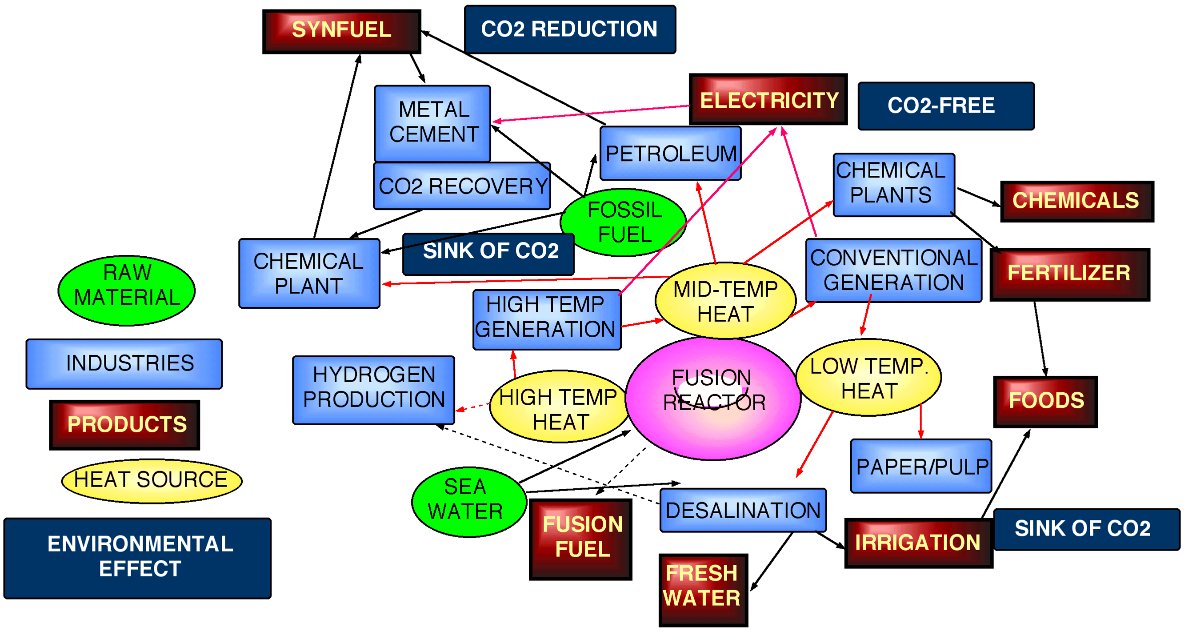


Fig. 1.3.7-1 Multipurpose use of fusion energy

1.3.7.2 Transformation of Minor Actinides

A fusion reactor has the capability of generating one order of magnitude excess neutrons compared to a fission reactor having the same thermal output. With the development of advanced materials and when high neutron wall loading becomes possible, considerable fast neutron flux will be available. A fusion blanket has a large surface area and volume, so a large neutron irradiation volume is available in the blanket. Using these advantages, transmutation of minor actinides contained in the high-level radioactive wastes from a fusion reactor is possible.

An example of the potential for transmutation of actinides is shown [1.3.7-1]. Assuming a neutron wall load of 10 MW/m^2, the reduction of the total amount of actinides by neutron irradiation for 3 different blankets is shown in the Fig. 1.3.7-2. The blankets are a thermal neutron type blanket with graphite moderators, a pure fusion type reactor blanket that simulates a fusion reactor neutron spectrum, and a fast neutron type blanket that simulates a fast breeder neutron spectrum. Table 1.3.7-1 compares the effectiveness of these methods in transmuting TRU for one year. The thermal neutron type blanket has a fast transmutation speed, but is not suitable for processing a large quantity of actinides since the loading amount is rather limited because of the fast reduction to thermal neutrons. The fast neutron type and fast breeder type need tens of years of irradiation, but make it possible to transform large amounts of minor actinides. In this case, neutron wall loading is assumed to be 10 MW/m^2, but even with a lower neutron loading, a fast neutron type fusion blanket would be expected to process a considerable amount in a year.

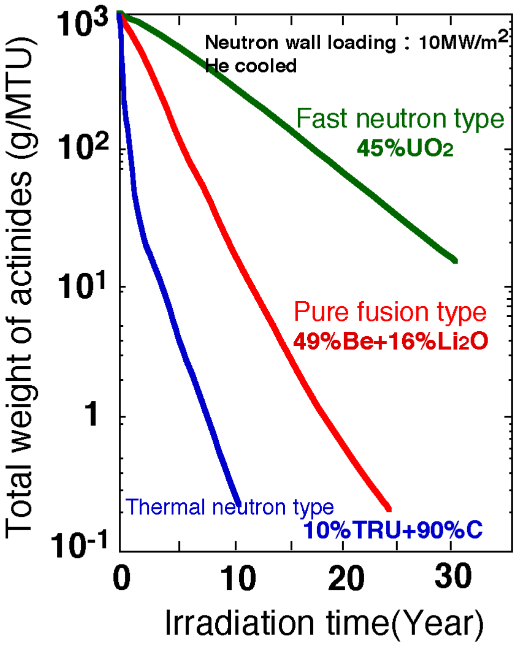


Fig.1.3.7-2 Calculation of transmutation by fusion reactor

The advantages of the transmutation with fusion reactor are as follows. 1) Relatively high transmutation ratio and possible complete transmutation. 2) Amount of actinide loading is larger than other methods. 3) Energy balance is good and simultaneous electricity generation is possible. 4) System is sub-critical and there is little concern of reactivity accidents. On the other hand, indicated disadvantages are; a) Need to develop materials to withstand high neutron flux, b) Requires sophisticated group separation following irradiation, c) Need to develop blanket technology compatible with tritium breeding and heat removal.

From above considerations, a fusion reactor has the potential capability for transmutation of minor actinides. Considering the identified disadvantages, the application of a fusion reactor for transmutation is complimentary with a fast reactor. The development of transmutation processing with a fast reactor needs to be advanced while the development of an optimum blanket for transmutation in conjunction with development of first wall material for fusion reactor technology is being advanced.

	Thermal neutron type	Pure fusion type	Fast neutron type	FBR
Total neutron flux ($10^{14}/cm^2s$)	6	17	41	50
TRU effective half life (Year)	0.34	1.6	4.7	4.7
Annual transmulation (g/kgTRU)	870	350	140	140
Initial TRU loading (t)	0.28	—	140	6

Table 1.3.7-1 Comparison of amount of transmutation of TRU by various techniques

1.3.7.3 Production of Radioactive Isotopes (RIs)

Since a fusion reactor is a large fast-neutron source, it is possible to produce a significant amount of radioisotopes that cannot be made with fission reactors. Parent nuclides are loaded in a fusion reactor blanket and extra neutrons are used for the production of the RI. With a fusion reactor of 1~4 MW/m^2 neutron wall loading, specific activity of 10~40 Ci/g cobalt-60 can be produced at the rate of several tens of millions of Curies per year. With an optimized blanket, 200 Ci/g Co-60 can be produced at several hundred MCi/y [1.3.7-1]. Molibdenum-99 can be produced in large quantities. Moreover, by irradiating relatively inexpensive rhenium, the precious metal osmium can be made with the excess neutrons of a fusion reactor.

Neutrons generated by a fusion reactor have an energy of 14 MeV and have larger reaction cross sections for (n, p), (n, α), (n, 2n) than those from fission neutrons, thus it is possible the fusion reactor may be used for the production of special RI. It is necessary to develop efficient production technology through the compiling and evaluation of nuclear data and the design of blankets for this purpose.

1.3.7.4 Development of Multipurpose Use in the World Market

It is expected that the above multipurpose fusion reactor may be more valuable and attractive in developing countries than in developed countries. The low radiological toxic hazard potential is meaningful in a country unwilling to assume greater radiological risks. In addition, resilience to proliferation of nuclear materials and the elimination of transportation of spent nuclear fuels are important from the aspect of security. Since the fusion fuel cycle can be contained within a closed facility site, it can be self-dependent and is not affected by external fuel supplies. This feature improves the energy security of developing countries, and is expected to contribute to the regional political stability of such countries. Desalination of seawater could produce drinking and irrigation water particularly in areas with large populations or in arid regions. The possibilities for applications in a developing country will drastically increase the potential demands for a fusion reactor compared with the limited demand for electricity. This form of fusion reactor also enhances the attractiveness of fusion in added value, utility value, and socioeconomic value more than simply as a source of low-cost electricity. Particularly in this case, fusion will be regarded as an effective energy source that contributes a stable affluent future for the populace in developing countries, especially for alleviating food problems and improving the economic growth and living standard.

It is necessary to advance fusion reactor R&D in the stepwise phases of the ITER experimental reactor, the DEMO reactor, and the commercial reactor to make the fusion reactor an attractive electricity generation system. In each step, the projected mission must be completely achieved and its outcome must be reflected effectively in the next phase. The research and development of multipurpose applications of fusion energy must be promoted in conjunction with the above phased development program for the electricity production system. Tests of the blankets for various purposes may be considered in the experimental reactor phase. In addition, as in the case of RI production, technology that requires flexible operation must be thoroughly planned to fully use the demo reactor phase. It is necessary to rapidly feed back the resolution of each technical issue effectively and efficiently to the entire fusion reactor research community. Fusion reactor development must be promoted in a phased approach, and must effectively and rapidly disseminate the various scientific and technical outcomes of the results, especially the aspects of multipurpose uses of fusion energy

Reference:

[1.3.7-1] Direction for the multipurpose use of fusion reaction, Japan Nuclear Industry Forum, March 1988.

1.4 Overall Assessment

1.4.1 Providing Major Potential Energy Option

The discovery of huge amounts of fossil fuels and the improvement of technologies for their use in the 20th century increased the consumption of energy in the fields well beyond that of electricity. As discussed in the Section 1.3.1.2, these resources will not be exhausted for about 200 years, but this period is just a moment compared with the long history of humankind (Fig. 1.4.1-1, [1.4.1-1]).

On the other hand, fossil fuel energy is an energy source that has large CO_2 emission unit that is regarded to cause the warming the of global environment. The large consumption of fossil fuels carries with it the potential danger of causing a catastrophic global-scale climate change.

In the case of fission nuclear-powered electricity, even Light Water Reactors can be a virtually inexhaustible energy source if the recovery of uranium from seawater proves to be economically feasible. However, although fission-powered electricity is operated with maximum precautions to secure safety, the public support for nuclear power is waning due to safety concerns about the perceived large biological hazard potential to both the public and the environment.

Renewable energies such as photovoltaic or wind power are recently of interest, but are essentially unstable in supply and have a reliability problem as a large-scale main energy source.

Thus, there remains a large uncertainty in the long-term energy strategy in the 21st century and beyond. As discussed in the Section 1.3.3, Fig. 1.4.1-2 [1.4.1-2] compares fusion power with two present-day major electricity sources (light water reactors and fossil fuel power stations) from the viewpoint of two major potential risks. When the global environmental problem (warming) becomes serious and hence fossil fuel power must be limited, or when the fossil energy era has ended a few hundred years from now, which is only a moment in the long history of humans, the necessity to secure alternative energy options will be understood. The interim report of the Special Committee for ITER Project states that "Funding for ITER is insurance for energy," and that should be understood as an excellent point of agreement on which the majority can agree.

Fusion development is important from the viewpoint of energy security. To develop fusion as a major potential energy option for the future will insure the uncertain future, remove concerns, and provide a defensive power against disadvantages that could be caused by unexpected events. In the sense of security or defensive power, the fact that fusion could be a "back stop technology" that could be our main energy source when all other energy sources are not or cannot be available. For fusion to be more than a potential option, it is necessary to solve the many issues required for the actual production of energy. It is necessary to develop fusion toward its targeted cost, and the present effort to reduce the cost of ITER is thus important.

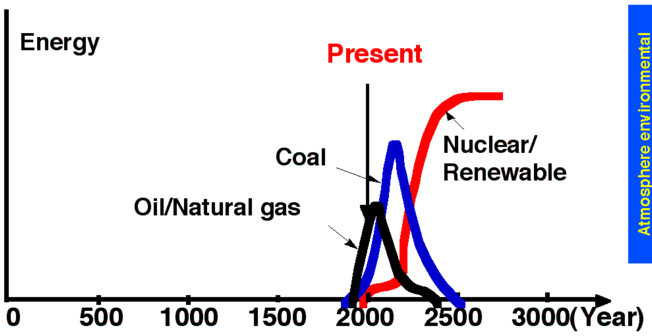


Fig. 1.4.1-1 Fossil fuel era as a moment in human history [1.4.1-1]

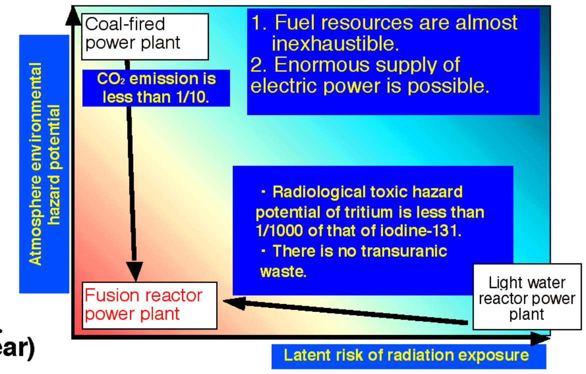


Fig. 1.4.1-2 Fossil fuel, fission, and fusion from the viewpoint of two major potential risks for the energy systems in the 21st century [1.4.1-2]

For the funding for fusion not to be a gamble but to be meaningful to provide a possible option, it is important to recognize that even if fusion energy proves to work it may fail if it is not eventually competitive in the market. In thinking of this as insurance, the coverage must large, broad, and long. The capability to insure society requires fusion power has the potential to be a main energy source. This can be technically foreseeable by the successful completion of the third phase program, the core of which is ITER. From the viewpoint of the recovery of funding for the research and development, it is important to show the capability to reliably supply energy, even if the cost of the electricity produced would be expensive. For fusion to be available when other energy sources are not, it is necessary that fusion be free from the limiting factors that are anticipated to cause the downfall of these other energy sources. Fusion does not essentially involve exhaustible resources, CO_2 emissions as in fossil energy, potential hazards, or spent nuclear fuel, and thus could be dependable for very long period. Another important technical point is the capability to substitute for an existing energy source. The capability of fusion to provide energy is currently anticipated to be as a base load source of electricity, that would be a substitute for nuclear or fossil fuel powered electricity. Fusion may also provide electricity by the load follow mode, and by utilizing heat that has higher a temperature than that of a light water reactor, it may have a much larger potential as a heat source.

REFERENCES

[1.4.1-1] Central Research Institute of Electric Power Industry (ed., S. Yoda), " Plan for Next Generation Energy, Denryoku Shinpo-sha(1998)."

[1.4.1-2] M. Kikuchi, This sub-committee, Document No 4-5.

1.4.2 Fusion as a Balanced Energy Source

Important criteria required for energy supply technology in the late 21st century and beyond will be the following:

1) Energy resources should have a large reserve and be equally distributed

Not only should there be ample reserves of energy, these reserves should be widely distributed, avoiding localized distribution, as is the case of oil localized in the Middle East.

2) Energy should be as clean as possible for the environment

Energy conversion and production should be environmentally friendly, essentially eliminating CO_2 emissions, all other pollutants, and radioactive wastes, even if technology exists to control them or dispose of them. If we pursue the complete zero concept, even natural energy will fail to satisfy this requirement completely. We should still strive to achieve the closest to ideal.

3) Energy costs should be within a reasonable range

Idealistic concepts that ignore economics will be of no value even if they are technically possible. No energy business can exist without the consideration of cost.

4) Reliability and sufficiency as a basic energy source

The basic energy supply system should sufficiently meet our energy needs. For example, natural energy alone cannot satisfy this requirement, and a more reliable system is needed to supply our basic electricity demands.

5) Energy system must have safety assurance and be felt at ease by the public

Technical safety does not always provide relief to the public. It is not possible for the public to feel at ease only by presenting the proof of safety. In considering safety and hazards, we may misjudge under certain occasions or overlook hazardous situations because of our logic in selecting related criteria and assumptions. It is important that the hazard potential to be absolutely minimized and that no grave danger might occur under the most unrealistic conditions. Further, it is important that this philosophy be understood and fully accepted by the public.

In the Section 1.3.6, Economic Efficiency, technical safety in operation is assumed and the above items 1) ~ 4) are discussed. However, because retaining public support during the actual operation is as important as the initial support received before implementation of the system, it is necessary to satisfy the high level of safety by the addition of item 5). We consider this issue further.

Conventional fossil fuel power cannot satisfy item 2) completely. Renewable energy and natural energy cannot be regarded as having equal distribution or as having sufficient resources, but they generally meet items 2) and 5). However, it is anticipated that satisfying items 3) and 4) at same time, even in the future, will be difficult. Light Water Reactors have limited resources of uranium from mining, and at least at present it will be difficult to satisfy 1). Possible candidates of the future energy supply must satisfy all criteria to a certain level and provide a sufficient amount of base load energy for at least several hundred years. Among technologies regarded as somewhat feasible are the fast breeder (FBR), the light water reactor using uranium from seawater (LWR-SW), and the fossil-fuel power plant with CO_2 sequestration (coal, conventional or unconventional natural gas). If fusion energy is developed, it will be in this category.

Among the above, the cost of electricity produced by fossil fuel power including CO_2 sequestration and disposal of the CO_2 in the ocean using present-day technology is anticipated to be 15 yen/kWh or more. In foreign countries, there may be other possible options of relatively low environmental impact for CO_2 sequestration such as pumping into an exhausted gas field, but there is no such place in Japan and only ocean disposal seems to be feasible. If the cost could be reduced by technical developments, environmental concerns about depositing a huge amount of CO_2, approaching 100 million tons annually in Japan, on the ocean floor will remain. Besides Japan, there are many other regions where no adequate site is available for CO_2 disposal.

Fission nuclear power including the FBR is expected to continue to supply energy over a long period at sufficiently low generation costs. However, it has the problem of disposal of high-level radioactive waste. After the Three Mile Island and Chernobyl accidents however, the siting of new plants has encountered considerable difficulty worldwide. In addition, concerning the nuclear fuel cycle, unavoidable transportation of nuclear materials such as uranium or plutonium is a concern from the viewpoint of nuclear proliferation. Recent accidents in Japan at fuel cycle facilities (not at power plants) have caused extensive anxiety and distrust in the Japanese public. For these reasons, a new energy system based on an innovative technology like fusion is needed as an energy source in the 21st century and beyond.

In the Section 1.4.1, concerns were expressed about whether fusion reactors could satisfy the above criteria. In the Fig. 1.4.2-1, the results are summarized by comparing the characteristics of a coal-fired plant with CO_2 sequestration, a light water reactor with seawater uranium, and a fusion reactor. Included in this comparison are:

- Resource/consumption ratio as an indicator of the remaining fuel resource
- Inverse of CO_2 emission unit as an indicator of the effect of CO_2 reduction
- Inverse of Cost of Electricity (COE) as an indicator of economy
- Inverse of Biological Hazard Potential as the indicator of hazard in operation
  (Evaluation addresses the total amount of movable radioactivity in the operational reactor.)
- Inverse of Biological Hazard Potential of reactor 20 years following cessation of operation as an indicator of the radiological risk of the waste
  (Assuming 30years operation, all wastes of the decommissioned facility for fusion and light water reactor, and for coal-fired plant, the ash from coal for 30 years)

These items are now discussed. Biological Hazard Potential is defined in Section 1.3.3.1. Because the three systems compared here will have no difficulty concerning the reliability of supply listed as 4), this indicator is not discussed. Indicators of the operational hazard and radiological risk of the waste are normalised by that of the light water reactor as a standard (=1.0), and other indicators are normalised with the present coal-fired plant without CO_2 sequestration as a standard. The center point is less than 0.0001. For fusion reactors, values for the demo reactor SSTR are mostly used, but for the economy, the target for an early commercial reactor is also shown. Table 1.4.2-1 summarises the values used in preparing this figure. Although the FBR is not compared here due to the lack of the Biological Hazard Potential value, the characteristics would be anticipated to differ little from the LWR-SW except for the COE, which could decrease as the technology matures. In the figure, the larger the area surrounded by the heavy line, the more of these 5 criteria are satisfied. The area for fusion reactor, even for the demo, is rather large. If the target for the commercial reactor could be achieved, the pentagon will be balanced, which would indicate its superior features as an energy source.

Briefly summarising this section, the fusion reactor is concluded to have the following features concerning the criteria 1) to 5) listed in the previous section as follows:

1) [Resource]

The required resources including mandatory structural materials, etc., are sufficient. Particularly for the fuel resource, even in comparison with uranium, mined and from seawater, lithium is now 200 times more plentiful, and is virtually limitless in seawater.

2) [Environmental Effect]

The generation of CO_2 in a fusion reactor is as low as a fission nuclear power, and a fusion reactor generates very little other waste that would be considered an impact to the environment.

3) [Cost]

It is expected that the cost of fusion electricity will be in the competitive range with other future energy sources.

4) [Stability]

Fusion power is adequate as a large-scale energy source (including uses for other than electricity).

5) [Safety and Acceptance]

The Biological Hazard Potential during fusion reactor operation will be 1/1000 of that of fission power. After cessation of operation, all decommissioned waste from a fusion reactor decreases its BHP to the level of ash from coal of a fossil fuel power plant in 20 years, and it continues to decrease after that. The fuel cycle of a fusion reactor can be closed and completed at the site, so transporting or handling nuclear materials such as uranium or plutonium is not involved. At the same time, safety features such as no criticality concerns, a lower heat density in structural material in the case of a loss of coolant accident as compared to a light water reactor are recognised.

Therefore, if the commercial fusion reactor is realised, the important criteria for future energy sources listed in the beginning of this section are expected to be met at considerably high levels. Fusion energy has the overwhelming advantages of being a rich resource and having a widespread distribution, while having no serious disadvantages. There are other attractive alternative energies but that lack total balance; for example, they have excellent cleanliness but have very expensive construction and electricity costs. The largest feature of a fusion reactor could be considered its good total balance. Although the cost of electricity in the demo reactor phase may be high, it is still less than twice the cost of present-day electricity; and it will be in the competitive range when fusion reactors are commercialised. In actual operation, systems having good balance will be preferred over the systems with that are unbalanced. In as much as fusion is the only technology within the range of the current technical capabilities that can satisfy all the criteria with good balance, it is expected to be one of the promising major energy sources of the later 21st century. Therefore, industrial countries like Japan must take the lead in realising this technology.

Table 1.4.2-1 Assumed values used in the overall assessment of the system (see Fig. 1.4.2-1)
(Values in rectangles are the standard for normalization)

	Present-day Coal-fired Without CO_2 sequestration	Coal-fired plant with CO_2 sequestration	LWR with sea water uranium	Fusion DEMO plant
CO_2 emission unit (Carbon g/kWh)	$\boxed{270}$	85	5.7 assumed to be same as LWR	6
Biological hazard potential of movable radioactivity in reactor (m^3)	Assumed to be sufficiently low	←	$\boxed{5.4\text{x}10^{17}}$	$3.5\text{x}10^{14}$
Inhalation BHP 20 years after operation (m^3)	Ca. 10^{16}	←	$\boxed{10^{20}}$	Ca.10^{16}
Reserve/product Ratio (year)	$\boxed{231}$	←	More than 10,000 years virtually unlimited*	More than 10,000 years virtually unlimited*
Normalized cost of electricity	$\boxed{1.0}$	1.8	Ca. 1.3 at the price of uranium 5 times of present-day value	Demo: ca. 1.7 commercial: 0.7-1.2

* Reserve/Product Ratios of uranium and lithium in seawater are estimated to be respectively 750 thousand and 15 million years, both are virtually regarded as infinite.

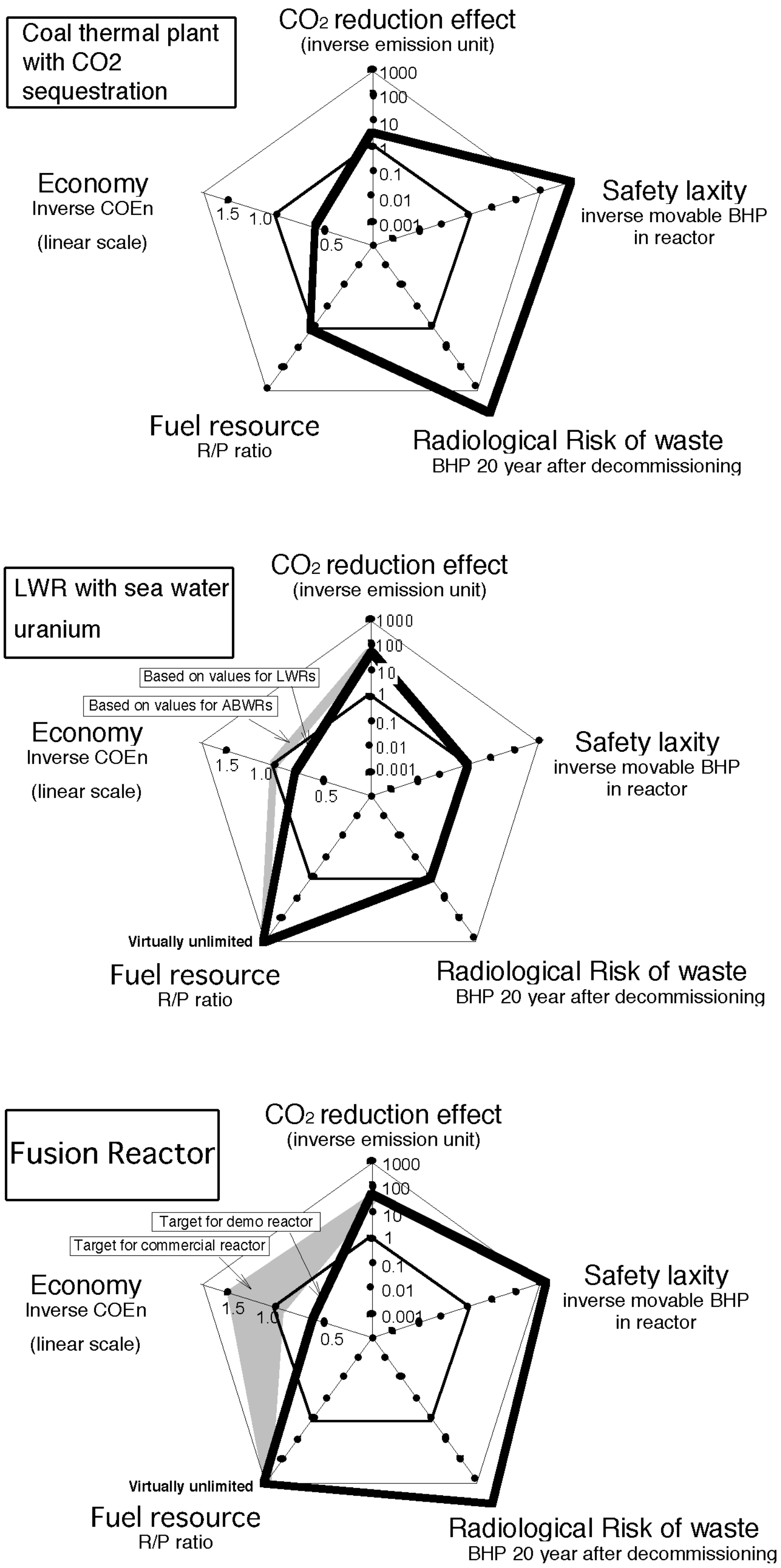


Fig. 1.4.2-1 Overall system assessment

Values are normalized to present coal-fired plant (without CO_2 sequestration) except that safety and radiation risks of waste are normalized to light water values. Note that economy has a linear scale.

Chapter 2 Development strategy for realization of fusion energy based on ITER project

2.1 Approach to practical utilization of fusion energy

2.1.1 Research and development necessary for realization of fusion energy

The objective of fusion energy development is to ensure a stable source of energy for all humankind. This lofty objective can not be attained until fusion power plants are supplied to the market. To make fusion energy practical, it is necessary to establish the technology of a fusion power plant and to make the fusion power plant economical enough to be competitive with other energy systems in the market. At this time, it is difficult to define the detailed specific objectives of a practical fusion reactor, as was described in the interim report of the Special Committee on ITER project. In this subcommittee, the power generation production cost, which is important in establishing the specific objectives of a commercial fusion reactor, has been estimated to be 7~15 yen/kWh (0.7-1.5 COEn) as the appropriate range to provide competitive power in the later half of the twenty-first century. Accounting for the cost of the site and the reactor decommissioning costs, both of which affect the cost of energy, a target of the plant construction cost is expected to be 300~500 k yen/kW. To achieve this target, the development of a comparatively high-power reactor is promising because there is scale merit in constructing a fusion power plant.

On the other hand, social demand on energy source arises from the fact that the way to produce and supply energy possibly has a strong social impact. It is necessary to keep in mind that the social demand depends not only on consumption and economic aspects, but also on other aspects, such as the protection of the environment, safety, and the reliability of energy supply. In addition, it is necessary to develop the fusion power plant as the energy system which can be used immediately when other energy source could not be used, and as the energy system in which its unique features, i.e., supply stability of a large-scale energy, safety, an independent fuel cycle, etc., are sufficiently utilized.

For realization of fusion energy complying with these social demands, a technological development strategy having a balance between fusion plasma technology and reactor technology is needed. In the demonstration phase of fission power reactors, for example, many types of reactors including the heavy water reactor, the graphite reactor, and so on were built and then tested. The selective trend toward the light water reactor has occurred in the competition phase. In the phase to develop a fusion demonstration reactor, the technological developments and the demands of the market are not clarified, and it is thought to be necessary that extensive research and development be advanced while the possibility of a flexible selection of the reactor-type remains open to make the development beyond the demonstration reactor phase robust.

(1)The route from research and development to practical use

From a long-term view, there are two phases necessary in the process to establish a new energy source in the market. One is the phase of “research and development,” which is performed until the technologies for energy production are established. The other is the phase of “practical use and utilization” where the developed technology blends itself into the market through usage and continual improvement. The technical feasibility in the “research and development” phase is to show fusion energy is in the position that it can function as an alternate energy source when other energy sources are not competitive or available. In other words, the power plant must verify that the fusion energy produced is actually stable and that power generation at a level of several hundreds of MW is feasible. The device needed in the “research and development” phase is one that has all equipment a commercial reactor would have and that can demonstrate the economic prospects showing sufficient attractiveness of a fusion a commercial reactor in the following phase of practical use and utilization. It means that the first fusion power plant that generates electricity is to be a “prototype” of a commercial fusion reactor. Establishing fusion power plant technology, which compares to that for the present fossil and fission energy, attains the goal. It is also the device that should be demonstrated irrespective of all approaches applied in previous phases and that makes the phase “research and development” complete. It is a necessary path that must be followed regardless or whether research and development for ITER is defined as “energy development” or as “energy science,” since the development of a fusion reactor aims at realization of fusion energy.

On the other hand, a central part in the goal that the demonstration reactor should aim at as a power plant has been made more clear by the R&D results of magnetic fusion and the continued design research of a fusion reactor during this decade.

The phase “practical use and utilization” following the phase “research and development,” will be promoted over the latter half of the twenty-first century. In the phase, the fusion power plant will be improved by the private initiative, and its social and commercial feasibility as one of energy source will be examined. For the development of fusion energy, continuous improvement of the technology will be sought, such as the pursuit of simplification, standardization, efficiency upgrading, and performance enhancement, to form a source superior to other energy sources in a competitive force in the market and to make it commercially attractive. It will be similar to the present case for other energy sources including fission reactors. In this phase, multiple design options will coexist with the various needs and social demands for energy production, and upgrades to comply with the economic aspects and social acceptability will be done in a process of product development and improvement. The period needed for this phase will be decided by a factor apart from technology, for example, energy demand and supply, safety, public opinion for environmental problem, and other choices. Therefore it is appropriate that development programs to be included in the phase of “practical use and utilization” are not decided in detail at present. It is regarded that this issue ought to be addressed in the future.

As a technical and social request in considering long-term development programs, the guideline “that a fusion reactor is to be able to be realized with the smallest cost, in the shortest period, and with the minimum development risk as a whole” should be sufficiently taken into account. However, as described above, in long-term research and development, the content of technology development on the phase of “research and development” of a demonstration (the DEMO reactor) reactor is different from that directed to a commercial fusion power reactor; although a goal in the phase of “research and development” is the demonstration of the technical establishment, the phase of “practical use and utilization” makes the extension of the participation to the market with improved economy and reliability a respectable goal. Moreover, the goal extends broadly due to the possibility of various ways to utilize this energy, and this provides incentive to proceed with advanced R&D. In the above request, therefore, it is regarded as important that the research and development program for the demonstration reactor be drafted rationally in light of present knowledge, while the ideal form of the practical usage phase is recognized.

In addition, various designs are possible in the case of the fusion reactor because fusion output is dependent on many plasma parameters. Therefore, it is suggested that the optimum device concept not be stressed, but instead that the goal area of the fusion plasma parameters be addressed. Furthermore, for such a fusion reactor the directivity of the goal can also change due to discoveries in future plasma physics and the development of reactor technology including the improvement of precision on performance predictions. Therefore, periodic review of the results and adjustment of the goal are indispensable.

(2) Necessary research and development on the demonstration reactor

The requirements of a power plant such as the demonstration reactor are:

1) verification of technology that can generate power by practical fusion energy, and
2) the incorporation of the economic aspects that lead to a practical reactor.

It is required, therefore, that the device be compact, that the operation of the steady-state fusion plasma have as high an energy multiplication factor as possible, and that the self-supply of tritium fuel during power generation is possible. Also demanded is that the fundamental problems on the materials to be used be solved and then that a good prospect of their use for practical reactor applications be obtained.

Some specific technological problems exist when such a demonstration reactor is nearing the final goal of the development phase. In fusion plasma research and development, a steady-state fusion burning plasma with a Q value over about 30, which is needed to supply net energy, has become an important subject. Also, the high ratio of the bootstrap current to the total plasma current, the control of the driving current, and the compatibility between fusion plasma performance and steady-state operation of the divertor are essential issues to attain steady-state burning plasma. In the field of fusion reactor technology, the technology for the main components are basic, such as the superconducting coils, which compose a reactor in themselves, the remote maintenance techniques, the tritium fuel cycle including initial loading, the breeding of tritium, the treatment of waste and disposal, the development of low-activation materials, the heating and current drive equipment, and the measurements and controls. The development of materials that withstand the high fluence of neutron irradiation and the development of blanket technology for power generation dominate the demonstration reactor phase. Advanced technical demonstrations, experiments, and tests are imperative.

Since a demonstration reactor cannot be constructed without conducting experiments and research and development on these subjects, it becomes an arguing point whether one can reach the goal within the goal of reasonable risk, with rational funds allocations, and within a reasonable period of development by taking the R&D approaches.

More concisely, the following problems are arguing points:

1) Can we undertake construction of the demonstration reactor only based on the tests for various technologies and plasma characteristics necessary for it when it must operate with multiple devices that share different goals?

2) Is it necessary that prior to construction of the demonstration reactor, we demonstrate the engineering system and total plasma performance by an integration device that includes many elemental technologies that indeed constitute the demonstration reactor?

3) What interim step should we proceed to, if it is required?

On the other hand, it becomes an object of discussion as to whether the present technological level is at the level with which we can undertake the construction of the integration device when it is required.

(A) The multi-path approach (modular approach)

In multi-path approaches, the subject of research and development needs for the demonstration reactor are divided, and they are examined without constructing a large integration device like ITER, but by using multiple devices. Then, a determination is made of whether one can construct the demonstration reactor directly from those results.

In this case, a division of the development subjects on the fusion plasma becomes an essential arguing point. As a representative example, there is a modular approach that was once considered in the US. It was considered that the research subjects, which are the physics research on the behavior of the plasma under the self-ignition condition and on the operation of long-time steady-state plasma, are needed before construction of the experimental reactor. Then, from this viewpoint, the multiple test devices were recommended with lower total cost and at smaller size than that of the experimental reactor for the following purposes:

- Plasma burning device (using the normal conduction copper coil) during a short time pulse (about 10 seconds) for the purpose of achievement of the self-ignition condition by a D-T reaction.
- The steady-state plasma device (several hundreds of seconds in pulse length when the superconducting coil is used) by a D-D reaction.
- The neutron irradiation facility for development of the materials.

The premise is that the research of plasma control and plasma physics is advanced in parallel by using newly constructed devices of a) through c), and then one proceeds to construct the DEMO reactor (the demonstration reactor for the purpose of the demonstration power generation).

It was argued in the international community, including the US community, whether such approach could be a substitute for the ITER project, or not [2.1.1-1]. As a result of the argument, it was recognized that the experiments in device a) and b) were far short of the pulse duration needed and far below the fusion output needed. The properties of integrated plasma, for example the physical performance of the fusion burning plasma in steady-state operation and the interrelation between plasma heating by fusion burning, confinement barrier, pressure and current distribution control, and compatibility with the divertor, cannot be tested without an integration device. In addition, it was recognized that the subjects of integration of the plasma and engineering cannot be examined, for example the coupled electromagnetic and thermal problem for the structure and the plasma under large-scale conditions in the fusion reactor cannot be resolved. It was also noticed that the problem concerning the equipment characteristics under exposure to the complex environment (neutron radiation, high temperatures, corrosion, and fatigue) could not be resolved. Further, the development of technology to test the blanket module for the demonstration reactor is not possible. Thus, the approach considered could not be a substitute for the experimental reactor ITER project, since the long-term view was lacking. After all, the integration device similar to ITER before the demonstration reactor and the associated large extra funds would be required. It was concluded again that research and development on fusion would be delayed more than 10 years by this approach.

The modular approach would contribute to understanding the behavior of the plasma under the self-ignition condition and advance the control technology of steady-state plasma. However, it is considered in an extrapolation for the demonstration reactor and in view of the system integration, risks on the development period and the cost greatly increase, since the modular approach cannot examine the steady-state performance of the burning plasma. This subcommittee supports the present conclusion, and recognizes that the integration device is needed at the intermediate-stage that precedes the demonstration reactor, so that engineering developments will be advanced.

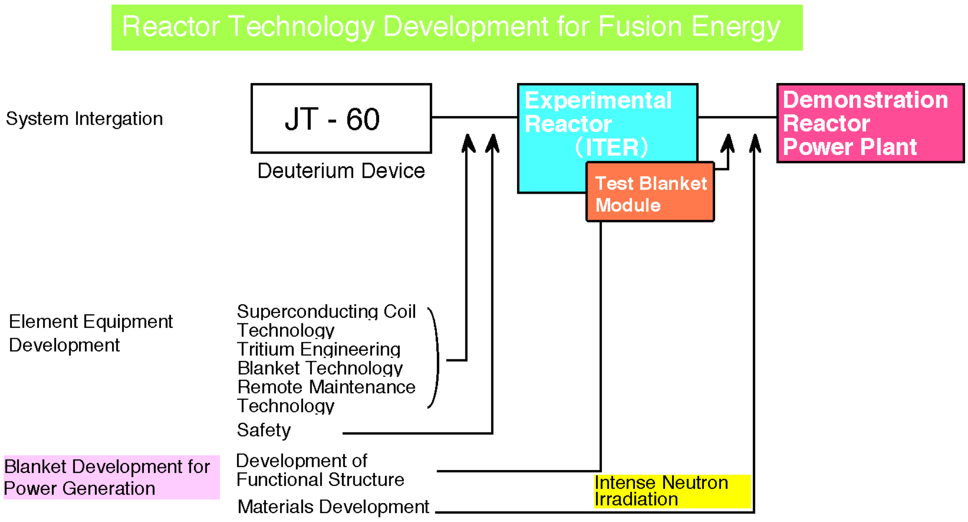
(B)The role of the integration device and the necessity of the experimental reactor

Electric power generation, from which we receive many benefits daily, technically consists of the processes of design, production, operation, and management. Usually, the design is proper, and the devices function approximately as designed. However, it must always be remembered that individual devices do not always provide 100% performance even when the technology is put to practical use. This is because in the design stage it is not always possible to predict what may happen. It is also impossible to accurately predict time-related developments. From this point of view, to adopt the phased development of technology is a wisdom for complementing human imperfection. It is a very reasonable approach for people to carry out the integration using core devices step by step and to gain relevant experience.

By building the integration device, we can also verify the technical formation as a system as well as individual elements including elemental technology and plasma performance. In addition, such integration should be conducted when practical, but as early as possible, in case of long-term research and development. This would reduce the uncertainty related to establishing the objectives of the next phase. The degree of maturity of the technology is advanced by the precious data on the operational limits and the trouble situations, obtained through experiment, operation, maintenance and modification, and by the valuable experience of solving problems. The integration also makes it possible to evaluate the rationalization of efficiency and cost at an early stage. Namely, by testing and operating an integrated system, the total performance is verified as an integration of the elemental technologies. In addition, the demand on each element of equipment and the system control technology necessary for enhancement of reliability and the cost reduction becomes clear. The improvements based on this process will be utilized for the integration at the next phase. This repetition is indispensable to increase technical maturity. Therefore, it is necessary that the large device be constructed and operated under an environment as close as possible to that in the demonstration reactor. This will provide integration of the elemental equipment for the future fusion reactor. However, if immature elemental technologies with poor chances for success are integrated without care, the entire system will not exhibit its function, and the purpose of the integration will not be accomplished. Accordingly, the phase in which individual elemental technology should be integrated should be decided as a very important matter of balance in long-term research and development.

It is especially difficult to predict the behavior of steady state burning plasma, in which self-organization dominates, by merely advancing elemental R&D and theoretical analysis, in view of the long-term development. However, it is the centerpiece and the biggest issue in fusion development to understanding the behavior of steady state burning plasma experimentally, since the plasma performance strongly affects design of the future fusion reactor. The experimental reactor should be constructed as the device in which these problems are discerned and solved prior to the construction of the demonstration reactor.

There are two kinds of engineering technology, the engineering technology inevitably integrated into the experimental reactor with such a role, and the engineering technology whose integration is not necessary until the demonstration reactor at the next phase. All reactor technologies except the breeding blanket and power generation are inevitably needed by the experimental reactor. The only thing we should consider carefully is whether the power generation blanket technology including the selection of the blanket material should be integrated in the experimental reactor or not. The phased integration of the technology is shown in Fig.2.1.1-1. Taking the high importance of the burning plasma and technical maturity of the materials development into account as described later, integrating the power generation blanket technology into the demonstration reactor is an appropriate approach.



- Technical Feasibility for a Fusion Reactor will be demonstrated by Demonstration Reactor (Completion of Development Phase)
- The Experimental Reactor
  1) Synthetically demonstrate the Reactor Technology Element Equipments except for the Power Generation Blanket
  2) Be a Test Bed for Development of Power Generation Blanket

Fig. 2.1.1-1 Integration of technologies to achieve fusion power reactor

(3) Phased development and core device

Plasma physics and reactor technologies are combined at an integration device when research and development are advanced. A “phase” is defined as corresponding to the period in which an integration device plays a central role. The necessary preparations are made for the next integration device, and then the evaluation of all results and the decisions for the next plan are made following the completion of the preparations. We call it “phased development,” which is development promoted in a stepwise manner. In this case, it is important for continuing investment in the R&D that the research progress is large enough to meet the research program and to satisfy social requests. In addition, it is also important to always establish a clear mid-term goal to maintain the vitality of the research program. Therefore, it is also necessary to establish sequential milestones, which correspond to goals of the core devices, in long-term development programs, where roles of the core devices and their linkage to the next phase have to be clarified.

In the fundamental approach of phased development, (1) we integrate the knowledge of technology and plasma obtained in the previous phase, we construct and operate the core device, and we then achieve the mission. Simultaneously, in parallel, (2) we attempt to upgrade fusion plasma technology and reactor technology, including the development of materials, to advance technological development, which is necessary for planning and selecting the system of the next core device.

The factor of judgment now being considered regarding construction of the experimental reactor results from the fact that until now core devices provided the break-even plasma achievements and the generation of DT fusion energy in JT-60 of Japan, JET of Europe, and TFTR of the US. Addressing the allocation of funds, although the majority of funds are put into construction and operation of the core device, it is necessary that appreciable funds be used for parallel developments other than for the core device to minimize the cost, period, and risk of the entire program. On the other hand, the construction costs of the core device for each phase will be used as a reference for the construction costs of a commercial reactor. It is therefore required that an allocation restriction be established, based on the upper limit of the investment amount acceptable for Electric Power Companies. Then the technologies for the cost reduction have to be developed so that the mission of each phase may be attained within that restriction.

(4) The experimental reactor as a core device prior to the demonstration reactor and the Third Phase Basic Program of Fusion Research and Development

Prior to the demonstration reactor, it is necessary that the most certain and leading-edge fusion plasma and engineering technologies available be concentrated, that the experimental reactor be constructed as a core device aimed at an achievable goal, and that the technical basis be established to proceed to the next reactor, the demonstration reactor. In addition, the core device prior to the demonstration reactor, the DEMO reactor, should be one device, if possible, to minimize the costs of the development phase of the entire program. This is the basic idea (single step to the DEMO reactor) behind this experimental reactor phase.

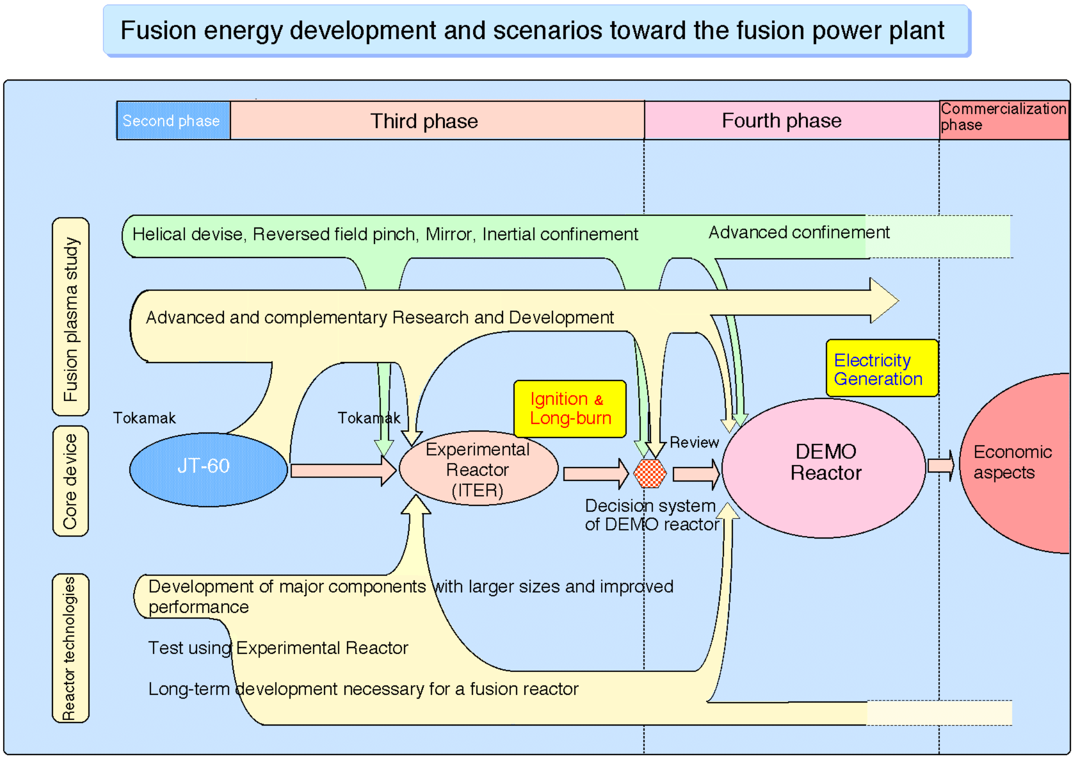


Fig. 2.1.1-2 Third Phase Basic Program of Fusion Research and Development

In viewing fusion power generation from the present research levels, achievement of the self-ignition condition and the long time operation of steady-state fusion plasma are indispensable as primary milestones. These achievements will clarify the prospects of fusion power generation, and allow rational judgement of suitable further monetary allocations to the demonstration reactor phase. Although we aim at a new field that includes the self-ignition condition and long burn plasma in the experimental reactor, we can, with sufficient confidence, foresee the success of our efforts based on the results obtained from the large tokamak research in JT-60, JET, and TFTR.

For the ITER now under consideration, although its design has been modified to decrease its energy multiplication factor, this change has made the single step to reach the demonstration reactor more suitable. Namely, the basic concept for the experimental reactor remains unchanged, that it be a single step from ITER to the DEMO reactor system. Furthermore, since the new ITER can retain the tolerance for density limit and the energy multiplication factor in a steady-state operation, as well, it is judged that the ratio of performance to cost will be improved.

It has been mentioned here that the integration device like ITER prior to the construction of a demonstration reactor is necessary. Namely, the phase in which the experimental reactor (ITER) plays a central role is necessary. In Japan, this is identified in “Third Phase Basic Program of Fusion Research and Development.” (Fig. 2.1.1-2).

Table 2.1.1-1 Fusion energy development strategies for Europe, Japan, and the US

	Phased Development (Japan) (Single Step to DEMO)	Modular Development (USA)	Phased Development (EU) (Single Step to DEMO)
Phase of Development	Status - Experimental Reactor - Prototype Reactor - Commercial Reactor	Status -D-T Burning Device and D-D Device in parallel Development -(Integrated Device) -DEMO	JET Experimental Reactor DEMO
Risk estimated from the current technology (Reliability etc.)	Experimental Reactor: low Prototype Reactor : low Commercial Reactor : low (There is the necessity of improving the economic efficiency more.)	D-T Burning Device and D-D Device: low DEMO : high (The extrapolation possibility to DEMO is the indistinctness. The leap for system integration of the individual equipment is large.) DEMO : low(in the case of integrated device between them)	JET -Experimental Reactor: low Experimental Reactor - DEMO: low
Period of Development	Experimental Reactor: 10 years for construction Prototype Reactor : 8-10 years for construction	D-T Burning Device : 8 years for construction D-D Device : 8 years for construction Integrated Device :About 10 years, because the ITER class is assumed) DEMO : (?)	Experimental Reactor: 10 years for construction DEMO : 8-10 years for construction
Cost of Development	Experimental Reactor: About 500 billion yen Prototype Reactor : almost the same as Experimental Reactor Commercial Reactor : less than Prototype Reactor	D-T Burning Device and D-D Device: about 1/4 of ITER-FDR Integrated Device: more than half of ITER-FDR DEMO : (?) Research and development expenses of each elemental technology are necessary.	Experimental Reactor: predictable DEMO : predictable
Extrapolation of Results at Experimental Reactor	Applicable to the demonstration reactor.	To DEMO from D-T Burning Device and D-D Device: not applicable	Applicable for DEMO
Other	For prototype reactor • The positioning as final test device of the research and development stage • The attempt to improve efficiency and make cost reductions as long as they are possible. • The neutron radiation equipment for materials development for prototype reactor is necessary.	• Low risk due to separation of the missions and flexibility of the development subject solution. • Degree of freedom of the experiment. • Test of high heat load equipment revealed in the D-T environment and the burning test are not possible. • The neutron irradiation equipment is necessary for the materials development for DEMO.	• The neutron irradiation equipment is necessary for the materials development for DEMO.

While the experimental reactor is to be the core device in the Third Phase Basic Program, the research and development necessary for a demonstration reactor (the DEMO reactor) is to be advanced as well. That is to say, it is necessary to first confirm the behavior of the steady-state burning plasma in which the autonomia is high, and the ITER has been designed as the core device that accomplishes this mission.

In addition to the ITER development, R&D on tokamak devices for advanced and complementary research and R&D on devices other than the tokamak such as the helical type system should be pursued. In these devices, new concepts will be tested prior to ITER, and experiments that examine regions that cannot be explored with ITER will be performed aiming at higher plasma performance. On reactor technology, the major technologies for the DEMO reactor are adopted and integrated in ITER. The materials and blanket development for the DEMO reactor will be promoted in parallel with ITER, as well as by using ITER. Therefore, not only ITER but also the intense neutron source for materials testing and the facilities for the development of blankets and the fuel cycle, etc., are considered necessary. By using these facilities, it is possible that the development of the blanket and materials, etc., for the DEMO reactor element development, with its many options, will be carried out in domestic and overseas programs in parallel. And it is also possible that by applying the test results obtained to ITER and the DEMO reactor simultaneously, technological development for integration can be advanced with continuing efficiently.

Table 2.1.1-1 lists and compares the fusion power development strategies of Europe, the US, and Japan. Fundamentally, Europe has chosen a phased development program (single step to the DEMO reactor), which is similar to the plan decided by Japan. In both, the DEMO reactor, which is assumed to follow the experimental reactor, is used for the purpose of demonstration of economical power generation. The purposes of the European the DEMO reactor are the demonstration of the first wall and the low-activation material used in the blanket, of the tritium fuel cycle including the production of tritium in the breeding blanket, of safety, of the environmental protection system, and of full-scale remote maintenance, all in addition to power generation.

The US has changed its viewpoint regarding fusion power from “energy development” to “energy science” in the last few years and is now stressing the importance of the concept of the advanced tokamak with respect to fusion development as its domestic program. The long-term strategy seems to have been entrusted to the future, while the US is keeping the possibility to re-participate in ITER. On the other hand, the necessity of the fusion burning experimental reactor is recognized again in the USA, and it has been concluded in the Department of Energy Director General advisory committee (SEAB: Secretary of Energy Advisory Board) that re-participation in should be examined if the ITER construction starts.

2.1.2 The master development plan

(1) The numerous features of fusion

Since plasma is governed by various parameters, the optimum regions of plasma performance and control methods based on present knowledge should receive sufficient review with the results obtained at each step of phased research and development. Therefore, the goal for the next step should not be regarded as being fixed. In the case of fusion development, progress in the understanding of plasma can make the direction more certain. Therefore, at the moment of phase completion, optimization of the goal for the next phase should be attempted with the deep and comprehensive evaluation. A large step in the long road is inevitably needed because the confinement study of fusion plasma requires a large device.

On the other hand, it should be mentioned that an outstanding feature of a fusion reactor as a nuclear energy conversion system is the ability to individually develop tow technologies, the one is for producing and controlling the fusion reaction, and the other for energy conversion and energy extraction. In developing a fission reactor, since the type of reactor is decided for a specific combination fuel-moderator-coolant, various types of reactor in demonstration and prototype reactor phase have to be developed to meet various demands and possibilities. For a fusion reactor, however, it is possible to simultaneously and progressively develop various types of blanket system, which includes the function of the energy extraction system, and various types of fuel circulation systems, apart from the development of fusion plasma (Fig. 2.1.2-1). These features are especially remarkable for the development of elemental engineering technology and in materials evaluation. Fusion as an energy source having such merits has an ability to immediately respond to various needs of society in the future with the extensively developed technology.

In the present state of reactor technology, though the technology necessary for the experimental reactor construction has almost been established, the research and development of energy extraction is in the very initial stage. The development for combining the requirements for the power generation blanket and a fuel cycle component starts and becomes the main issue in the experimental reactor phase. Using ITER as a test stand with a fusion reactor environment will advance this development. In addition, the development and testing of some energy extraction systems will be performed in the demonstration reactor phase. Except for the economic aspects, the technological development for fusion power generation will be almost complete by the end of the demonstration reactor phase. It means that the demonstration reactor will technically attain the performance required as a power generation system and also that fusion energy has become an alternative energy source which can reliably supply energy. In this phase, the "insurance" role as power source that is required from fusion will be fulfilled.

In the phase of a future practical fusion reactor after the demonstration reactor the DEMO reactor, the evolution to the market will be attempted with improving power generation technology, while competing commercially with other energy sources. Then, various developments for producing energy corresponding to the demands of industry and the public, such as a heat source, in addition to being a source of electricity, will also be carried out. Although the system will be restrained by the selection and standardization of the market, a fusion power plant's ability to respond to a change in the demand of its power output by exchanging its blanket will be an excellent feature of the fusion energy.

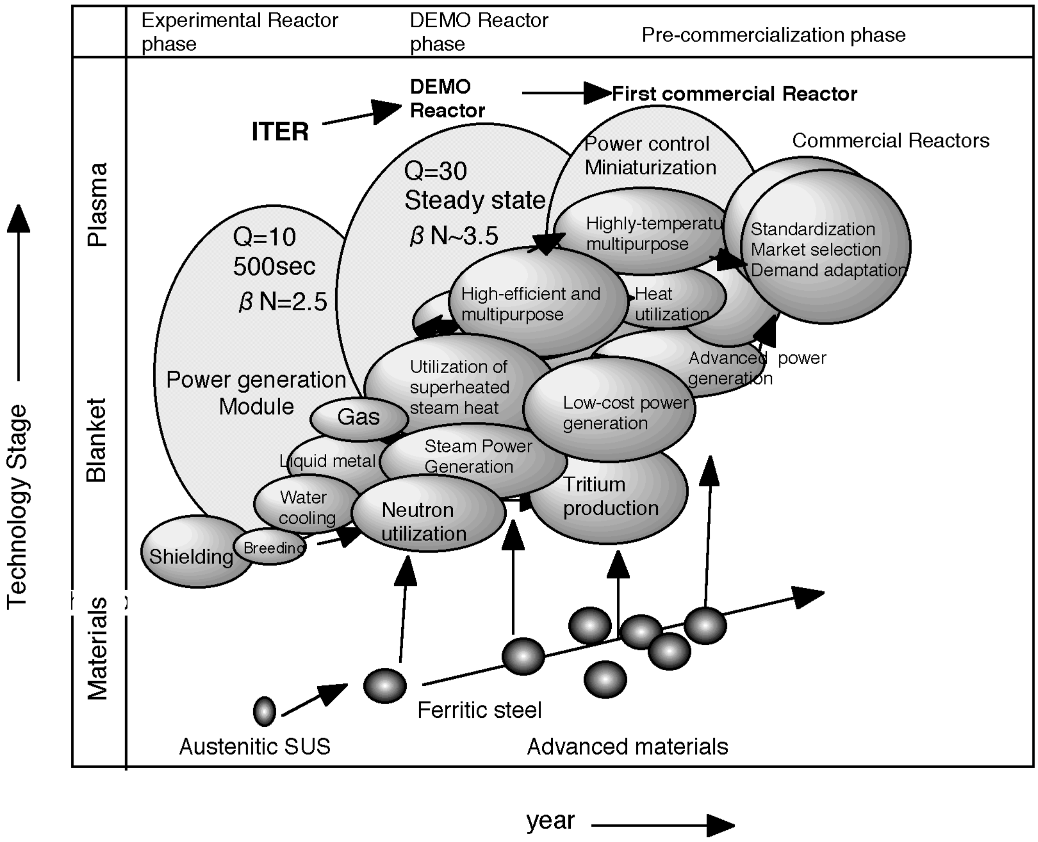


Fig. 2.1.2-1 Steps to achieve practical utilization of fusion energy from the viewpoint of reactor technology

(2) The master plan

As a premise for this section, it is assumed that the combustion of coal resources will be restricted as a countermeasure to the global environmental impacts described in Chapter 1, therefore the necessity of other forms of energy will increase, especially after 2050. Consequently the need for fusion energy will enter a practical utilization phase at that time. That is to say, about 2040-50 is assumed to be the time when the industrial world can judge a construction of a commercial fusion reactor with the technical and economical prospects.

To advance fusion to that situation, the following are necessary:
1) To undertake the construction of the experimental reactor ITER after the end of EDA, and to achieve the basic performance (self-ignition (Q is over about 20), long time 1,000 second burning plasma, and steady-state operation with Q>5) in the basic performance phase of ITER, the period of about 10 years after the start of operation.
2) To select the type of the demonstration reactor system considering the progress of other confinement systems, and then to advance to the demonstration reactor phase, namely, to proceed to the engineering design, construction, and operational phases of the demonstration reactor. This is feasible because the advance from the experimental reactor phase to the demonstration reactor phase is possible as a result of achievements of the basic performance in ITER.

After progressing to the demonstration reactor phase, the experimental reactor continues experiments aimed at performance expansion, that is the demonstration of the high-power density and high availability operation required for the demonstration and commercial reactors concerning the economic performance, will be attempted. At the same time, it is appropriate to advance a blanket test for the demonstration reactor. By doing that, continuity of the development of fusion plasma, operational technology, and reactor equipment are ensured, and smooth progress of the program would be expected. The schedule for the development described above is shown in Fig. 2.1.2-2.

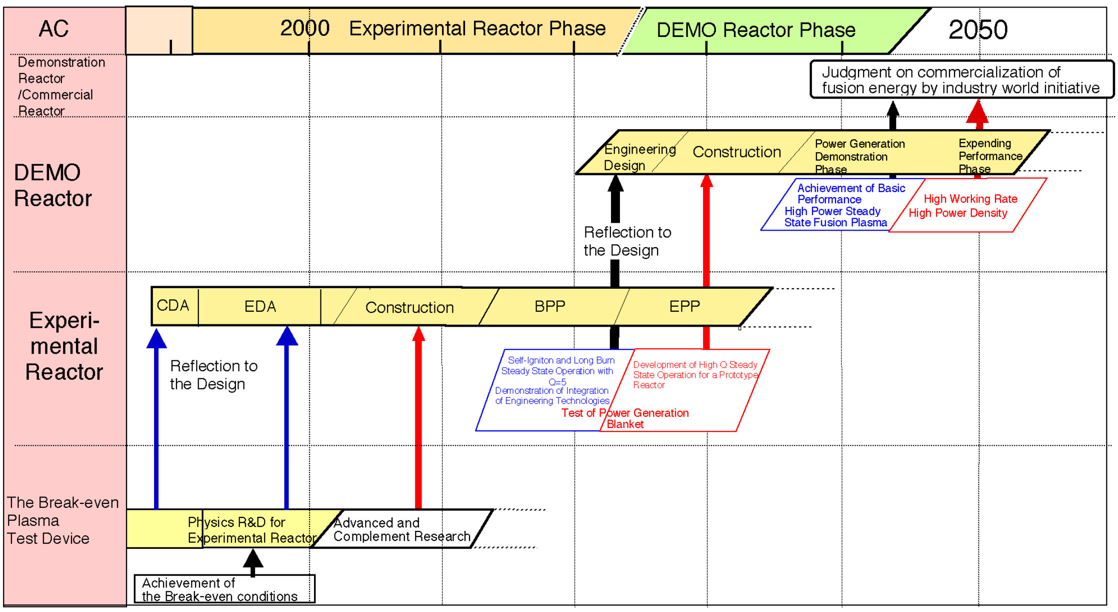


fig2.1.2-2 : Simplified schedule of tokamak fusion reactor development program

Table 2.1.2-1 shows the performance characteristics of representative tokamak fusion reactors. To make the prototype reactor more compact than the DEMO reactor, as required from its economic aspects, the following three developments are important, 1) the enhancement of plasma performance, 2) the development of high magnetic field coils, and 3) the development of the high-neutron radiation-resistant materials. The increase in fusion power density can allow a compact fusion reactor if 1) and 2) increases the plasma pressure. However, it is also necessary to simultaneously attain the development of the materials of 3) because the increase in the power density inevitably will increase the neutron radiation to the first wall. By anticipating the likely hood of strengthening the magnetic field and improving plasma performance, a plasma pressure more than double that of ITER is expected. In combination with the advanced ferritic steel presently considered a candidate for the DEMO reactor, it seems possible to satisfy the economic aspects addressed in Section 1.3.6. Moreover, please refer to Section 3.1.3 for the prospects of future plasma performance improvement, Section 3.2.3 for a description of the development of more powerful magnetic coils, and Section 3.7.4 for a discussion on the improvement of the economic efficiency.

Table 2.1.2-1 Performance characteristics of representative tokamak-type fusion reactors

	Present state	ITER	DEMO reactor	Prototype Reactor
Major Radius of Plasma	2.5-3.4 m	~6.2 m	7 m	5.4 m/6 m
Plasma Current	2-7 MA	15 MA	12 MA	12 MA/12 MA
Plasma Volume	30-100 m^3	~700 m^3	700 m^3	500 m^3/600 m^3
Operation System	inductive/ non-inductive	inductive/ non-inductive	non-inductive	non-inductive
Fusion Output	~15 MW	~500 MW	3 GW	3 GW/3.5 GW
Electric Power Output	-	-	1.08 GW	1.15 GWe/1.6 GWe
Energy Multiplication Factor	~1	more over 20	50	31/59

There is not enough time to reflect the experimental results obtained at each phase into the program of the next phase, as seen in a very tight schedule of Fig. 2.1.2-2. Therefore, it is necessary to realize the experimental reactor program as early as possible so that a commercial fusion reactor may enter the market early in the second half of the twenty-first century to cope with expected energy needs and environmental problems. Furthermore, the aspect of technological developments will have difficulty meeting the present predictions for the developments following the demonstration of power generation in the DEMO reactor, because the developments will be done under a private initiative and affected by the international and domestic competitive principle strongly after the extraction of the fusion energy has been proved. Therefore, although the objective concept of a commercial reactor should always be examined under the present situation especially in view of the economic aspects, it is considered suitable that the details of the approach and plan should not be fixed based on the present knowledge, but be fixed after the phase is advanced.
In the following sections, each phase is described in detail.

2.1.3 The phase of development for the experimental reactor

With the Third Phase Basic Program of Fusion Research and Development (Third Phase Basic Program) drawn up by the Atomic Energy Commission, the purposes of the experimental reactor are to achieve a self-ignition condition (Q is over about 20) by deuterium (D) and tritium (T) burning plasma, to realize the burning plasma for hours (about 1,000 seconds), and to form the basis of the reactor technology necessary for development of the demonstration reactor. In addition, it is required in this phase that the experimental reactor should be constructed as a core device, and that one understands the behavior of fusion burning plasma in which the alpha particles generated by the DT fusion reaction are the main heat source. As a specific subject, the energy multiplication factor Q over about 20 should be realized in the inductive operation, and the burning must be done for long time (over about 1,000 seconds), which seems necessary for having a good prospect of achieving a steady-state fusion plasma in the demonstration reactor. And, Q~5 should be realized in non-inductive operation, which is supposed to be the operation in the tokamak type fusion demonstration reactor. In addition, significant fusion energy generation (for example: 0. 3 MWa/m^2 corresponding to several hundreds of MW×10

years×0.05) should be verified within the limits of possible tritium supply, and the effect of the 14-MeV neutron radiation in the low fluence region should be evaluated.
As a fundamental issue for the fusion burning plasma of the experimental reactor ITER, the following items are mentioned.

a) Burning control in highly self-heated plasma where the heating power coming from 3.5-MeV alpha particles is 67~80%
b) Evaluation of the effect of macroscopic MHD modes, i.e., the toroidal Alfven eigen-mode etc., on the alpha particles, and their control
c) Understanding and control of thermal and particle transport properties in the reactor plasma where a value of L/ρ_i is significantly large compared with JT-60
d) Understanding and control of helium ash exhaust characteristics, and its compatibility with the high radiative divertor
e) Understanding behavior of resistive MHD modes, i.e., neo-classical tearing mode etc., in burning plasma and their control
f) Understanding and control of behavior of burning plasma with high bootstrap current fraction, where the bootstrap current profile, the plasma pressure profile and the self-heating profile strongly couple each other

And, if the confinement type of the DEMO reactor is decided to be tokamak, it seems to be appropriate that control of highly self-heated plasma with $Q > 5$ and high bootstrap current fraction and the steady-state operation of burning plasma with β_N = 3.5-4.0 should be established as physics R&D for the DEMO reactor in the extended performance phase of ITER. Furthermore, the technical basis of steady-state fusion plasma of the DEMO reactor with high Q (over about 30) must be established.

As described in “The Promotion of Fusion Research and Development” prepared by the Fusion Council (May 18, 1992), it is important that the following R&D be advanced with tokamaks as R&D of fusion plasma technology in the experimental reactor phase, in parallel with the experimental reactor.

On the issues of the fusion plasma technology necessary for constructing and operating the experimental reactor, the research and development should be advanced by utilizing existing facilities. The main issues are to realize a cold divertor plasma, to establish control technology of the disruption and an H-mode confinement control method, to optimize the condition (the operation scenario) of production and sustainment of the plasma, and to understand alpha particle behavior, and so on. In parallel with these, complimentary and advanced research and development mentioned above is needed to advance overall reactor technologies. Main issues are the reduction of circulation electric power in the plant by realizing high-beta plasma with a high bootstrap current fraction and the suppression of thermal flux to the divertor plate by the combined use of the remote radiative cooling and the separatrix sweep technique.

The problems of fusion plasma technology that are necessary for the construction of the experimental reactor are almost resolved at present. Now studies on the issues related to the operation of the experimental reactor and the advanced and complimentary research and development become important. It is also vital to advance the research on the future issues described in Section 3.1 of this report.

The research and development of the reactor equipment in the experimental reactor phase gives the technical basis for realization of a fusion power reactor, and open up the integrated reactor system technology and the advanced material technology. In development research of the main reactor equipment for realizing ITER, by utilizing existing test facilities and analysis methods until now through the ITER Engineering Design Activity (EDA) as much as possible, individual technological problems that concerns superconductor technology, high heat flux and heat removal technology, heating technology, maintenance and preservation technology have been steadily overcome. The development of the equipment was advanced remarkably; their performance necessary for ITER has been demonstrated.

In research and development addressing the DEMO reactor, the development of the blanket necessary for electric power generation is the most important from a reactor technology point of view. To develop the blanket with both functions of tritium production and the power generation (heat extraction), improvement and development of materials and the development of the blanket system as a functional structure must be promoted in parallel (Fig. 2.1.1-1). Low-activation structural materials that withstand the 14-MeV-neutron radiation of 100~200 dpa has been the goal for the improvement of ferritic steels for more than the past 10 years. The irradiation strength of 40 dpa has been presently confirmed by radiation test in a fission reactor carried out under conditions simulating fusion. Continuously, some material samples are being tested under higher neutron irradiation conditions. Finally, the heavy irradiation test by the 14-MeV-neutron source, whose neutron spectrum is similar to fusion spectrum, will be required. Following the research and development of the power generation blanket performed in parallel with the construction and operation of ITER, it is necessary that a test module of the blanket as a functional structure be installed in ITER and functional tests be carried out the under the neutron radiation environment. These tests can ensure the function of the power generation blanket in the DEMO reactor. A neutron irradiation facility, like ITER, for large volume specimens is indispensable to develop the blanket for the DEMO reactor.

Furthermore, high magnetic field (high temperature) superconducting magnet technology necessary for a high-beta plasma in the demonstration reactor phase and the blanket structural materials capable of withstanding the high neutron fluence of about 200 dpa, which will reduce the blanket replacement frequency and thus contribute toward practical utilization of a fusion reactor, should be developed. Conclusively, it is important to contribute to the design study and system selection of the DEMO reactor. Safety research on the environmental radiation, safety technology, and safety evaluation research are to be conducted to increase the safety of the fusion reactor. It is requested that the design study of the DEMO reactor and the commercial reactors be proceeded based on the latest knowledge of fusion plasma and reactor technology, and that the guideline of fusion development be given.

The parameters for the DEMO reactor and the experimental reactor must be appropriately chosen so that the parameters for the commercial reactor, obtained as a result of consideration, be achievable by a reasonable step from those for the DEMO reactor and the experimental reactor, when development is advanced toward the demonstration reactor. It is appropriate from the viewpoint of cost and technical subjects that a commercial reactor be a tokamak type reactor, in which most development advances have been made to date, and that the development programs on the basis of a steady-state operation system be implemented, rather than a pulse operation system. In the tokamak reactor for steady-state operation, the reduction of circulation power including electric power used for non-inductive current drive is needed to keep the energy efficiency of the entire plant high. The energy efficiency of the entire plant is a function of the energy multiplication factor Q of the plasma and thermal conversion efficiency of the generator. The design point for the demonstration reactor for each country is that Q=20~50, and the energy efficiency ranges 30-40%. Therefore, operation parameters of the experimental reactor ITER are suitable as parameters for the device from which we will proceed to the DEMO reactor by only one step.

2.1.4 The phase of the demonstration reactor

The demonstration reactor the DEMO reactor is constructed as an integration device, and its power generation of net electric power 500–1,000 MW is to be demonstrated. For that, steady-state fusion plasma with the high-energy multiplication factor (Q is over about 30) should be realized in the DEMO reactor. The power generation is achieved with a power generation blanket designed in accordance with technologies (structural materials, coolant, operating temperature, etc.) selected based on the progress of structural materials. In addition, a high availability attempted by improving the reliability of operation makes it possible that the first wall of the blanket is exposed in high neutron fluence (3-7 MWa/m^2). It is then that the structure soundness of the blankets will be verified. The DEMO reactor is a large device constructed at a country's initiative--and its purpose is to complete the final phase of fusion research and development. At the same time, it is indispensable as a device that has a technological connection with a commercial reactor.

It is presently nearly meaningless to discuss in detail the research and development promoted in parallel to the core device in the demonstration reactor phase. This is because important subjects strongly depend on results obtained in the experimental reactor phase using ITER as the core device and the needs and requests from a changeable society. Such R&D, therefore, has not been described in Fig. 2.1.2-2. Needless to say, it is inevitable to aim for higher reliability, economic efficiency, and environmental preservation for the practical use of fusion energy by the utilization of the extended performance phase of ITER, as described in the previous section. As the research and development on the experimental reactor advances, a series of evaluations are to be made. As the requirements and features of a commercial reactor become clearer, it then may be appropriate that the research and development promoted in parallel to the demonstration reactor be clarified.

At the conclusion of the research and development in the demonstration reactor phase, the results gained is expected to lead to construction of a commercial reactor. In other words, it is necessary to include the following results at the conclusion of the demonstration reactor phase. These results will provide the needed economic aspects that private electric power companies will consider when making decisions about the construction of a commercial reactor, as well as the prospects of the improvement of the plant characteristics. That is to say, technological issues from the DEMO reactor toward a practical (commercial) fusion reactor should be made clear and should be resolved. They should offer the technological bases related to operation and maintenance, cost reduction, and systems management. Performance enhancements and advanced technology with the aim of cost reductions, obtained in the experimental reactor phase, are to be adopted and then verified. The breeder blanket and the low-activation materials should be technically verified.

The design parameters for the demonstration reactors by a conceptual design study in each country are shown in Table 2.1.4-1 in comparison with ITER. In this table, the design results of a commercial reactor are also included as a comparison.

Table 2.1.4-1 Design Parameters for Various Tokamak Reactors

	ITER	ARIES		CREST-1	DREAM	IDLT		SEAFP	SSTR	
		ARIES-Ia	ARIES-RS			Demo	Commercial Reactor		SSTR	A-SSTR
Major Radius R (m)	6.2	6.75	5.52	5.4	16	10	10	9.4	7	6
Aspect Ratio A	3.1	4.5	4	3.4	8	3.5	5.4	4.5	4.1	4
Plasma Current Ip (MA)	15	9.7	11.3	12	9.2	20	12	10.4	12	12
Safety Factor q (95%)	3	4.5	3.5	4.3	3	3	3	3.9	5	4.8
Ellipticity κ	1.7	1.8	1.7	2	1.3	1.5	1.7	1.66	1.85	1.8
Bootstrap Current Ratio Ibs/Ip (%)		68	88	85	87	13	37	84	75	80
H facter (ITER89p)	2	3.7	2.4	2.9	2	2.0	1.8	(2.7)	2	2.7
β_N	1.8	3.2	5	5.5	3	1.2	2.7	3.5	3.5	4.2
$H\beta_N$	3.6	11. 8	12	16	6	2.4	4.9	9.5	7	11.3
Maximum Magnetic Field (T)	11.8	21	15.8	12.5	20	12.5	13	12.8	16.5	(20)
Neutron Flux (MW/m^2)	>0.5	2.5	4.0	4.5	3.0	0.4	2.3	2.1	3	6
Materials	SS	SiC	V	Fe	SiC	SS		V(Fe)	Fe	

The Japan Atomic Energy Research Institute made the conceptual design of a steady-state tokamak power reactor, SSTR (Steady State Tokamak Reactor) in 1990. The design was based on knowledge collected on fusion plasma and reactor technology including the demonstration of the high bootstrap current discharge in JT-60, high magnetic field superconducting magnet technology, and the advance of the high-energy neutral-beam injection technology. This design showed that energy balance of a fusion power plant of about 1-GW electric generating power was feasible using realistic plasma physics and engineering technology (Fig. 2.1.4-1).

A high poloidal beta value (β_p ~2) and a high safety factor (q_a~5) were employed to realize the high-efficiency steady-state operation feature of SSTR. Furthermore, it had features such as high-efficiency steady-state operation by the 2-MeV beam current drive, reduction of disruption frequency by a high magnetic field (9 T) and a high safety factor operation, a higher normalized beta by current profile control, low-temperature high-density radiative-cooling divertor utilizing gas puffing. In addition, the largest experienced magnetic field of the coil was chosen to be 16.5 T, and the compactness of the equipment was also envisioned for the economic and efficiency improvements. The helium refrigerator capacity was a total of 64 kW (electric power 22 MW for the refrigerating machine) including the liquefaction load. The neutral beam injector (50% system efficiency) with a beam energy of 2 MeV and a beam power of 60 MW was employed as its heating and current drive system. The size of the beam line was enlarged to deflect the beam. In the blanket structural materials, low-activation ferritic steel F82H, which has excellent irradiation resistance characteristics, was adopted, and a cooling medium in this operating temperature region was high-temperature pressurized water (15 MPa) which is excellent in heat removal performance and in shielding structural materials from neutrons. The solid breeder Li_2O was adopted as a tritium breeding material and beryllium (Be) was adopted as a neutron breeder material. A high tritium breeding rate (TBR = 1.2) and multiplication factor (1.36) of neutron energy were calculated. The permission fluence was set in 7 MWa/m^2 to avoid excessive risk for the fusion blanket structural material. The blanket was a two-layer structure, and a system was applied in which the thin blanket (~20 cm) was to be replaced every two or three years. Stationary blankets, which were equipped behind the exchangeable blankets, were designed to be used during the entire reactor life (30 years).

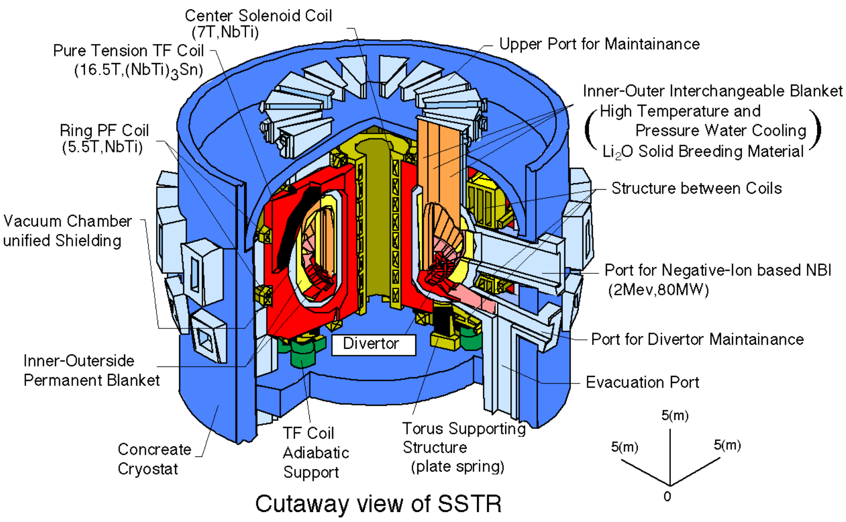


Fig.2.1.4-1 Conceptual scheme of SSTR power reactor

2.1.5 The phase of the commercial reactor

In the phase of the prototype reactor, it is presently difficult to predict the necessity of one integration device (the prototype reactor) having the specific purpose of economic aspect (establishing the basis for becoming a competitive force in the market). The necessity strongly depends on depth of technology achieved in the demonstration reactor phase and on the needs of the public for energy source a half century from now. According to the results obtained in the demonstration reactor phase and the needs and attitudes of the public, it should be possible to introduce the commercial fusion reactors into the market, thus, no reactor is needed for integration after the DEMO reactor.

The practical utilization of a fusion reactor will be realized in the phase of the commercial reactor. In spite of the present intense sense of the economic advantages of fusion power generation, its commercial success will be strongly dependant on the results obtained in the demonstration reactor phase and domestic energy costs at that time. In the commercial reactor phase, although substantial improvements will have been made in economic efficiency and reliability, the performance of the first commercial reactor is based on estimates relating to present designs and knowledge. The generation of net electric power (1~1.5 GW) and operation with high availability (~ 70%) are considered with the system (structural materials, coolant, operating temperature, etc.) chosen based on the progress of structural materials and power generation technology that are expected to be developed in the demonstration reactor phase. Through these operations, high neutron fluence (7 – 14 MWa/m^2) to the first wall blanket should be realized, and the operation should verify the strength and long-life qualities of the structure.

The DEMO reactor must be provided with the optimization of the operation, the maintenance, the system management, and the reduction of costs, and also with the potential to compete in the market, as well as existing power plants. As such, many technologies that would be verified in the demonstration reactor phase would mainly be improved based on the DEMO reactor from the viewpoint of risk management in the commercial reactor phase.

In the development of fusion aiming at commercial reactor, attention should be paid to the necessary conditions for putting the fusion reactor to the practical use. That is to say, reasonable construction costs and low-cost power production are the most important, and furthermore high safety and the reactor's contribution to environmental preservation are also important.

A pulse operation, using an inductive current drive, and a steady-state operation, using a non-inductive current drive, are the two operating scenarios of a tokamak fusion reactor. For the pulse operation using a long-pulse operation of about 12 hours without the use of a heat accumulator, the construction cost is estimated to be approximately 1.5 times that of the steady-state reactor. The cost is 1.3 times higher even with short-pulse operation of about one hour and the use of a heat accumulator. A similar US evaluation estimated that the construction cost would be 1.5 times that of the steady-state reactor. In addition, the pulse operation has technical problems such as thermal cycling fatigue that result from the cyclic operation. Hence, it is reasonable to establish a development program on the basis of a tokamak reactor with the steady-state operation. However, the circulating electric power for the non-inductive current drive must be reduced and the energy efficiency of the entire plant must be increased in the steady-state tokamak reactor. The energy efficiency of the entire plant is a function of the energy multiplication factor Q of the plasma and the thermal conversion efficiency of the power generator. The design goals of the demonstration reactor in each country are Q = 20~50 with an energy efficiency of 30-40%. Therefore, in a fusion commercial reactor, Q = 20~50 in a proper steady-state operation is required; it is unnecessary to aim at Q =∞.

Although design research on fusion power reactors began in Japan in the 1980s, a realistic design based on the research results obtained at the large tokamak JT-60 is represented by the design study (at JAERI) of the steady-state tokamak reactor SSTR in 1990. In the SSTR, as a demonstration reactor, low-activation ferritic steel was employed for structural materials from the viewpoint of a conservative engineering design--this reactor only relies on materials that function in temperature conditions found in a light water reactor. There are later fusion reactor conceptual designs, the A-SSTR (JAERI) and the CREST (Central Research Institute of Electric Power Industry), where the economic aspects are improved as required in the commercial reactor phase. In addition, there is another conceptual design, DREAM (JAERI), which has excellent safety and environmental preservation features.

The SSTR concept of a fusion reactor, which was designed based on the current knowledge, is scientifically feasible and has a comparatively high chance of meeting the design goals. However, the construction costs then were double those of a light water reactor. In addition, the cost of electricity was also double, assuming the fuel costs were in proportion to the construction costs and that the expense rate of the operating cost was equal to those of a light water reactor. The cost of electricity remained 1.5 times higher even if fuel costs and operating costs were remarkably reduced. The motivation that promotes the development of the fusion energy to replace exiting energy sources is difficult to justify. This is because the burden of the Electric Power Companies that are in charge of construction and operation of the reactor becomes excessive, even if a fusion reactor has the advantage of inherent safety.

Based on the above general formation of a fusion commercial reactor, the requirement of a fusion reactor was examined from the view of the user, the electric utility company. The A-SSTR fusion reactor, a more current conceptual design, can satisfy the requirements. The features it offers are the high magnetic field and the reduction of the refrigerating machine capacity by using high-temperature superconducting coils at a low-temperature (27 K). Using oxide dispersion strengthened (ODS) low-activation ferrite steel for the structural materials, the plant can realize high thermal efficiency, efficiency which is the same as a general thermal power plant. The normalized beta value has been set at the $\beta_N = 3.7$, which is a little higher than SSTR. The fusion output of the reactor is 3.53 GW and the thermal output is 4.3 GW; accordingly the net electric generating power is 1.63 GW. By improving the thermal efficiency and increasing the beta value, the net electric output is increased. Furthermore, two ideas, the sharing of peripheral facilities by siting two plants at one location and the use of a compact-sized reactor design, are adopted in the design. The diameter of the main body for the tokamak fusion reactor is about 22 m and the gross weight is the about 20,000 tons. The primary candidate of the blanket structural materials is ODS low-activation ferrite steel, which is expected to have its maximum working temperature increased to about 600ºC.

References

[2.1.1-1] ITER special working groups second task report, January 1999.
In the meeting, as the retreat from the ITER activity by the US became clear, the four parties (including the US) debated the relationship between ITER and the fusion strategy, taking a broad view of things.

2.2 ITER as an experimental reactor and its evolution

In the Third Phase Basic Program of Fusion Research and Development, which was defined by the Atomic Energy Commission, the objectives of the fusion research and development in this phase are described as follows.

The principal objectives in the third phase are achieving the self-ignition condition, realizing a long burn, and forming the reactor technology bases for the DEMO reactor development. A tokamak fusion experimental reactor is to be developed as a device to play a central role in attaining the goals of this phase. This third phase of research and development should result in sufficient prospects to continue fusion research and development into the fourth phase and beyond. To achieve these objectives, research and development should be conducted in the following subject areas.

(1) Research and Development to Achieve Self-Ignition and Long Burn in a Tokamak Fusion Experimental Reactor

(i) Self-Ignition Condition

To achieve the self-ignition condition (i.e., a fusion energy multiplication factor of above around 20), it is essential to extensively improve the energy confinement of high-performance plasmas and to enhance the heating power ratio of the high-energy alpha particles in the total input power.

(ii) Long Burn

To attain a long pulse operation (i.e., a duration of longer than about 1,000 sec), which is prerequisite for realizing a steady-state fusion reactor plasma, research and development must be done on such subjects as the high-efficiency current drive, the reduction of heat flux onto divertor plates, the removal of helium ash, and the avoidance of disruptions.

In addition, development of reactor technology is described as follows.

(2) Reactor Technology

The development of major components with large sizes and improved performance is necessary for an experimental reactor. Furthermore, research and development including tests using the experimental reactor should be done with the aim of forming the bases of reactor technology necessary for the development of the DEMO reactor. Therefore, research and development should be pursued on such technical issues as large, high-field super-conducting coils; remote maintenance technology and reactor structures feasible for remote handling; plasma facing components durable in a high heat load and with favorable heat removal capabilities; high power and long pulse heating and current drive devices; the technology for tritium production, breeding and handling; the blanket technology; etc. Concurrently, technology will be required to integrate these components and devices into a single system.

The performance of ITER, as an experimental reactor for achieving the above-mentioned goals, and the results that will be obtained by research and development using ITER are described in this section. The significance of ITER from the point of view of international cooperation, consideration of the domestic lure of ITER, and tokamak research that will support the ITER project are also described.

2.2.1 ITER

2.2.1.1 The goal of ITER

The overall programmatic objective of ITER is to demonstrate the scientific and technological feasibility of fusion energy 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. There is considerable width on the specific design parameters in ITER that will achieve these goals and, depending on the extent of proximity to the DEMO reactor that will follow the experimental reactor, construction costs and scale. The final design report, referred to as the ITER-FDR, where the results of the 6-year design effort, from1992 to 1998, are described, was completed in July 1998. However, continuation into the construction phase in 1998 as scheduled was not realized. Then, the parties attempted to again initiate ITER construction by the reduction in the size of the reactor. Japan asserted that steady-state operation was a high priority in research and development toward a fusion power reactor according to the Third Phase Basic Program of Fusion Research and Development, and obtained the agreement of other parties whom initially had been opposed to size reduction. As a result, it became possible to design a reduced-cost but still viable ITER by fixing the concept of the steady-state operation as a very important target.

2.2.1.2 Outline of the technical guideline for achieving the goal

According to the above-mentioned new policy of ITER, the ITER council set a new technical guideline in 1998. The design of ITER that satisfies this guideline is now being promoted (Table 2.2.1.2-1). This guideline is intended to identify issues to be solved for the DEMO reactor: 1) the steady-state burning plasma with high bootstrap current and current profile control with non-inductive current drive, 2) the high-performance plasma with the high-efficiency divertor including the components that can withstand steady high-heat flux, 3) the superconducting magnets compatible with high-power DT burning plasma, 4) the remote maintenance in a nuclear fusion power system of a reactor scale, 5) the test of tritium breeding blanket and structural materials and 6) the tritium technology.

According to these new technical guidelines, top priority is to be given to achieving an extended burn with the ratio of fusion power to auxiliary heating power of at least 10 and with a pulse operation of 300-500 sec and with a major radius of 6~6.5 m. In addition, steady-state operation and the possibility of controlled ignition are retained.

Table 2.2.1.2-1 Technical guidelines of ITER.

category	technical point	Former technical guidelines (1992)	New technical guidelines (1998)
Goal	Demonstrating scientific and technical possibility of the fusion energy	same as the left	• To minimize the extrapolation error to DEMO • To enable the DEMO construction (without any stop) as the next step
Plasma performance	Controlled ignition Burn pulse length	Q = ∞ and, long burn (about 1000 sec in induction system)	Long burn in Q≥10 (300~500 sec in induction system) The possibility of Q~∞ should not be precluded.
	Steady State Operation	aiming at demonstrating steady-state operation using non-inductive current drive with Q of at least 5	
Engineering performance and Testing	Demonstrating the technology which is indispensable to nuclear fusion reactor	• Demonstrating the availability and integration of technologies (such as superconducting magnets and remote maintenance) • Testing high heat flux component and blanket	
	Testing components for a future reactor	Average neutron flux : > 1.0 MW/m2 Average fluence : > 1.0 MWa/m2	Average neutron flux : > 0.5 MW/m2 Average fluence : > 0.3 MWa/m2

This guideline strongly stresses the early realization of a nuclear fusion reactor based on the latest results of plasma physics and verified technologies obtained in the ITER-EDA for the last 7 years. The guideline also reflects the accords of "The Third Phase Basic Program of Fusion Research and Development of Japan," shown in Fig. 2.2.1.2-1, and acknowledges the demands described in Section 2.2.1.

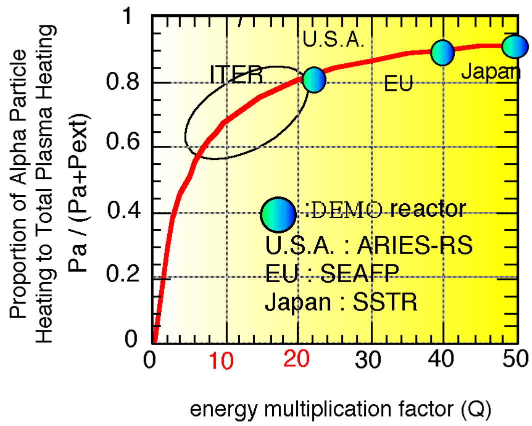
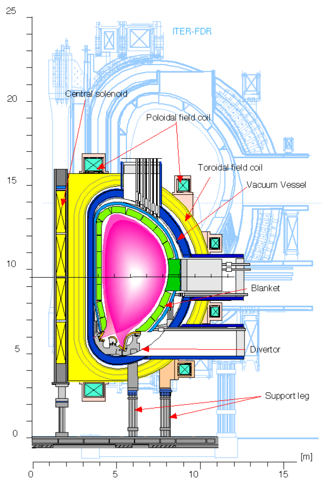


Fig.2.2.1.2-1 The characteristics of ITER and demonstration reactor designs of the EU, Japan, and the US

2.2.1.3 The fundamental design philosophy

Along with the new technical guideline, the conceptual design of the reduced cost option is set forth aiming at a 50% reduction in cost for the reactor described in the FDR (ITER-FDR). The design of the new ITER, which is shown and compared with the ITER-FDR in Fig. 2.2.1.3-1, enables plasma operation with a maximum burn time of 400 s (with an inductive current drive), a plasma current of 15~17 MA (about 71~81% of ITER-FDR), a nuclear fusion output of 500~700 MW (33~47% of ITER-FDR), and an energy multiplication factor larger than 10.



Major parameters	ITER-FDR	A new design of ITER
Major radius (m)	8.14	6.2
Minor radius (m)	2.8	2.0
Fusion Output (MW)	1500	500~700
Burn Time (sec)	1000	~400
Plasma current (MA)	21	15~17
Toroidal field (T)	5.7	
at the plasma center	12.5	5.3
at the maximum point	∞	11.8
Energy amplification factor	100	10-(50)
Maximum heating power. (MW)	1.0	40~(100)
Neutron wall load (MW/m)	1.0	~0.6
Neutron fluence (MWa/m)		> 0.3

Fig.2.2.1.3-1 Comparison in tokamak cross section between ITER-FDR and a new design of ITER

2.2.1.4 Plasma performance

It is necessary to consider both sides of engineering performance and plasma performance to satisfy the new technical guideline. In ITER, the optimization is being attempted on the basis of the latest database to ensure the appropriate confinement margin in operation with a finite energy multiplication factor (≥ 10).

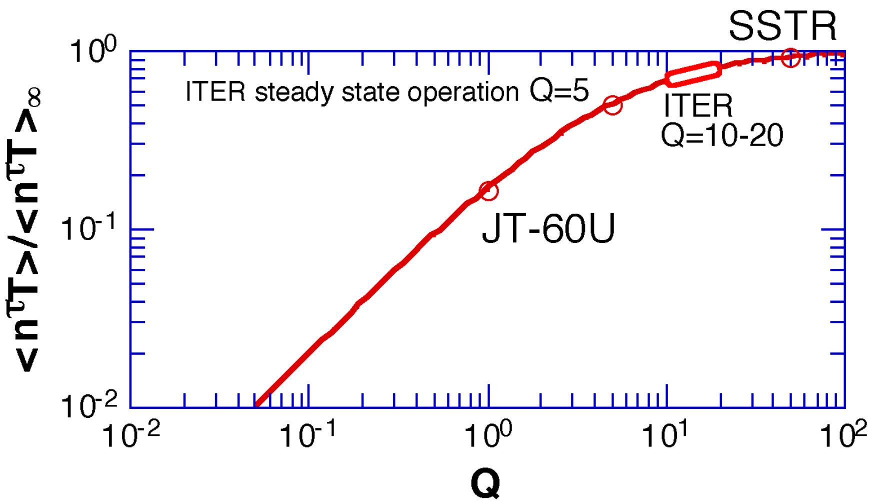


Fig. 2.2.1.4-1 Relationship between Q-value and fusion tripple product

The fusion triple product <nτT>, which indicates confinement performance, and the energy multiplication factor Q have the relation of $\langle n\tau T\rangle/\langle n\tau T\rangle_\infty = Q/(5+Q)$, where $\langle n\tau T\rangle_\infty = (35 \pm 5) \times 10^{20}\ m^{-3}$ s.keV has a value of <nτT> at Q=∞. As shown in Fig. 2.2.1.4-1, <nτT> is proportional to Q when Q is smaller than 1, but it approaches saturation after Q becomes larger than 5. The difference of values of fusion triple product, which represent confinement performance, between ITER and SSTR has proven to be small, though the Q-values are much different.

1) The performance of ITER for self-ignition (energy multiplication factor Q is larger than 20)

Figure 2.2.1.4-2 shows the predicted operation domains of ITER in the self-ignition region, (Q more than around 20). The operation domain is restricted by various operational boundaries such as the plasma density limit (the Greenwald density), the plasma pressure limit (beta limit), the H-mode/L-mode transition condition, etc. Here, the confinement enhancement factor, which corresponds to the horizontal axis, is defined by the ratio of required energy confinement time over the estimated value using the experimental database that is important in evaluation of plasma performance. As seen in the figure, the Q = 10 operation at about 80% of the Greenwald density and a normalized beta less than 2 can be achieved with a confinement margin of about 15% in ITER. The Q = 20 operation is also expected to be achievable with a confinement margin of about 10%. Furthermore, the Q = ∞ operation is expected to be feasible by raising the plasma current, as is shown in Fig. 2.2.1.4-2, at the operation condition covered by the present database (HH = 1 and 90% of the Greenwald density). Therefore, it is considered that ITER designed in accordance with technical guideline described in Section 2.2.1.2 can achieve a Q-value of around or larger than 20, which is required in The Third Phase Basic Program.

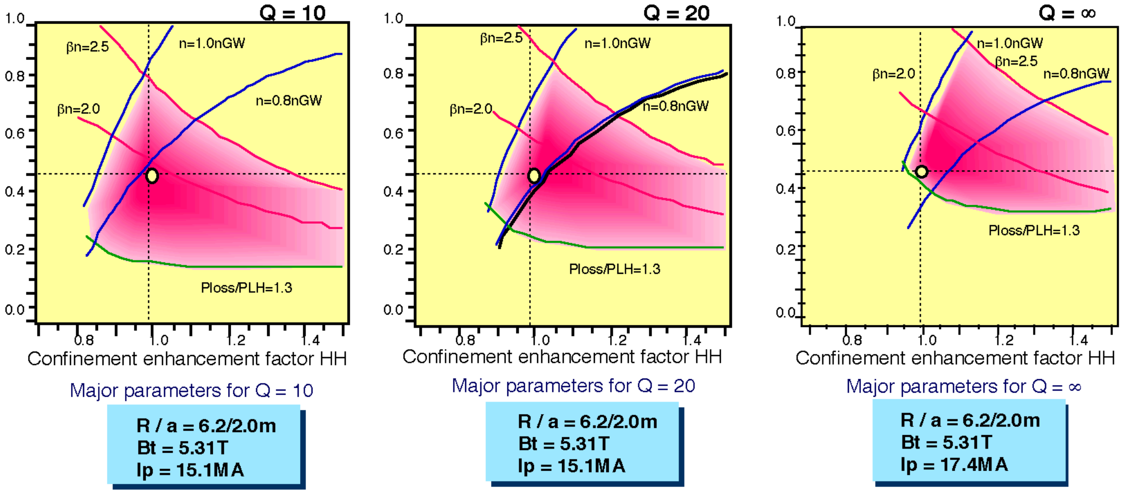


Fig. 2.2.1.4-2 ITER operation domains of Q=10, Q=20 and Q=∞

2) The long burn performance of ITER The burn time of ITER is described as 300~500 sec in the technical guideline in Section 2.2.1.2. However, ITER is designed so that various types of operation, such as steady-state operation, the newly considered H mode-hybrid operation in which inductive and non-inductive current drives are mixed, fully steady-state operation using the reversed magnetic shear mode and so on, can be investigated. Here, ELMy H-mode Hybrid operation is a newly added operation mode with high feasibility, and is suitable for integrated technology tests in the burning environment. An evaluated relation between the energy multiplication factor Q and the burn time of ITER is shown in Fig. 2.2.1.4-3. Burn times at Q = 5, Q = 7~8, Q~10, and Q~20 are expected to be about 2,500 sec, about 1,000 sec, about400 sec, and about 200 sec, respectively.

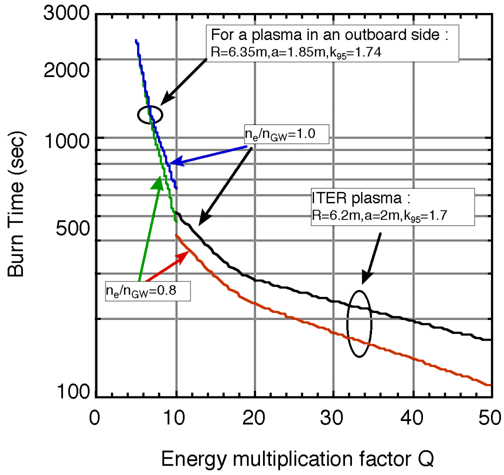


Fig. 2.2.1.4-3 The Q dependence of Burn Time in ITER (Evaluation assuming ELMy H mode confinement (HH=1))

(1)The long burn with the inductive current drive

Long pulse operation is described in the Third Phase Basic Program as prerequisite for realizing steady-state fusion reactor plasma. Relaxation time of the current profile is one of physical time scales that distinguish a long-pulse operation. Since the current profile of the steady-state operation is achieved in about 200 sec with the inductive current drive, as seen in Fig. 2.2.1.4-4: a simulation result of ITER, the current seems able to achieve the situation in which the current profile is relaxed to a steady state, even in a high Q burning experiment of Q=10-20.

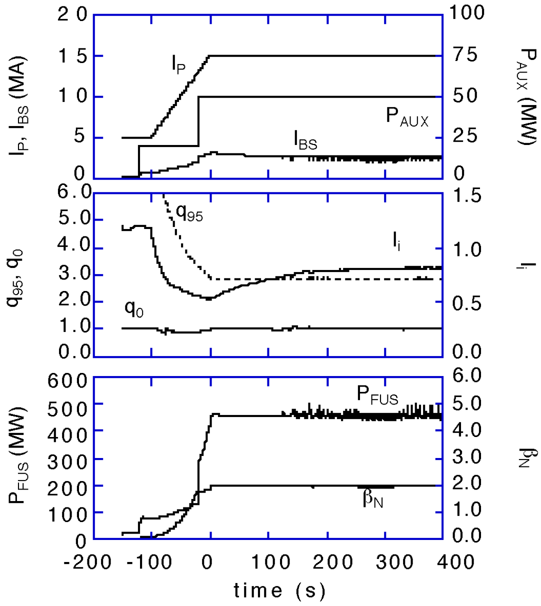


Fig. 2.2.1.4-4 The evaluation example of the current distribution relaxation in ITER

In the meantime, on the time scale of plasma wall interaction, it is necessary to consider the particle recycling process, in which the absorption and de-sorption of the gases greatly depend on the wall temperature. It has been confirmed experimentally that it takes several tens of seconds for the particle absorption of the wall to reach a quasi-equilibrium condition. The thermal time constant of a high heat load component that is facing the ITER plasma, i.e., the first wall or divertor, is expected to be about 300 sec. It takes about 700 sec for the whole blanket-module, the main part of which is the neutron shield, to reach a thermal steady state. However, the pumping time of the divertor is about 10 sec and the particle flux, which is more than 10 times of that of the wall, can be exhausted by pumping.

Therefore, it is predicted that it will be possible to achieve the quasi-steady plasma property condition with the inductive current drive in 300-500 sec of the burn time (a burn time of 400 sec is possible in ITER).

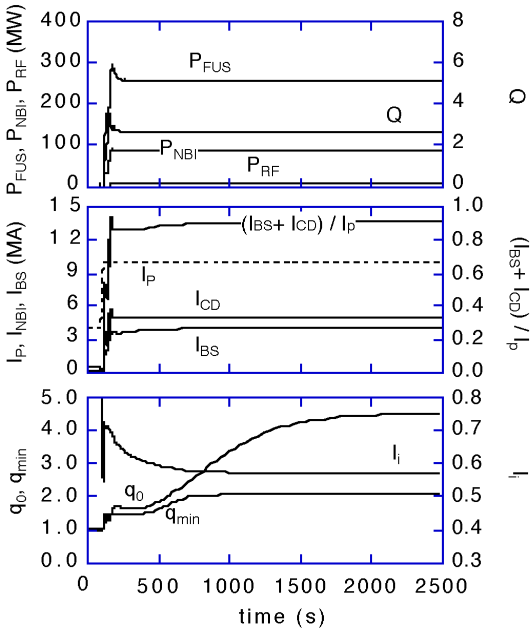


Fig. 2.2.1.4-5 The evaluation example of current distribution relaxation in the ITER hybrid operation where the proportion of non-inductive current to the total plasma current is about 90%.

(2) The long burn in the hybrid operation

Since current penetration is the slowest plasma phenomena in the non-inductive current drive, as shown in Fig. 2.2.1.4-5, a burning time of more than 1,000 sec is sometimes needed to confirm a steady state for q(0). Therefore, ITER is designed so that an operation time of about 2,500 sec is possible using hybrid operation.

(3) The long burn with full non-inductive current drive

The realization of steady-state operation in ITER is very important to develop a prototype of a highly efficient steady-state tokamak fusion reactor following ITER. In Fig. 2.2.1.4-6, an analysis of the ITER operational domain realizing steady-state operation is shown on the plane of the confinement enhancement factor HH normalized by the scaling of the ELMy H-mode confinement (IPB98 (y,2) described in Section 3.1.4) and fusion output, where the various operational boundaries (density/Greenwald density (n_e/n_{GW}), normalized beta value (β_N), current drive power (P_{aux}), and energy multiplication factor (Q)) are also traced. Steady-state operation with Q = 5 is expected when the confinement is improved by 10-20% compared with the ELMy H-mode, while the Q-value is limited to about 3 when the confinement time is similar to that in the ELMy H-mode.

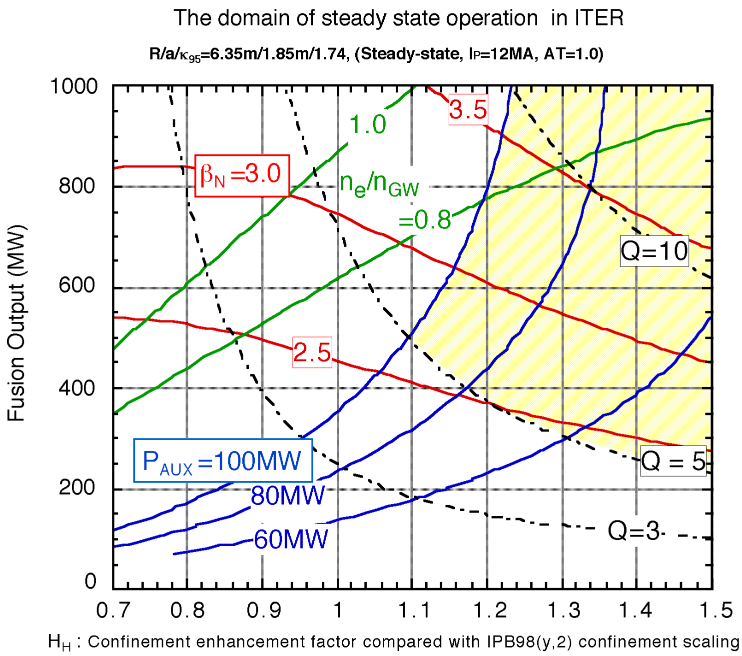


Fig. 2.2.1.4-6 The domain of ITER steady state operation on the plane of confinement enhancement factor compared with ELMy H-mode confinement scaling and fusion out put (assuming parabolic temperature profile). The region where energy amplification factor Q is over 5, and current drive power is 100MW or less is shown in the yellow.

If further improvement of confinement (HH~1.5) and a normalized beta value of about 3.5 are obtained, a steady-state operation with Q~10 would also come into view. Although we have to wait for future research to know whether such plasma performance surpassing ITER physics basis (HH = 1, normalized beta value ≤ 2.5) can be obtained, ITER is at least designed with such potential. Since relatively high-field and high-density operation is possible in ITER, we can expect that the heat load to in-vessel components, especially the divertor, will be reduced drastically, and that there is the flexibility to optimize the divertor shape for the development of the DEMO reactor.

2.2.1.5 Tokamak components

The tokamak reactor is comprised of the following systems: the central solenoid system, which provides the main induction current in the plasma; the toroidal field coil system, which provides the high magnetic field necessary for confining the plasma; the poloidal field coil system, which controls the position and shape of the plasma; the vacuum vessel, which maintains the plasma space in a high vacuum and confines the tritium; the blanket, which absorbs most neutron energy and protects the vacuum vessel from high-temperature plasma and the neutrons produced in burning plasma; and the divertor, which exhausts and controls the impurity particles and helium particles produced in the burning plasma, etc. Except the blanket for the neutron shield, these systems are fundamentally similar to existing tokamaks such as JT-60 and so on. A significant difference is the ability to confine burning plasma from real fuel, and to do so while having a high-energy multiplication factor for a long duration. For this reason, it is necessary that ITER have the tritium fuel cycling system and that each component satisfies the required performance in steady- or quasi-steady-state operation.

The ITER design has flexibility; ITER can be equipped with a tritium production system and a small-scale electricity generation system as the need arises. An example of the ITER system configuration and the major parameters are shown in Fig. 2.2.1.3-1. Tokamak internal components and the tritium fuel recycling system exposed to neutrons beside the burning plasma cannot be realized in the present research devices. The ITER configuration allows important technologies to be developed for the DEMO reactor. Meanwhile the demonstration of various technologies described in the Third Phase Basic Program becomes possible in ITER. In the specific design of ITER components, the optimization of the superconducting magnets that incur a large proportion of ITER cost is noteworthy. Especially, the significant miniaturization and performance improvements of the toroidal field coil (TF coil) are realized by the use of a superconductor that is usable at a higher operational current with an appropriate technical margin than the conventional conductor (Fig. 2.2.1.5-1). This very important improvement was achieved through advancements in fabrication technology and the database developed through R&D for 7 years.

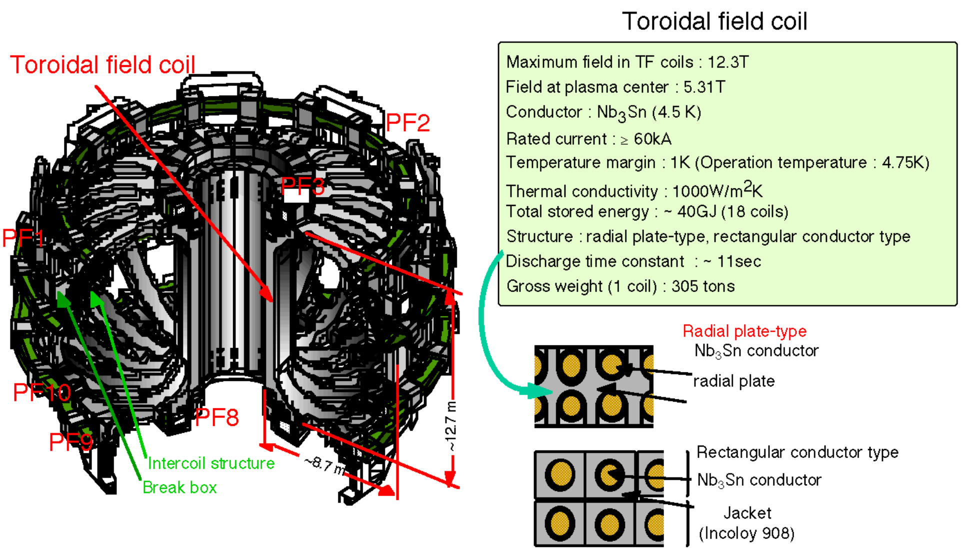


Fig.2.2.1.5-1 Design Performance of ITER Toroidal Field Coils (an example)

The central solenoid is also divided vertically, and each division is controlled to realize the advanced plasma shape and the improved plasma position control. The vacuum vessel has a solid double-walled structure, which is reinforced by ribs set between the internal and external walls, to provide the electrical resistance demanded by the plasma control as well as structural strength. The design minimizes nucleus exothermic reactions in the superconducting magnet and neutron radiation damage to the electric insulators in the coil conductors by providing a neutron shield between the double walls.

For components in the vacuum vessel, the blanket structure has been improved. In the ITER design, the blanket has been composed of a reusable neutron shielding structure and the exchangeable first wall. This attractive design concept, which achieves functional separation of the structure and the first wall, will reduce radioactive waste. The design will be fully verified in the DEMO reactor. The divertor, which has a cassette structure, is installed at the location where the lines of the confining magnetic field cross in order to exhaust helium and impurity particles coming from the plasma and to withstand the high heat load. Each component exposed to high heat flux has a modular structure and can easily be exchanged.

2.2.2 What we will realize on ITER

2.2.2.1 The achievement of the technical objectives

The International Thermonuclear Experimental Reactor (ITER) project is an attempt in which humankind strives to achieve plasma confinement performance necessary for the nuclear fusion reactor and verifies the generation of enormous amounts of energy (500-700 MW for 300-500 sec) by the DT fusion reaction for the first time. Since plasma behavior is nonlinear, extrapolation for fusion reactor physics is more difficult than for fission reactor physics, where linearity is ensured. As representative examples of nonlinear phenomena solved empirically, fluid heat transfer has been determined by a dimensional analysis and plasma phenomena also can be determined by a dimensional analysis. To counter the nonlinearity barriers to scaling, it is anticipated that the experience with ITER may provide new issues only visible in ITER. However, we can conclude with confidence, based on the research history on JT-60 (described in Section 3.1.2), that the technical objectives of ITER are sufficiently achievable, if the entire plan has the robust structure, including the design flexibility of ITER, the resolution of the issues with supplementary devices, and the extension of organization and human resources needed, etc.

2.2.2.2 Massive energy generation and demonstration of integrated technology

Since ITER will achieve plasma confinement performance necessary for the nuclear fusion reactor (Q is about 20) for the first time in nuclear fusion research and development, ITER will have the honor of opening the way for the peaceful use of fusion energy from fusion reactors. It will follow the application of the peaceful use of nuclear energy from fission reactors by about half a century and will finally realize the long-sought dream of nations and researchers, that of "the sun on the ground."

Consider the massive fusion energy generation in this nuclear fusion experimental reactor. The thermal output of ITER will be more that 20 times that of the world's first experimental fast breeder reactor, Clementine (25 kWth), and will have almost the equivalent thermal output of the demonstration fast breeder reactor Monju (714 MW). In addition, the scale of this tokamak device is as same as nuclear fusion tokamak demonstration- and prototype reactors that are presently in the conceptual design stage. Six reactors, Clementine, EBR-I, EBR-II, E.FERMI I, SEFOR, and FFTF, have been constructed in the fast breeder reactor development in the US, even if limited to experimental class reactors. However, the development strategy of the fusion reactor differs from that of fast breeder reactors in that ITER has many roles including the role of opening the way to the peaceful use of fusion energy. In addition, ITER offers many fields of science insights into various and complicated phenomena, as described in Section 4.1.

In a nuclear fusion reactor, the burning medium is the high-temperature (several hundred million degrees) plasma, which does not exist in nature on earth. Therefore, it is necessary to develop the various advanced technologies such as the superconducting magnets for providing the strong magnetic fields needed to isolate the plasma from the wall, the high heat-flux components, the 14-MeV neutron/gamma-ray shielding blanket, the vacuum vessel, the ultra-high vacuum techniques, the tritium handling technology, the high energy neutral beam technology, the high power radio frequency technology, the ultra high-temperature plasma measurement technology, and the remote maintenance technology. The technologies necessary for a nuclear fusion reactor were (and are being) greatly advanced in ITER-EDA (1992~2001) as technologies needed to realize ITER. Except advanced materials technologies and electricity generation technology, the nuclear fusion research and development efforts performed in ITER will establish most major reactor technologies. Such advanced technologies were unneeded in the development of fission reactors, but the development of heat removal and high-temperature structural materials technology was the core development of the fast reactor. As seen from the above description, an important role toward the integration of technology for nuclear fusion reactor application as a power generating system is given to the ITER nuclear fusion experimental reactor. The present ITER omits only the technology that is considered necessary to remove the high-grade heat produced in the tritium-breeding blanket.

The mission of "self-ignition and long burn" of the experimental reactor defined in the Third Phase Basic Program is close to the role of the Critical Assembly in fission reactor research. The characteristics of the nuclear fusion experimental reactor are different from those of fission experimental reactors, which included the testing of heat transfer technology under the nuclear environment. The development of the tritium breeding blanket and electricity generation are matters to be considered as important options in the latter half of the ITER operation plan. Assessment of these options is included in the check & review of the ITER project, which will be carried out on the basis of the results and trends of structural material/blanket research and development performed in the first 10 years of the ITER program. Retaining this flexibly is desirable to deal with the changes in development periods for practical applications and changes in the international and domestic energy situation. However, there is no necessity to assign rigid specifications to ITER in the beginning. In the fission reactor, heat transfer technology under the nuclear environment was (and still is) studied for many years. Therefore, it is necessary to upgrade that technology by making full use of the knowledge obtained in fission research.

2.2.2.3 Physics R&D and technology tests for DEMO using ITER

The efforts of research and development in the experimental reactor in the Third Phase Basic Program have appropriately been defined as the goals that can be foreseen in the current stage. Further, these goals can be considered achievable in research and development performed with ITER. After the technical objectives of the experimental reactor ITER are achieved, it will be necessary to establish the physics and technology bases to decide the construction of DEMO. They are control of high-beta steady-state operation, control of divertor thermal flux that exceeds the flux of ITER, establish methods for the avoidance and relaxation of disruptions, and develop and test blanket/divertor materials and designs for power generation. The confirmation tests of these bases will be carried out at the enhanced performance phase of ITER (the operation phase in the latter half of the ITER program), which will follow the completion of sufficient development activities.

2.2.2.4 ITER from the viewpoint of practical application

Fusion energy is not a practical industrial technology in its current stage of development. It is necessary to show the step of fusion research to practical application, since the purpose of this research is the practical utilization of this form of energy. The stepwise approach in nuclear fusion development is similar to that in fast breeder reactor development. In this program, ITER will generate a massive amount of fusion energy and DEMO will demonstrate electricity generation and the Prototype will demonstrate economic efficiency. However, ITER as an experimental reactor is different from the experimental fast breeder reactor as described above. The technical requirements of the core device for each stage, which are decided by considering the characteristics of nuclear fusion, are not equal to those of the fast breeder reactor, and should not be so.

The steps where the government takes the initiative are fundamentally only two: the experimental reactor, ITER, and the demonstration reactor DEMO. In these two steps we have to establish the technology base with which the industrial world can decide, on its own initiative, if it should proceed with the construction of the first commercial nuclear fusion reactor. The economic efficiency of this reactor will be the biggest subject.

Construction costs of the core devices in each step always attract much attention as an index to the cost of the commercial reactor. Therefore, it is important to advance research and development in ITER in a cost-effective manner, yet without missing the goal to be achieved.

2.2.3 Common understanding of philosophy to share ITER construction cost and its significance in the frame of international collaboration

The “confrontation” between the two major powers has shaped the political structure of the world in the last half of the 20th century. There has been much concern about easing the tensions caused by the confrontation and establishing harmony within the international political structure. International collaboration in the science and technology areas has also influenced the world’s political structure. With such background, ITER design activities started as a pioneering, symbolic work of détente from a cold war lasting a half-century. Similarly, in the 21st century, a world structure based on “confrontation” will almost certainly change to that based on “co-operation.” This change is currently leading to a new era, in which every necessary function for humankind’s activities should be discussed in the international collaborative frame. As examples of a large-scale international collaboration with participation of scientists and engineers worldwide, the space station project hosted by the United States, the accelerator science project LHC hosted by the EU, etc., are in progress.

Most international collaborations that included the participation of Japan have been proposed or coordinated by other parties at the first stage. Japan usually joined the collaborations at the signature stage. Therefore, if Japan succeeds in hosting ITER construction, this would be the first occasion Japan has played a lead role in a large-scale international collaboration, which is suitable in the coming era of “cooperation”.

As will be discussed in Section 4.5, international collaborations in the field of nuclear fusion are being proposed and conduced vigorously. In recent years, Japan has been asked to participate in all new international collaborations in fusion research. This implies that the level of fusion research in Japan is highly rated, and it also indicates that Japan presently has the scientific and technological potential to host an international “cooperation.” Although an international collaboration has the sublimity of contributing to the world from a political and strategic point of view, it is a reality that the benefits will be provided to every nation participating in the collaboration. This basic understanding would be required for successful international collaboration. In addition, it is necessary to recognize the collaboration is a strategic type of international contract in fields of science and technology.

2.2.3.1 A philosophy for sharing ITER construction costs

----- What benefits could balance the expenditure for ITER -----

The benefits that could balance the expenditure for ITER, especially the costs involving hosting ITER, are now being debated. Two case studies will be discussed for the possible answers to the question.

(Case 1) The costs and benefits are balanced within the ITER project.

This concept is to balance the costs with the benefits within the ITER project. There exist two ways of thinking:

- The first idea: The basic principle in ITER engineering design activities (EDA), the philosophy of equal contribution and equal benefit, would be employed for the cost sharing of the construction costs of ITER. The total cost except the cost of certain exempted items that only the host party could share should be shared equally by the participant parties. The results should be shared, and the speaking rights in the steering committee should be shared equally.
- The second idea: Since it is, in practice, difficult to share the costs equally, the concept that the distribution of the results to the parties could correspond with each party’s cost sharing ratio is being explored. This idea has been currently adopted in the preparatory discussion under the ITER agreement.

(Case 2) The costs and benefits are balanced totally within world fusion research.

- This concept is that the costs and benefits are balanced, not necessarily within the ITER project, but within the fusion research activities in the world. This idea was proposed for the case where it would be difficult to keep the balance of costs and benefits within the ITER project. An example of this balance would be that if party “A” hosts ITER construction, then party “B” would host the construction of the 14-MeV neutron source, and party “C” would host the plasma experimental facilities to support the ITER project. In the early period when Japan, the United States, and the EU were each eager to host ITER, this idea, called “basket of fruit,” was considered as a solution.

Regarding these cases, the following statements have been presented:
(1)Since an international collaboration should reduce the cost to each party and utilization of a wider knowledge base than with independent development by a single party, it would never create a loss for the participant parties. Therefore, how to make profits exceed the shared cost would be the essential issue for each party.

(2)Fusion research is based on highly advanced technology and development. In contrast, construction (manufacture) of a major fusion device is the main goal of the development. Since only one fusion experimental reactor is supposed to be built in the world, the technology developed for ITER would likely become the world standard for fusion devices. This is a substantially different philosophy from developments in accelerator science, where construction of the large device is not a major mission, but is merely a tool to attain the physics research mission, although some R&D for manufacturing is needed. Consequently, the manufacturing technology is accumulated only in the nation responsible for the actual manufacture. This widens the gap in key technological capabilities between the nation in charge and other participants that expect only technical information. If nations could share manufacturing techniques and procedures in proportion to the amount of cost contribution by each nation, nations with high sharing ratios would never lose their investment in a relative comparison among the parties. This investment will also effectively contribute to risk reduction in the construction of the fusion DEMO reactor.

(3)In the progressing Exploration meeting on ITER construction and operation, an idea is employed that the balance be kept within the ITER project. Since the ITER/EDA principle of “equal contribution and equal benefit” had been simply applied to the ITER reactor which would be built at a single site, there have arisen a lot of difficulties. Now by employing a practical principle of “fair return for costs shared,” an agreement for the ITER construction has been explored in the meeting.

(4)The significance (necessity and importance) of fusion in the energy development strategy of each individual party is different from that of other parties. Unfortunately, this is considered unavoidable. The United States, which has abundant domestic energy resources, and our nation, Japan, which has few domestic energy resources and must rely on imported resources, would, at this moment, never have the same views on collaborative fusion research and development. In the case where the parties participating in the ITER project share the total cost unequally, the idea “fair return for costs shared” is considered acceptable.

(5)However, correspondence between the shared part of the cost and the amount of return benefit/result cannot be clearly specified under the common criteria with international consensus. Since the criteria would depend on the parties’ subjectivity, such as their accumulated technology, their views of the future, etc., an agreement about the idea “fair return for shared costs” should be made under international discussions.

(6)In regard to the balance made within the world fusion research community, mentioned in (2), this concept is based on “equal contribution and equal benefit.” However, since other fusion device developments are smaller in scale than the ITER project, the principle of “equal contribution and equal benefit” had not been applied to the actual fusion projects. Therefore, the Sub-committee considers it is appropriate to adopt a principle “fair return for shared costs,” although other views on more flexible sharing are never to be excluded.

2.2.3.2 Significance of international collaboration in fusion research from the social, economical, and security points of view

--- A linkage with the international community ---

Effective accomplishment of an international collaboration is often expected from the finance and manpower viewpoint. However, an international collaboration should not be evaluated only from the aspect of science and technology. A social, economic, and cultural evaluation is necessary to understand an international collaboration as an issue of a nation's significance in the world community.

(1)Technology for fusion, and advanced science and technology created by fusion

--- ITER is a pioneering project for the energy source of the future ---

After World War II reconstruction, Japan, with limited domestic resources, has rapidly become a major economic power. Moreover, she has endured two world oil crises by improving her energy consumption efficiency and correcting constitutional (governmental) defects. Japan has finally has come to be one of the most advanced nations in the world, which certainly is a remarkable success story. Japanese economic growth has been driven by mass production of high quality produces coupled with sizable domestic consumption, such as products from the automotive and electronics industries. Economic growth has also been driven by export of both consumer and industrial products. Recently the absolute superiority in manufacturing technology enjoyed by Japan is being challenged because of the rapid growth of the developing nations. Therefore, if Japan is to retain its competitive power, it should be understood that continued improvements are necessary. Consequently, the enviable position enjoyed by Japan within the international community cannot be guaranteed in the future.

Since Japan is a nation poor in energy and mineral resources and depends on food imports, it should always continue the development of advanced technologies, and to export these technologies as well as the products resulting from them. This is necessary to keep pace (and even to survive) in today's changing international economic situation. Therefore, it is a crucial issue for Japan to retain its international position in the fields of science and technology for which it is now recognized as having expertise. We should now plant the seeds that will provide a rich harvest in a quarter or half-century. A fusion device is an integrated product that uses advanced technology, and this technology can be widely applied to various industries and products. If Japan takes the lead in the world fusion development by hosting ITER, and makes the applicable technology available to other users; the export of this technology can provide substantial economic returns for Japan. In addition, the export of technologies that have propagated from the fusion should be noted, because fusion technology is totally related to a wide network of fields of science and related technologies.

Whether Japan, one of the front runners in the world fusion research, can take the lead in the ITER project, seems to be strongly correlated to whether Japan will be a leading position in extending fusion to affect industry, and domestic economy a quarter century after ITER program. In this sense, we now stand at this important crossroad of the future.

(2) Multi-sided personal exchange, mutual understanding in depth, and social, economic and political roles toward détente

International collaborations in fusion research have been strongly encouraged since the late 1970s. The collaborations have taken a great variety of forms: large international conferences, large/small workshops, mutual participation in experiments, joint experiments of the super-conducting magnet "LCT project," etc., joint design of the experimental reactor INTOR, the material irradiation facility "IFMIF," etc. Through those collaborations, research scientists, administrative personnel, and bureaucrats in each nation have established mutual exchanges and trustworthy relationships. It is a reality that the success of early collaborations has lead to the establishment of wider collaborative relationships.

The presidents of the United States and the Soviet Union proposed an ITER project as a symbol seeking east-west peace in 1985 when East-West wall still exists. The background basis of international collaboration had been established, so a great advance seemed promising. The ITER project is the largest and broadest fusion research plan conducted until then. Indeed, many people have supported and collaborated on this project in various aspects: Research scientists and engineers from the research organizations, universities, and industries participating in the project from the science and technological aspects. Bureaucrats of competent authorities to establish agreement frameworks. Support staffs such as CAD designers and computer system engineers in the ITER joint central team. Staff members who provide assistance for a pleasant life for visiting researchers and their families. People who support the international school for the children. Citizens and staff members of local governmental organizations who accept foreigners enthusiastically. The number of people is far greater than that in past international collaborations with Japan.

Before the ITER project, each party made use of almost all collaborations as a means for attaining their own research programs or objectives. In contrast, in the ITER project, all parties have fully cooperated to produce the project objectives and approaches, which greatly differs from the former collaborations. Hence, the mutual understanding and consensus of the participating research scientists and engineers have required much more depth than previously. In addition, the government staffs supporting the project have made discussions and negotiated with a strict attitude. Furthermore, the participants in the ITER project have made exchanges with site workers and Japanese people through their social, cultural, and economic activities as a result of living in Japan, and this has increased the depth of mutual understanding at their personal level. If ITER is constructed in Japan, exchanges of this type will be expanded in scale, and the role of the exchange people will be expected to contribute much the mutual understanding in fields outside of science and technology.

(3) Peaceful debate, a method of reconciliation with neighboring countries and the world

It is necessary to consider the significance of the Japanese contribution to the world community in future international collaborations from the viewpoint of human history. For several decades after World War II, or even back to the Meiji Restoration, Japan has achieved economic and social advances using the slogan “catch up and get ahead of advanced countries.” Now our nation has reached the point where the significance of Japanese research and development can be discussed from the position as a leading contributor to the world community.

Japan could make a leading contribution to the welfare of all humankind through fusion development. This, however, does not imply that our nation would obtain superior experience in fusion technology. Because of its universality in science and technology and its objective to contribute to the humankind, Japan would obtain effective experience in debating and also in negotiating in other fields. This method of reconciliation is far from a military or economic one. It is completely a peaceful way. Being the lead contribution to ITER is an appropriate means for our nation to explore the prospects to attain world peace.

Japan’s neighboring countries are recently succeeding in rapid economic development, or at least have developed that potential. For the moment, they are not in a position to perform fusion research or to rapidly accept its benefits. However, when our nation proceeds further with fusion development, the expected scenario is that neighboring countries can profit from the long-term prospects. Japan can then establish a system whereby our neighbors can participate in the development. Our cooperation in ITER could give us the opportunity to consider our national role among advanced countries, and our lead role in ITER offers the possibility of reducing the gap between the “have” and the “have not” countries.

2.2.4 Value to Japan of hosting ITER

In this report, the value of constructing ITER in Japan has been already described. In this section, on the assumption of ITER construction in Japan, we describe the value and key issues to invite ITER to Japan.

2.2.4.1 The international collaboration for the ITER construction phase and a principle of equality

At the initiation of the international collaboration to design the ITER experimental reactor, “a principle of equity” was employed as follows: (1) Equal provision of resources (manpower, goods, and real estate). (2) Equal cost sharing. (3) Equal contribution (design, manufacture, installation, testing, construction, and operation). (4) Equal distribution of the results.

In this international ITER/EDA collaboration, equal contribution has been attained. Equal sharing of the results has also been attained by sharing the information of the results obtained. On the other hand, in the construction phase the participant parties would build a single experimental reactor in collaboration. Hence, it would be an unrealistic scenario that the opportunities and amount of ITER construction could be distributed with complete equality and then that the parties would accept the results equally. The construction of the only one device would inevitably imply an inequality. At this point, all parties seem to have the view that "the principle of equality" is inappropriate from the point of cost sharing. With this condition, if each party expects to build the next reactor, DEMO, in their own country following the ITER experimental reactor phase, it is considered that the country building DEMO will seek the opportunity (apart from the quantity) to manufacture the major equipment indispensable to obtain the key technologies.

The basic philosophy to invite (host) ITER and its merits and demerits are described in the following section.

2.2.4.2 The merits and demerits of inviting ITER to our nation

(1)The basic philosophy to invite ITER to our nation

The ITER program offers important milestones in world fusion research and development, to attain a self-ignition condition and a long burn, and to establish the basis of fusion technology necessary for a fusion DEMO reactor. It is understood this goal is common for all participant parties.

Therefore, a nation considering inviting the program to site the ITER in its country should not only take notice of the merits as being the host, but also should show that the feasibility of achieving the mission of ITER would increase with the acceptance of the invitation. For this reason, if Japan hosts ITER, it should promote ITER steadily, with a strong determination and sense of responsibility. In addition, Japan would be required to express its clear resolve to the other parties to make the maximum effort to construct ITER.

Our nation is poor in domestic energy resources, and should become independent of international energy supplies. Doing so would contribute toward the establishment of world energy security. Since the necessity for fusion energy development in Japan is keener than that in the EU or Russia, our nation should be a leading position to promote fusion energy research and development. Japan, an advanced country in the field of fusion energy development, could be a leading country on the basis of high technology and various human resources that can create and use technology. It is noted that diversified industries exist that have manufacturing and construction experience of the device including large-scale experimental facilities. This signifies that Japan has a developed industrial foundation sufficient to host ITER.

After declaration of its candidacy for hosting ITER, Japan should show its attitude and enthusiasm with firm responsibility to promote the project, and should provide proper infrastructure for advanced research facilities to the participants from the other parties. In addition, the host should make an effort to consider their smooth access to various items of information, and to give wide opportunities to join research.

(2)Total promotion of the international project, system integration technology, experiences of construction, operation, maintenance, and management

Japan, as the host, would obtain experience in taking the leading role in the total promotion of the international project. There would be merits for many researchers and engineers to have opportunities to learn system integration technology, and to accumulate construction, operation, maintenance, and management experience in ITER. This is a crucial element to enhance the technological potential of industries dealing with manufacture, operation, maintenance, and management. The Japanese attitude toward contribution to the international project would be expressed worldwide if Japan becomes a hosting country. The domestic infrastructure built in this project could be reused, and would be great advantage for promoting the phased development of fusion energy in Japan.

In research and development with a big project, the following steps are generally taken: A conceptual study is first performed on the line of extrapolation from the current state of advancement. Second, considering the entire project coordinates the design and R&D. Third, optimized facilities are constructed. Fourth, total system integration tests are conducted to confirm the specified system performance. The system integration functions and roles through these procedures are very important. This is because an activity in the system integration process requires all the important information such as the performance confirmation results (this is a key in the project) to overcome technical problems and issues. It is also because this activity requires decision-making. Accumulated technology with the combination of human expertise and technical information will play an important role in the future developments and project advances. If the success of the project is required, and if the technology in the experiment is important for the future fusion reactor, then qualified personnel should be positioned in the organization in charge of system integration to assume a significant role in the project. This will allow the technology to be advanced most effectively, and this trains a promoter for a future project. If the ITER would be constructed at a domestic site in Japan, more Japanese engineers than those from other parties could work in construction, and they could observe and experience the process of system integration. The chance to accumulate technology would also be increased.

Our nation as the host would make every effort to support the ITER project to achieve the mission within the pre-determined budget and period. For example, if an opposition campaign by the local or national populace against ITER would occur in Japan, and if this would be a serious obstacle to the ITER project, Japan and the other parties would suffer damage. Therefore, it is of great importance to build a consensus of national opinion for hosting ITER.

(3)Further public understanding and acceptance of fusion energy development

There exist 51 operating nuclear (fission) power plants in our nation. The public has a deep understanding of the current status. However, since fusion energy research and development is being conducted by several laboratories and universities at this point, the public understanding of the achieved level of research and its significance is not sufficient enough.

If our nation hosts the international project of ITER construction, public interest and understanding of fusion energy research and development will greatly increase. If the device is constructed in our country and if international researchers use it in the frame of the international collaborative project, Japan could show that it is playing a major role for a significant contribution to the world. In addition, this might stimulate the public to understand the fact that our nation considers fusion power an important energy source and has been promoting it. This also should have the positive effect of deepening the public understanding of Japan’s national nuclear power development and utilization program.

However, recent nuclear accidents and scandals, such as the dishonest altering of the test data, etc., are now eroding the public trust in the nuclear power to a large extent. To rebuild the public trust requires the guarantee that nuclear power is a product based on a "trustworthy technology," one that has reliability--and above all, safety.

Consequently, to invite ITER to our nation and promote it smoothly, it is important to carefully explain the following points to the people in various fields and classes, and to achieve a consensus on the invitation of ITER. Moreover, in the phases of invitation and construction, as well as in the long phase of operation, it is essential to promote the activities of research and development with a deep public understanding of the safety by providing the information to the public.

(i) Handling of radioactive isotopes and the environmental impact

The ITER site would contain several kilograms of tritium that would be imported from abroad. Until now, only a total of 60 g of tritium have been utilized in Japan for research and development of fusion fuel cycle systems, and the level of the research attained is nearly the best in the world. In fact, a model loop that simulates the ITER fuel cycle system exists only in Japan. About 100 g of tritium will circulate in the ITER plant, which can be sufficiently assessed on the basis of the technology and experience already present in our nation. However, the public does not have proper knowledge of this high research status. It is necessary to disseminate the ITER safety measures to the public for an understanding and acceptance of tritium handling.

The structural materials in a fusion reactor, irradiated by intense neutrons from the DT fusion reaction, would produce approximately 40 k-ton of activated waster materials at decommissioning. This requires us to safely manage the radioactive waste continuously for a certain period of time after the ITER project. The methods of safekeeping and disposal for the activated materials are similar to those for low-level radioactive waste from fission reactors whose technology is established already. It is necessary to explain how to guarantee this safety to obtain public understanding.

(ii) Influence on the surrounding region during the entire period of ITER construction, operation, experiments, and decommissioning

It is important to explain what experiments are being performed, and what influence ITER might have on the surrounding region during the entire period of ITER, from the start of construction to the completion of decommissioning, to gain public understanding and acceptance. It is also necessary to assess and explain the environmental influence of the heat exhaust and tritium release from ITER on the surrounding ecological system for the period of normal ITER operation.

(iii) Public understanding of ITER safety

The ITER construction will be conducted on the basis of ITER safety licensing examination, in which the hypothetical severe accidents are to be discussed prior to the approval of construction. It is necessary to explain in detail how to maintain ITER safety during the entire period to gain public understanding. After operation begins, it is necessary to maintain safety and to provide the confidence in the reliability of ITER safety for the members of the public that work or reside near the site. This should be accomplished through rapid announcements that provide the precise information.

(iv) Peculiarity as a project of research and development

Since ITER is the first large fusion reactor in the world, yet has the character of an experimental device, it is different that such a plant operate in a fixed scenario. This point should be explained and understood by the public. It is important to explain to the public for their understanding that ITER may have small unpredictable troubles within the bounds of safe operation, and that the plans and schedule of the ITER experiment and operation may change.

(4)Establishment of ITER regulations and licensing processes, and its influence on the cost

The licensing to construct the ITER experimental reactor will be to obey the regulations of the host party. If Japan hosts ITER; the safety regulations in our nation will be applied to ITER. Therefore, the standards and regulations that apply to earthquake-proof designs will likely be stricter than those in other nations will. If this yields to increase the construction cost, then this will be a demerit in expenditures.

However, the regulation, organization, and licensing process would be established more quickly than in other nations because of Japan's experience in these matters. If a fusion reactor for the development phase that follows ITER is built in another nation, the regulation and licensing process standards for other nations will be made using our ITER experience as their reference. This would be merit in favor of hosting ITER. Legislation of safety regulations and the licensing process for a fusion reactor in Japan will lead to a corresponding design standard. This standard would be essentially equivalent to the ASME code that was established for the siting process of a nuclear power plant in the United States. This implies that if Japan hosts ITER, the standards for the siting of a fusion reactor in our nation could be applicable elsewhere in the world. This is not only a contribution to the world, but also to our domestic industries, because the industries in our nation could take advantage of world trade for the export of future technology to build a fusion reactor.

(5)Cost share

In the SWG (Special Working Group) report submitted at the ITER Council meeting held in January 2000, the following statements should be noted. “The common area of construction (defined as items such as superconducting magnets, that could be produced in any of the Parties and transported to the site), which is estimated at about three-quarters of the total capital cost, should be shared among the Parties in a way which is as balanced as possible.” And “The Host Party should bear the remainder of the capital cost. In addition, site preparations to satisfy the “ITER Site Requirements” will be undertaken, in principle, by the Host as its cost.” Therefore, our nation, in hosting ITER, would bear the largest portion of the total cost. Since ITER is the core device in the Third Basic Program for fusion research and development of Japan, the success in the ITER project for a long period of construction and operation would be most important for the smooth promotion of fusion development in the future in Japan. On the other hand, the following concerns possibly exist: Does the amount of shared cost influence the development of other science and technology in Japan? Would concentration of research and development budget on ITER cause an unfavorable influence on wide varieties of creative research by a reduction of the cost-effective experiments? Consequently, it is strongly recommended that, in addition to the budget for the appropriate share cost to the ITER project as the host, the budget for other science research and the other domestic fusion research (excluding ITER) should be taken into appropriate.

(6)Effect of promotion of the site-regional development

In relation to ITER construction, improvement of social infrastructure near the site, procurement of building materials from local companies, enhancement of employment opportunities, etc., will possibly be made. Furthermore, in the phases of construction, operation, and decommissioning, researchers, engineers, construction workers, and their families from all parties will live in the area surrounding the ITER site for a long period. The accommodations for living and the facilities for education will be substantially improved. Consumption of living goods will be increased near the site area, and intercultural communication will stimulate the regional people to learn the global way of thinking (global common sense). These effects of promotion of the site-regional development are expected to be far-reaching. The international collaborative project would make a favorable associated image with the site region as a world research center of advanced technology.

As mentioned above, hosting ITER has various merits and demerits. The overall evaluation on hosting ITER concludes that it is significant for Japan as a host nation to construct ITER and to make great contribution to the world community toward realization of fusion energy.

References:

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2.2.5 Tokamak Research in support of ITER

2.2.5.1 Placement of complementary and advanced R&D

The necessity of advanced and complementary research is clearly identified in the Third Phase Basic Program of Fusion Research and Development (Atomic Energy Commission, 1992: the Third Program is used in this section unless otherwise noted), which establishes, guides, and regulates nuclear fusion research and development in Japan (Fig. 2.1.1-2). In following this program hereafter, advanced and complementary research should be conducted and support ITER in JAERI and the universities as the Third Program states that "In addition to the experimental reactor development activities, research and development on tokamaks should be undertaken as complementary efforts in the areas of fusion plasma study that cannot be covered by the experimental reactor. Furthermore, the advanced research and development to confirm and demonstrate new fusion plasma technologies before employing them into the core device in each phase, including the experimental reactor, should be conducted."

In addition to burning plasma research, fusion plasma research towards steady-state operation of a fusion reactor is increasingly of importance in the ITER program. This turns the design concept to the compact ITER, as mentioned in Section 2.2.1. Since the Steady State Tokamak Reactor (SSTR) is a most promising DEMO reactor concept, it is placed as an extension of scientific and technological bases to be established in ITER, since the advanced and complementary research in support of ITER are closely related to R&D issues for a DEMO reactor [2.2.5-1]. Therefore, it should be possible to conduct the advanced tokamak research to aim at steady-state operation of a fusion reactor and to make much effort for production and long-time control of dense and highly radiative divertor plasmas to be combined with high-beta and high-confinement plasmas [2.2.5-2].

On the other hand, many domestic and foreign devices for plasma experiments that have a medium or small size have played an important role in the history of tokamak plasma research as a tractor to yield many results associated with improvements in plasma performance, such as the discovery of improved confinement modes like the H-mode and experiments for proof of principle in radio-frequency current drive studies. Therefore, it is important to implement not only the complementary research directly related with an ITER plasma, but also the advanced plasma research from multi-perspective views with maximum utilization of highly adaptable devices of a medium or small size. In addition, these pioneering and germinating research activities should expand the range of fusion research and be absolutely vital from the point of view of completing the basis of ITER program.

2.2.5.2 Status of fusion research in support of ITER

1) Domestic status and research issues

JT-60 has continued to play a great role in determining the compact ITER policy. It has shown world leadership together with creative results of fusion plasma research for a steady-state fusion reactor such as proof of principle of the steady-state tokamak concept and production of the world highest plasma performance with an advanced operation [2.2.5-1]. As compared with the plasma conditions in ITER and SSTR, however, the issues on improvement in beta, density, and heat load controllability onto the divertor remain unsettled, and that is why the present tokamak experiments are pursuing the further improvement in coherent achievements of each performance (Fig. 2.2.5-1).

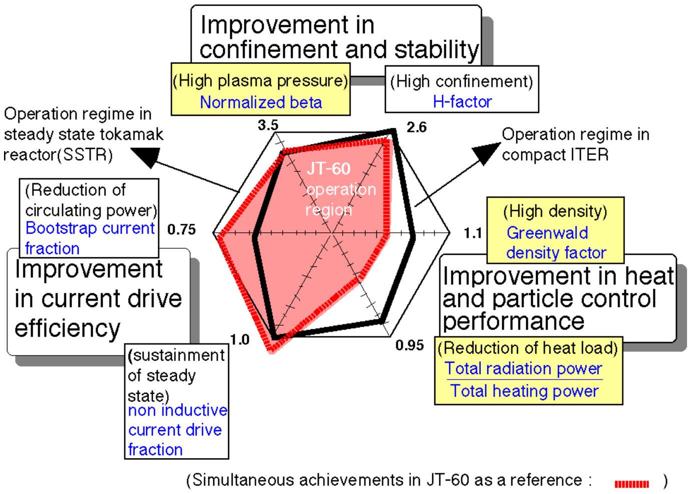

Fig. 2.2.5-1 Required enhancement of the major parameters achieved simultaneously in plasma

JFT-2M is a flexible medium-size tokamak addressing pioneering research issues. In these efforts, proof-of-principle experiments for advanced low-activation ferritic steel (which is the most promising candidate for structural materials for a fusion reactor) have been started (see Section 3.1). Promotion of these advanced material plasma test experiments has been placed as an important contribution for the fusion reactor structural material development authorized by the Planning and Promotion Subcommittee (Fusion Reactor Structural Materials Development Working Group) in the Fusion Council.

In universities, small tokamaks such as TORIUT, TNT, HYBTOK, NOVA, and OT were constructed and operated in 1970s, and pioneering and productive research with respect to plasma heating and control have been promoted. In particular, studies on the MHD characteristics in the past years have been applied to the present tokamaks. It should be emphasized that these activities also produced talented persons with research expertise. The JIPPT-II/T-IIU (Japan), operated for many years as a medium-size tokamak, promoted pioneering work in the fields of confinement evaluation for high-power RF heating, RF heating and current drive using a variety of methods, and study of confinement mechanism using sophisticated diagnostics. It also should be mentioned that JIPPT-II/T-IIU was in the forefront of the research on the relation between current profile control and plasma confinement characteristics. Experimental results from it pointed out the importance of a reversed magnetic shear profile before this research became a central issue in the present tokamak research.

Non-inductive current drive is necessary for steady-state operation of a tokamak. Using the WT-2/WT-3 tokamaks (Japan), researchers started RF current drive research in the early stages and demonstrated the lower hybrid wave (LHW) current drive. Following that, researchers using these tokamaks also succeeded in identifying current ramp-up by using only RF waves and current drive by using electron cyclotron waves, both of which played a leading role in current drive research [2.2.5-4].

The research for steady-state operation of a tokamak has been taken over by a super-conducting tokamak TRIAM-1M (Japan). Using this tokamak, researchers obtained an ultra-long-pulse discharge exceeding 2 hours using LHW (see Fig. 3.1.5-3). This result is an epoch-making achievement in comparison with the several tens of seconds at most obtained for operation of the present tokamaks and is still much longer than the several hundred seconds for a nominal operation in ITER. As an improvement for such a long pulse operation, the importance of the plasma-wall interaction and advances in the data acquisition system were identified, which greatly contributed to programs with such an ultra-long-pulse operation, such as ITER. Recently achieving the results of high ion temperature (several keV), negative shear current profile, high-density current-drive experiments, etc., the TRIAM-1M is continuing to play a role in advanced tokamak research by supplementing the work done in large tokamak devices in these research fields [2.2.5-5].

2) Present status of advanced tokamak research and future prospects

According to papers presented at the IAEA international conference in Yokohama held in October 1998, the present tokamak activities show that the research on steady-state operation of tokamak including long pulse

Table 2.2.5-1 Status of Advanced Tokamak Research

activity ● > ○

	Research subjects	Normal-conducting devices							Super-conducting devices	
		JT-60U (Japan)	JFT-2M (Japan)	JET (EU)	DIII-D (US)	Alcator C-MOD (US)	ASDEX-U (Germany)	TEXTOR-94 (Germany)	Tore Supra (France)	TRIAM-1M (Japan)
High performance	Confinement with high density	●	●	●	●	●	●	●		
	Internal transport barriers	●		●	●	○	●			●
	Edge transport barriers	●	●	●	●	●				
	Stabilization of NTM	●	○	●	●		●			
	Stabilization of RWM	○			●					
	Stabilization of TAE mode	●		●						
Steady state	Radiative cooling	●	●	●	●	●	●	●		
	Impurity control	●	●	●	●	●	●			
	Helium exhaust	●		○	○		○			
	Non-inductive current drive	●	○		●	●	○			●
	High bootstrap-current	●		○	●	○	○			
	Current profile control	●	○	○	●	○	○		●	●
	Long-pulse heat load								●	
	Long-pulse particle control									●
	Fueling		●				●			

(1998 IAEA Fusion Energy Conference)

operation is not yet sufficient while high performance research is widely carried out around confinement improvement in many normal conducting tokamaks [2.2.5-1].

In Fig. 2.2.5-2, the plasma size and shape are compared among ITER and major tokamaks (built or planned). Advanced R&D utilizing a full superconducting tokamak equipped with non-circular plasma shaping control is now necessary since the present superconducting tokamaks (TRIAM-1M, Tore Supra) adopt superconductors only for toroidal magnetic field coils and/or a circular plasma cross section. Currently, the only construction projects of tokamaks of a medium or large size that are authorized at government level are KSTAR in Korea (major radius of 1.8 m) and HT-7U in China (major radius of 1.7 m). The plasma performance in these tokamaks is limited at a low level (less than 0.1 of an equivalent energy multiplication factor) though both tokamaks adopt fully superconducting coils. The Snowmass meeting held in July 1999 in the US reported that JET was considering an extension of operation beyond 1999 and was investigating a modification program to enhance alpha heating with additional heating power ($Q_{DT} < 2$).

During the ITER construction phase of 10 years, it is crucially important to implement R&D to contribute to the construction of subsystems, such as the divertor configuration, and to be necessary for ITER operation (long time plasma control technology). Namely, improvement in divertor design, orientation for optimization and improvement in a variety of operation modes, simulation of ITER operation schemes including remote participation in experiments, and so on, should directly contribute to ITER. Implementation of experiments adopting ITER-relevant plasma shaping and divertor configuration during the ITER construction phase is necessary. Moreover, it is important to implement complementary research for ITER high-beta operation required for a DEMO reactor, the long time control at high beta, and to steadily proceed with R&D. From the point of view of the extension of research in support of ITER, it is important that studies of control methods for fueling, recycling, current drive, current density profile, plasma shape, etc., be continued in universities, institutes, etc. This would extend the range of fusion research and be useful for attracting young and energetic researchers.

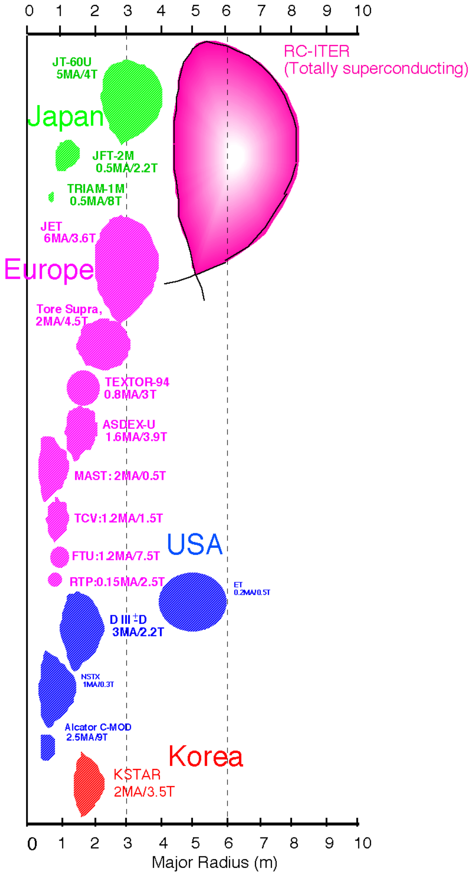

Fig. 2.2.5-2 Comparison of plasma sizes and shapes in ITER and other major tokamaks

2.2.5.3 Important research issues in the future

1) Long time control of high-performance and high-beta plasmas

The normalized beta value (β_N) necessary for ITER is ~2 for inductive operation and ~3 for steady-state operation (full current drive). To realize these values, ITER will adopt a plasma shape with a high elongation (κ~1.7) and a high triangularity (δ~0.35) to improve both confinement and stability performance. So far, quasi-steady operation around β_N~3 has been achieved in JT-60U and other tokamaks. Now, the demonstration of long pulse operation of plasma with excellent control capability for plasma shaping, which increasingly becomes important for ITER, is important to develop the steady-state research for high-performance plasmas toward improvement in the operational margin of ITER.

Moreover, normalized beta values (β_N~3.5-4) higher than the target of ITER are required for DEMO and economical reactors. Therefore, it is necessary to develop the technical basis for high-beta plasma control by introducing advanced R&D such as feedback control in combination with close conducting wall and high-beta stabilizing coils, plasma rotation control with beam injection, local RF current drive, and so on.

2) Long time control of high-density and highly radiative plasmas

JT-60 has demonstrated effective helium exhaust performance and a suppression effect for carbon impurity generation as a dome effect by means of the W-shape pumped divertor as a pumped divertor for a low triangularity configuration corresponding to the design of the ITER-FDR. However, no experiments have been yet carried out to demonstrate the capability of the advanced divertor designed for the ITER plasma configuration. Therefore, it is considered important to further improve the divertor design in ITER by developing divertor radiative cooling technology compatible with high confinement, with a pumped divertor consistent with the high elongation and triangularity plasma shaping required for ITER, with effective helium exhaust, and considering the baffle plate structure for suppression of back flow neutrals.

To realize steady-state operation (a bootstrap current fraction of 50-60%), which is considered important for ITER, the production of full current drive plasma with a high bootstrap current fraction and its long time control are important issues. While a high performance reversed shear plasma was sustained in a quasi-steady state with 80% bootstrap current fraction for the first time in JT-60, the Greenwald density factor and divertor radiation ratio in the discharge were lower than those values required in a steady-state scenario in ITER. The steady-state operation scheme for ITER and a DEMO reactor should be established by implementing long time control of the full current drive plasma, consistent with the reactor plasma conditions, by improving the Greenwald density factor and divertor radiation ratio, and by introducing non-circular plasma control and divertor control technology that will be adopted in ITER.

2.2.5.4 Future prospects

For extension of the ITER support research, it is rational to bring the most advanced domestic fusion infrastructure into maximum utilization. It is important to contribute to the optimization of ITER operation through the high controllability of plasma shaping and the high capability of steady-state operation and the heat and particle control by advanced divertor design and long time control. This could be done by introducing technologies and functions, which are being adopted for ITER, and putting much emphasis upon R&D to simulate the ITER operation.

International sharing of advanced and complementary R&D should be emphasized. The burning plasma research for Q~1 plasmas using tritium is left for JET, etc.

There is a great significance in the continuation of tokamak research during the ITER construction phase, not only to maintain research levels and to develop technical skills, but also from the standpoint of producing talented persons for the operation of ITER. In particular, this should be highly attractive for leading experimentalists and theoreticians to take the leadership role of R&D for ITER.

In the above roles, the roles of plasma physics as a common basis and the promotion of talented persons are also commonly important for the fields of non-tokamak confinement systems described in 4.3, so that such roles can generally play an important role in fusion research in support of ITER.

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2.3 The fusion demonstration reactor from ITER

2.3.1 Confinement scheme of the fusion demonstration reactor

In “ Promotion of Fusion Research and Development,” which the Fusion Council recommended when the Atomic Energy Commission was drawing up “ Third Phase Basic Program of Fusion Research and Development (Third Phase Basic Program)” in 1992, it is noted that adopting a tokamak device as a core device in the third phase is appropriate. Also, with respect to the choice of a confinement scheme in the following phase (the fusion demonstration reactor the DEMO reactor), the recommendation noted that the final decision of the choice should be reached from a comprehensive evaluation that includes results from the development of confinement schemes other than the tokamak.

As mentioned above, the confinement scheme of the DEMO reactor is not yet determined. However, the development of the tokamak-type the DEMO reactor would be promoted if the R&D proceeds steadily in the ITER tokamak reactor, if the tokamak configuration for the DEMO reactor is recognized as outstanding in the comprehensive evaluation with non-tokamak confinement schemes, and if the development of the fusion demonstration reactor is judged to be worthy of promotion from the point of view of the energy development strategy.

In the following paragraphs, development scenarios are described for promoting the development of the tokamak fusion demonstration reactor the DEMO reactor after ITER. In the case where a non-tokamak confinement scheme or an inertial confinement scheme shows characteristics superior to the tokamak, the usefulness of the knowledge from R&D in ITER is described.

2.3.2 Tokamak fusion demonstration reactor as an extension of ITER

The specifications for the tokamak fusion demonstration reactor the DEMO reactor to be developed as an extension of ITER depends on results of R&D in ITER (see Section 2.2.2). On the basis of the physics and engineering knowledge to be gained in ITER, a fusion reactor with so called "pulsed operation" can be designed even if the full current drive operation is not accomplished in the tokamak. For example, the design of a pulsed operation reactor, based on the same physics basis as the ITER design, shows that a tokamak fusion demonstration reactor capable of quasi-steady operation for several to ten hours with an operational intermission of about 10 minutes is possible and that a power plant can technically be realized [2.3.2-1]. However, such a reactor with quasi-steady operation becomes comparatively large, which is disadvantage to the power plant from the viewpoint of economics [2.3.2-2].

If steady-state operation technology, a high-temperature blanket, and so on, are developed in ITER in preparation for the DEMO reactor --and if neutron irradiation tests for materials are performed in parallel with the ITER program and are completed, the construction of a tokamak fusion demonstration reactor based on steady-state operation becomes plausible. The technical feasibility of fusion energy can then be demonstrated through the construction and operation of the DEMO reactor. Thus, the DEMO reactor would be a prototype of a practical fusion power plant and would complete the R&D phase of fusion power in Japan.

While the fusion plasma technology mentioned Section 3.1 should be strongly pursed to develop a tokamak fusion power reactor, the most important physics parameter is the normalized beta value, denoted by β_N. A high normalized beta exceeding the physics basis of ITER is a prerequisite for the DEMO reactor. If such a high β_N is achieved, it then becomes easy to increase a bootstrap current fraction so that a high-efficiency steady-state reactor with a low circulating power rate, as typical of SSTR [2.3.2-3], can be realized. In addition, if keeping a beta value close to the ideal beta limit becomes possible, the possibility of a highly economic power plant typical of CREST [2.3.2-4] is increased.

Reactor engineering technology needed to progress from ITER to the DEMO reactor includes the development of a power generation blanket and advanced low-activation materials, and the improvement in experimental reactor technology (superconducting coils capable of ~16 T, fast remote maintenance technology, advancement in safety technology). In particular, in terms of integration of the power generation system into the DEMO reactor, the choice of the test module type and the test results are important. In ITER, more than a few high temperature blanket modules (about 1m x 2 m) will be tested. Since the DEMO reactor whose size is same as SSTR needs a total of about 400 modules, performance testing of the blanket modules in the ITER test bed is essential for the demonstration of power generation in the DEMO reactor. Moreover, as pointed out in Section 1.3.5, performance improvements needed for a commercial power plant are the development of technologies to realize the reduction of start-up power and circulating power (a low-loss motor generator, an increase in operation temperature using high temperature superconductors (for example, 20 K)) and efforts to optimize equipment design and selection for economic improvement.

A list of the main performance improvements required to progress from ITER to the DEMO reactor (SSTR as an example) is shown in Table 2.3.2-1. Steady-state operation is a leading candidate as an operational scheme for the tokamak demonstration reactor. While a demonstration of high energy multiplication factor Q (exceeding 20) with inductive operation in ITER is important from the view point of understanding and control of the burning plasma, the demonstration of Q=5 with steady-state operation is also important in the sense of setting the scene for the operation scheme in a tokamak demonstration reactor. The Fusion Reactor Structural Materials Development Working Group, coordinated by the Planning and Promotion Subcommittee under the Fusion Council, discussed the use of low-activation ferritic steel, which is one type of ferritic steel like SS316, for the blanket structural materials. Vanadium alloys and SiC/SiC composite materials are also being considered as structural materials. Neutron fluence is an important factor for a functional test of a test blanket module in ITER. By using ITER, behavior evaluation and integrated functional tests for functional and structural materials of blanket are carried out in a low neutron fluence regime, because neutron fluence in ITER is limited in the range of 0.3 MWa/m^2 by the amount of tritium supply from outside of ITER facility. These blanket tests make it possible to get a prospect toward the DEMO reactor, in which the value of neutron fluence reaches several MWa/m^2. If these R&D studies successfully allow progress toward a the DEMO reactor, then we can proceed to the development of the DEMO reactor, of which energy flow and plant arrangement are expected to be those illustrated in Fig. 2.3.2-1.

Table 2.3.2-1 ITER to the DEMO reactor performance improvements

Item	ITER	1the DEMO reactor
Energy multiplication factor (inductive)	10 – 20	
Energy multiplication factor (steady state)	5	30 – 50
Plasma pressure	Several atm	~10 atm
Maximum magnetic field	12 T	16 T
Normalized beta	~ 2.5	~ 3.5
Blanket	Test module	Electric generation blanket
Structure material	SS316	Low-activation ferritic steel, etc.
Neutron fluence	0.3 MWa/m2	<10 MWa/m^2

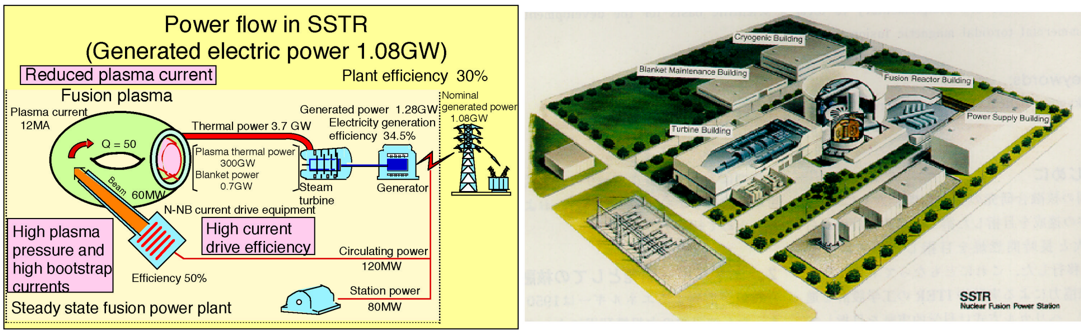

Fig. 2.3.2-1 Example of energy flow and plant layout in a highly-efficient, steady-state fusion reactor

References

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2.3.3 Usefulness of ITER if a non-tokamak confinement scheme for the DEMO reactor is selected

Even if a plasma confinement system for the demonstration reactor was selected to be a non-tokamak magnetic confinement system or an inertial confinement system such as laser fusion as discussed in Section 4, the R&D for ITER would be useful from a variety of viewpoints.

First of all, since the most engineering technology (superconducting coils, vacuum vessel, blanket, vacuum technology, heating equipment, tritium equipment, and so on) for magnetic confinement systems, like a helical device, have common elements and systems, the results from R&D in ITER can mostly be used to advantage. In addition, an experimental reactor plays the role as a test bed for the development of the power generation blanket for, as mentioned in Section 2.2.1. Thus, even if a concept different from the tokamak system were chosen for the demonstration reactor, the test results would significantly contribute to fusion reactor development. In terms of fusion plasmas, there is wide commonality in the behaviors of burning plasmas, so the knowledge gained from ITER would be extremely useful for another magnetic confinement system.

If the concept for a demonstration reactor were shifted to an inertial fusion system, the availability of the scientific knowledge would be reduced as compared with magnetic confinement systems, such as a helical system. However, the knowledge in tritium handling, safety technology, electricity generation technology, and so on, would be applicable.

2.4 Summary - Placement of ITER in development strategy

In this chapter, the development strategy towards the realization of fusion energy based on ITER has been described. As an approach to the realization of fusion energy, it is important to consider the desired specifications for a commercial reactor in addition to enhancing the advantages of fusion energy.

If one classifies the process of realizing fusion energy into a phase that requires “R&D” and another phase that makes it practicable and utilizable, the demonstration of steady-state electric power generation in the DEMO reactor as the first fusion power plant must be a major goal in the R&D phase. As an intermediate step to reach that goal, it is necessary to construct an experimental reactor and integrate DT burning plasma control and reactor technology (except blanket technology for power generation) into it. The current ITER design satisfies the requirements for an experimental reactor identified in the Third Phase Basic Program, in which long burns including self-ignition (the energy multiplication factor exceeds 20) and steady-state operation are all made possible.

While a variety of design studies on the DEMO reactor based on tokamak concept are considered as an extension of ITER, it is concluded by considering a representative example of the steady-state tokamak fusion reactor design that the prospect for the DEMO reactor having steady-state power generation can be developed by performing research for the DEMO reactor in ITER and other devices.

The tokamak based system for fusion energy development has the leading potential as the most advanced confinement system at the present stage, and thus is concluded to be capable of demonstrating burning plasma control and producing a large amount of fusion energy. However, to create a practical fusion reactor, it is necessary to improve the physics understanding and reactor technology on which the ITER is based. On that point, there is an area where advanced (alternative) concepts can be influential in the future, and the advanced and complementary research aiming at conceptual improvements in tokamaks have great significance.

The ITER is significant because of its place as an experimental reactor in Japan’s “Third Phase Basic Program” and as an international project implemented under international collaboration that most advanced nations are joining. Indeed there are a variety of merits and demerits to hosting ITER in Japan. All things considered, however, it is an immensely significant event that Japan may have the opportunity to host and construct ITER and thus provides an outstanding international contribution toward the realization of fusion energy.

Chapter 3 Technical Issues and Future Prospects for the Development of Fusion Energy using the Tokamak Device

This Chapter describes the present status and future prospects of the physical and technical issues for the development of the fusion reactor, aiming to address the "Technical Feasibility of Fusion Energy," which has been requested in the interim report of the Special Committee on ITER Project and charged to this Subcommittee of the Fusion Council for Fusion Development Strategy for investigation and review.

The technology issues necessary for the development of the tokamak fusion reactor are categorized into the following 5 areas. Namely, "Fusion Plasma Technology," where the fusion reaction is of main concern, "Reactor Technology," such as the vacuum vessel, superconducting coils, heating and current drive methods, "Blanket and Material Technology," which is directly relevant the power generation, amongst the engineering issues, "Safety Technology" for development of a safe fusion reactor, and finally the "Operation and Maintenance Technology" of the fusion plant. The following subsections are devoted to discussions of above issues. A step-by-step advancement in research and development in technology lead by the core device in each step should be performed for fusion reactor development, as described in the previous chapter. While the required technological standards will likely become higher and higher in each step, the major part of fusion plasma technology, reactor technology, safety technology, and operation and maintenance technology will be established by ITER and relevant R&D.

In addition, "Technical Issues for Production at Industries" and "Issues Relevant to the “Market Competiveness," which would be a major issue of concern in the commercial reactor phase, are discussed in Sections 3.6 and 3.7, respectively. Section 3.8 describes the technical feasibility of fusion energy, considering all these aspects.

3.1 Present status and future issues for the fusion plasma technology in tokamaks

3.1.1 Progress in confinement performance of tokamak plasmas

The tokamak fusion research and development was initiated in the former Soviet Union in 1960s and its superior confinement performance was pervasively recognized throughout the world in the 1970s. Japan has also been involved in the extended research and development of tokamak physics since 1974 with the JFT-2 facility. Substantial progress was made, as a result of competition in plasma performance, between the US and the Soviet Union in the 1970s, and the US and European countries in the 1980s. Accordingly, many medium- to large-size tokamaks were constructed. In the1990s, three large tokamaks, namely JET in the EU, TFTR in the US, and JT-60 in Japan, competed with each other for the world-record performance. Progress in plasma confinement during this period is shown in the so-called Lawson diagram in Fig. 3.1.1-1, where the product of central plasma density and confinement time is plotted against the central temperature.

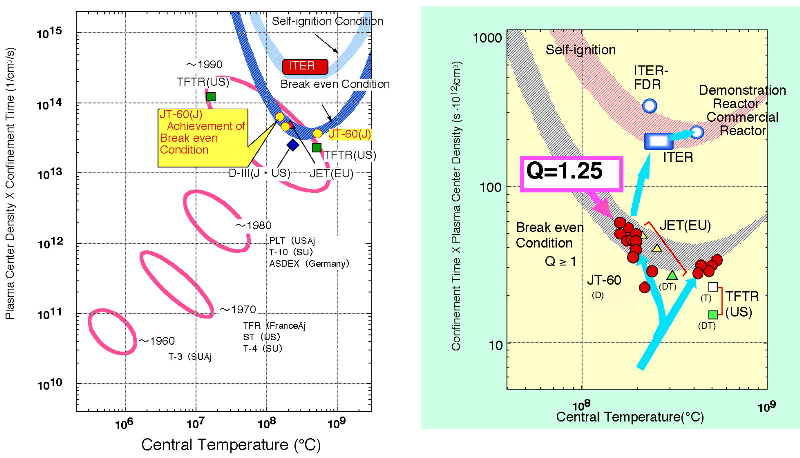

Fig. 3.1.1-1 Left hand figure is a Lawson diagram of improvement in confinement erformance of tokamak plasma, where the product of central plasma density and confinement time is plotted against the central temperature. Right hand figureshowstheprogressintheperformance in three large tokamaks, namely, JT-60, JET, and TFTR, as well as the target parameter range of ITER and DEMO reactor / Commercial reactor.

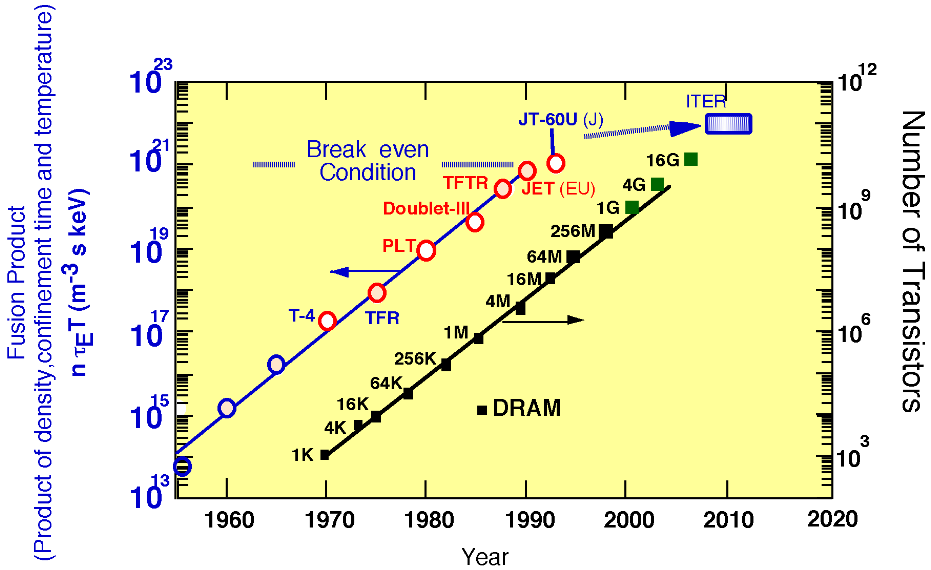

Fig. 3.1.1-2 Chronological progress in Fusion triple product, which is defined as the product of plasma density, confinement time, and temperature, compared with chronological progress in the capacity of integrated circuits (DRAM).

Chronological progress in plasma confinement performance for the 40 years that followed the 1960s is shown in terms of the "Fusion Triple Product," which is a product of density, confinement time, and temperature, in Fig. 3.1.1-2. The Fusion triple product increased one order of magnitude every five years, and approached a value close to the reactor domain. The rapid progress depicted here is comparable to the rate of increase in the number of transistors in an integrated circuit, which is representative of innovative technologies that were introduced after World War II.

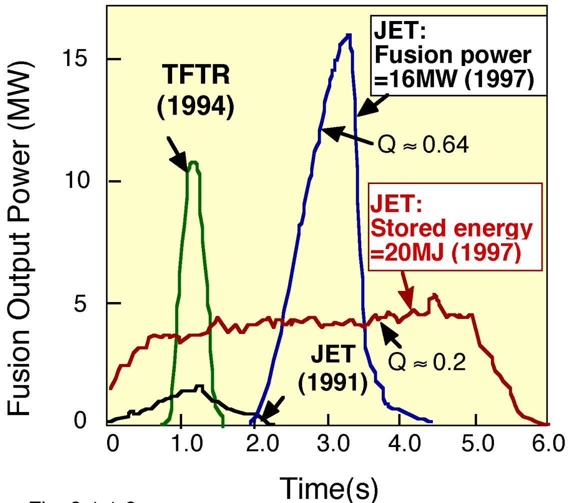

Fig. 3.1.1-3
Waveforms of fusion power output in TFTR and JET.

Moreover, TFTR and JET have demonstrated DT fusion burning, and both devices made great success in producing a fusion power that exceeded 10 MW [3.1.1-1, 3.1.1-2]. The characteristics of burning plasma were investigated with alpha particles produced in the DT plasma experiment. Figure 3.1.1-3 shows the waveforms of fusion power output in TFTR and JET.

Plasma performance and the initial DT burning experiments have progressed successfully as described above, and it can be concluded that we have reached the point where plasma confinement performance necessary for a fusion power reactor can be attained in the next-step device, which is the experimental reactor.

References

[3.1.1-1] K.M., McGuire et al., Fusion Energy 1996 (Proc. 16th Int. Conf. Montreal, 1996), Vol.1, IAEA, Vienna (1997) 19.

[3.1.1-2] The JET Team, Nuclear Fusion vol.39(1999)1227.

3.1.2 Progress in research and development in JT-60

The JT-60 (JAERI Tokamak 60) promoted as a Japanese national program was approved on July 31, 1975 by the Atomic Energy Commission. The construction of JT-60 started in 1978 and accomplished with installation of the heating devices in 1981.

The fusion research and development in JT-60, conducted for more than 14 years since the first plasma was produced in 1985, is aimed at the attainment of the breakeven condition and fusion plasma performance improvement. During this period, JT-60 attained many outstanding results and led the world in fusion research. The plasma performance reached the target regime of the breakeven condition in 1987, the world record of bootstrap current fraction of ~80% was achieved in 1989, the world's record for ion temperature was accomplished in 1992 and further increased in 1996, and a world-record non-inductive current of 3.6 MA was attained in 1993. Furthermore, injection of the negative-ion-based neutral beam was performed for the first time in the world in 1996, the DT equivalent breakeven condition in reversed shear plasma was achieved, following JET in 1998, the world's record for energy multiplication factor (Q) of 1.25 with a reversed shear plasma was accomplished in 1998, and the demonstration of the steady-state operation scenario with reversed shear plasma was performed in 1999, all of which are shown in Fig. 3.1.2-1. JT-60 took a longer time than expected to attain the DT equivalent breakeven condition, which was a major mission of the JT-60 program. The initial experimental results at JT-60 did not reach those originally anticipated despite its unique characteristics, such as the divertor, metal first wall, etc., which were not provided in other large tokamaks, i.e., TFTR and JET. However, after the high-current modification of the device that implemented the most advanced knowledge obtained to date in the complementary devices, such as the DIII program (Japan-US collaboration), JFT-2M, and other tokamaks in the US and European countries, JT-60 established its world leading status in fusion performance. It should also be noted here that an intensive effort has been devoted to the optimization of plasma performance. To lead the scientific program to ultimate success, it is prerequisite to allow robust and flexible engineering design in the development of experimental facilities so the necessary counteractions could be easily taken against unexpected obstacles. In the case of JT-60, large toroidal coils enabled the modification of the tokamak to accommodate higher-volume noncircular plasmas, and resulting modification lead to significant progress in plasma performance. At the time JT-60 was modified, more than ten tokamaks existed in the world, including DIII and JFT-2M, and the experimental results obtained from these tokamaks were applied to the JT-60 modification.

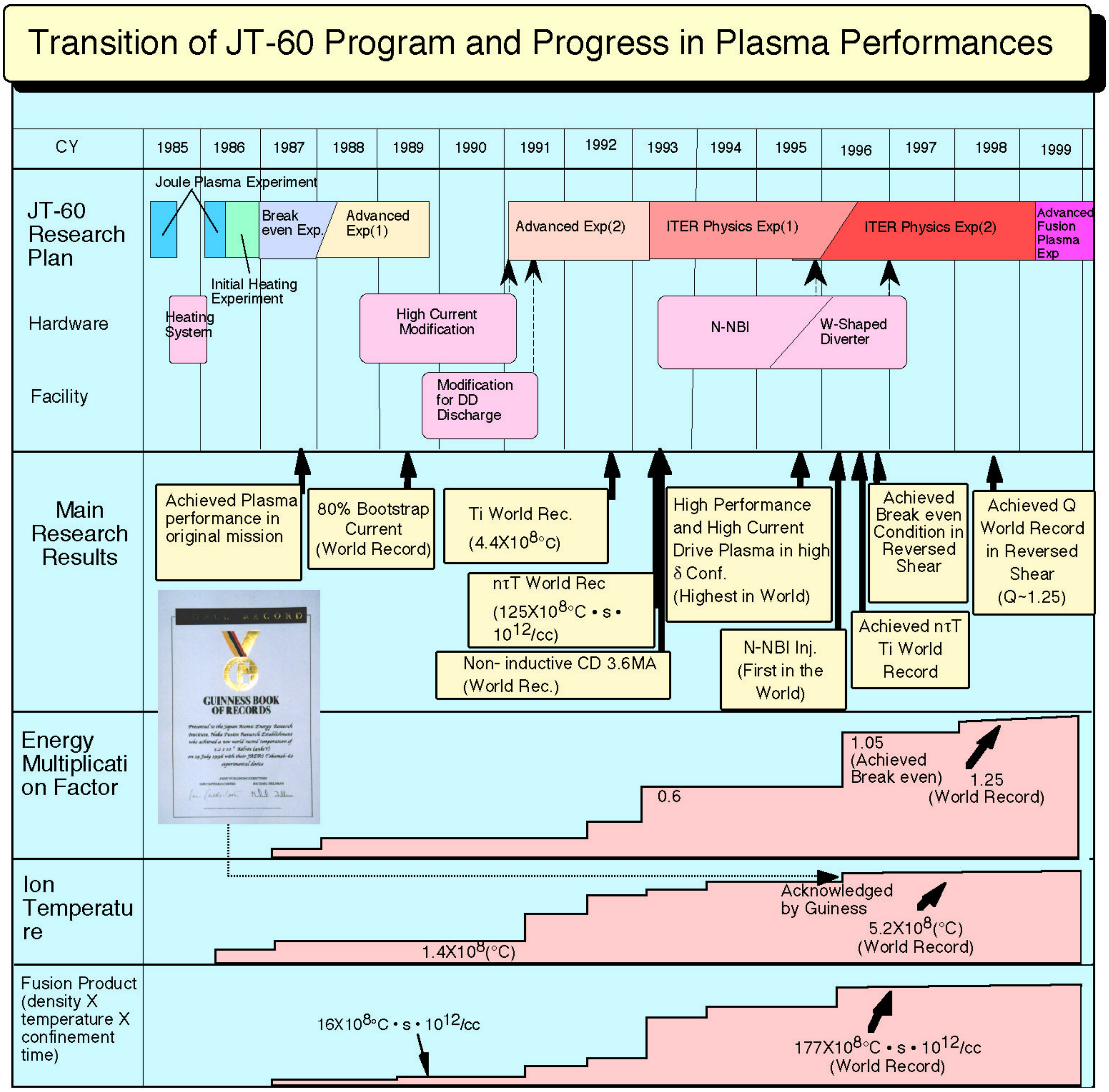

Fig. 3.1.2-1 Summary of JT-60 research results since production of its first plasma

3.1.3 Required advancement in fusion plasma parameters and development issues in the future [3.1-3.1]

As an example, the required progress in main parameters and future development issues of core devices designed for each phase are quantitatively shown in Fig. 3.1.1-3, and future development issues are described in this subsection. The parameters of three large tokamaks and DIII-D, representing medium-size devices, are shown here. Also shown are the design parameters of a medium-aspect-ratio option of the compact ITER, which are presented as representative of the experimental reactor, while the SSTR and CREST for the demonstration reactors and A-SSTR and CREST for the commercial reactors, respectively are also included.

(1) Fusion output power: A maximum fusion output power of 16 MW was attained in JET. After a fusion output power of nearly 0.5 GW is realized, the necessary fusion output power in the demonstration reactor phase would be increased to 3 GW level.

(2) Energy multiplication factor: An energy multiplication factor of 1.25 has been achieved in JT-60 and that of 1.14 in JET. In an experimental reactor (ITER), an energy multiplication factor of more than 20 would be achieved in the inductive operation, whereas a value of more than 5 is aimed at during steady-state operation. Furthermore, an energy multiplication factor of around 30 or more is necessary to be realized in the steady-state demonstration reactor. A confinement state, the so-called L mode, is insufficient for this purpose and a confinement improvement two or three times higher than the L mode is required. The H mode, which was discovered experimentally in ASDEX in 1980s, represents the improved confinement modes. The input power for plasma transition to the H mode needs to exceed some threshold power, referred to as the L-H transition threshold power (Section 3.1.4). The principal device parameters of the experimental reactor or the following reactor are determined by the H-mode confinement time scaling (Section 3.1.4). A confinement improvement higher than the H mode (Section 3.1.4) is desirable in order to reduce the reactor size and realize steady-state operation.

(3) Pulse duration: The present large tokamaks employ normal conducting coils to perform a pulsed operation of around 10 seconds. The pulse duration needs to be extended to around 1,000 seconds in the experimental reactor, and one day to a few months operation is indispensable in the demonstration reactor. Furthermore, high reliability of the device is required in a commercial reactor for continuos operation, except for periodic inspection/maintenance periods. For this purpose, it is important to develop technology for controlling a long burn plasma, the duration of which exceeds the current diffusion time, and to increase the operation margin for the beta limit by optimizing the magnetohydrodynamic stability by utilizing the current profile control scheme. It is also important to establish a method of avoiding disruptions or softening the effect of disruptions, even when a disruption occurs accidentally (Section 3.1.5).

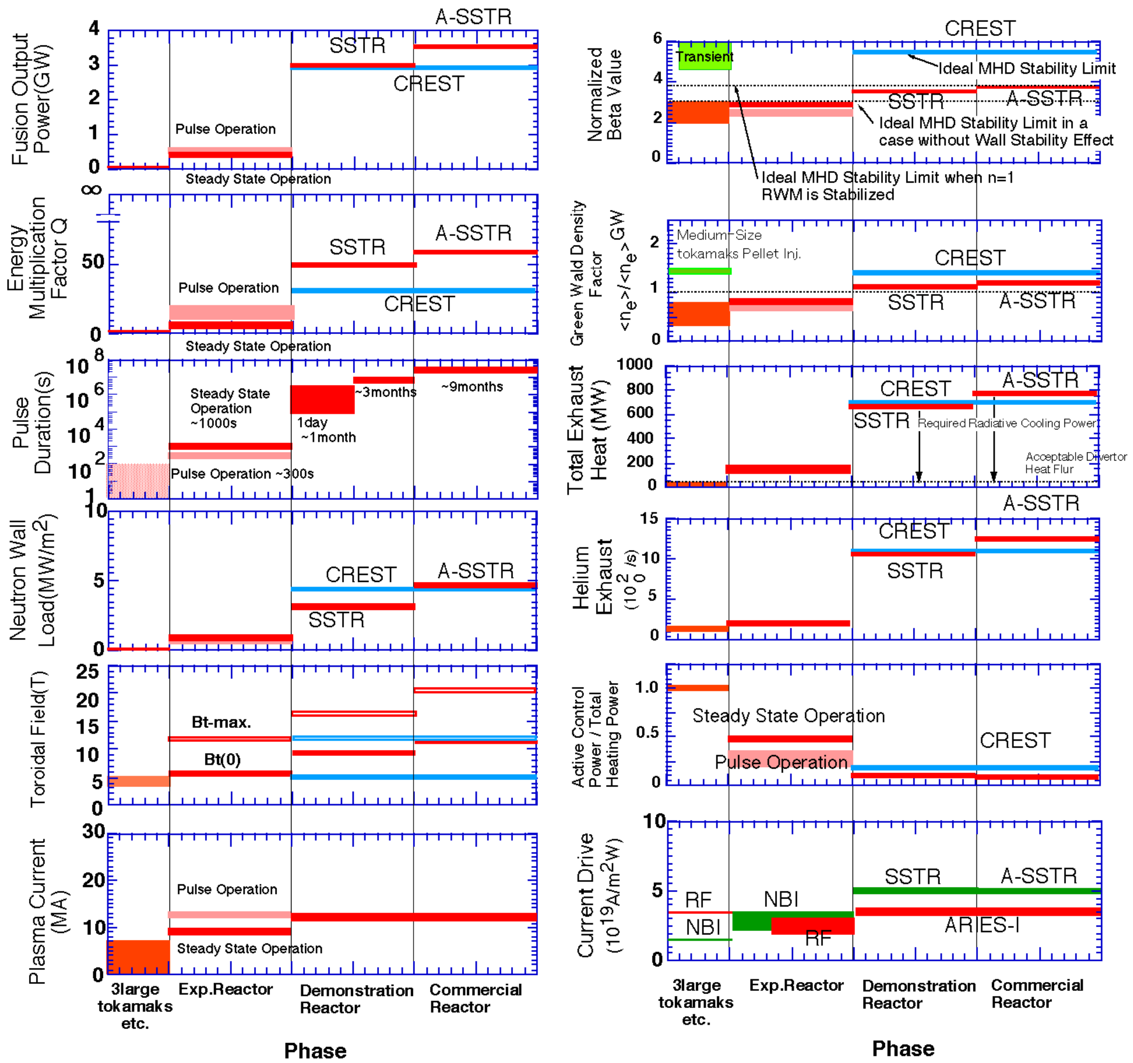

Fig. 3.1.3-1 Advancement in main parameters in each phase

(4) Neutron wall load: Although the neutron wall load is 0.5 MW/m^2 in an experimental reactor, high fusion output density, which produces a neutron load as high as 3-5 MW/m^2, is necessary in a demonstration reactor (DEMO) to demonstrate the economic aspects of the following commercial reactor. From a plasma performance point of view, high plasma pressure must be realized by increasing the beta limit to obtain such a high neutron load.

(5) Toroidal magnetic field: The toroidal magnetic field of JT-60 is 4 T and that of an experimental reactor such as ITER is about 5.5 T. The high toroidal magnetic field is not required, provided the high beta value (β) as assumed in the CREST design can be achieved in a demonstration reactor (DEMO). However, a high-normalized beta (β_N) of nearly 5.5 has not been attained in any experiment so far. Therefore, .a higher toroidal magnetic field is necessary in a demonstration reactor (9 T in SSTR, 16 T at the maximum) in order to achieve a high fusion output density. The increase of L-H transition power due to higher field and the degradation of confinement characteristics due to the reduction of ρ^*_i / L are anticipated (Section 3.1.4).

(6) Plasma current: A lower plasma current of 10-14 MA is needed to operate the demonstration reactor in a high-efficiency steady-state operation with a high bootstrap current fraction. On the other hand, plasma current should be as high as possible to produce the high-Q plasma, because the confinement time is proportional to the plasma current.

(7) Normalized beta: The normalized beta sustainable in a quasi-steady-state in JT-60 is around 2.5. This value is sufficient for the inductive operation in an experimental reactor. However, a higher normalized beta must be realized in a demonstration reactor. Normalized beta is limited at around 3 when wall stabilization is not relied upon. Even when wall stabilization is effective, normalized beta stays at around 3.5-4 in the case where only a resistive wall mode with the toroidal mode number n=1 is stabilized. If the resistive wall modes with all toroidal mode numbers are stabilized; the normalized beta can increase in principle to the stability limit against ideal MHD modes. However, optimization of the plasma current profile and pressure profile as well as the stabilization of the neoclassical tearing mode are necessary (Section 3.1.4).

(8) Operation density: In the present tokamak experiments, high confinement is realized at a density lower than about 80% of the Greenwald density (60% of the Greenwald density in the case of JT-60) suggested by Greenwald. Although the medium-size devices have shown that a high-density operation can be realized at a density higher than the Greenwald density by ice pellet injection, demonstration of the high-density operation in large tokamaks is still an issue of concern (Section 3.1.4). The experimental reactor ITER has been designed to operate under the Greenwald density. However, high-density operation exceeding the Greenwald density is desired in and after the demonstration reactor (DEMO) phase to increase fusion output.

(9) Heat exhaust and radiative cooling: The maximum plasma heating power in three large tokamaks is 40 MW in JT-60. The total plasma heating power increases to 200 MW (alpha-heating power of 100 MW and external-heating power of 100 MW) in ITER. If all of these output power flows into the divertor, the heat flux onto the divertor plate would become too high to be removed continuously for 300-500 seconds. To significantly reduce the high-heat flux, production of the low-temperature and high-density divertor plasma (about 80% of heat radiation) is necessary. For and after the demonstration reactor phase, the required performance for the low-temperature and high-density divertor plasma becomes more demanding, as the exhaust heat must be increased to 600-800 MW (Section 3.1.5).

(10) Helium exhaust: The alpha particles (3.5 MeV helium) produced by the DT fusion reaction must be exhausted from the fusion plasma as ash after thermalization. The helium quantity almost similar to that generated in ITER (1.5×10^{20} 1/s) has been successfully exhausted in steady-state in JT-60. In ITER, helium exhaust capability in a steady-state has to be demonstrated. On the other hand, a helium exhaust rate six to seven times higher than ITER must be realized for and after the demonstration reactor phase (Section 3.1.5).

(11) Self-heating fraction and heating control: A qualitative jump exists for extrapolation between the self-heating by DT burning, which we have not yet experienced, and the present experiment. In a step from the experimental reactor to the demonstration reactor, the fusion output and energy multiplication factor increases significantly. To retain the prediction capability for these situations, technology developments for

reducing the heat load and controlling the plasma with small external heating power would become key issues.

(12) Current drive efficiency: In a steady-state tokamak reactor, a high bootstrap current fraction is realized and the remaining plasma current is driven with a noninductive current drive method. In this scenario, high current-drive efficiency is required, in order to decrease the circulation power in a fusion power plant (Section 3.1.5). A high current-drive efficiency of 0.5×10^{20} A/m^2W is expected with the 2 MeV beam for SSTR and a relatively low current-drive efficiency of 0.35×10^{20} A/m^2W is expected with radio-frequency current drive for ARIES in the US.

(13) Simultaneous attainment: All subjects on fusion plasma technology described above must be realized simultaneously in a fusion reactor. A main point of fusion plasma development is to realize integrated performance that meets all necessary performance requirements simultaneously. Namely, it is necessary to produce the required fusion output with high confinement, to maintain the heat flux onto the divertor plate below a maximum permissible flux, to improve the economic efficiency by providing a high- output power density and a small circulation power, and to drive the plasma current non-inductively. Furthermore, the burning rate, plasma stability, etc., would have to be controlled externally with small heating and current drive power, while maintaining the desired plasma condition. The fusion plasma of a steady-state reactor could be realized when all these requirements are satisfied simultaneously. Figure 3.1.3-2 shows the requirements in the achieved level of integrated performance in each phase, based on the steady-state reactor concept SSTR, designed by the Japan Atomic Energy Research Institute.

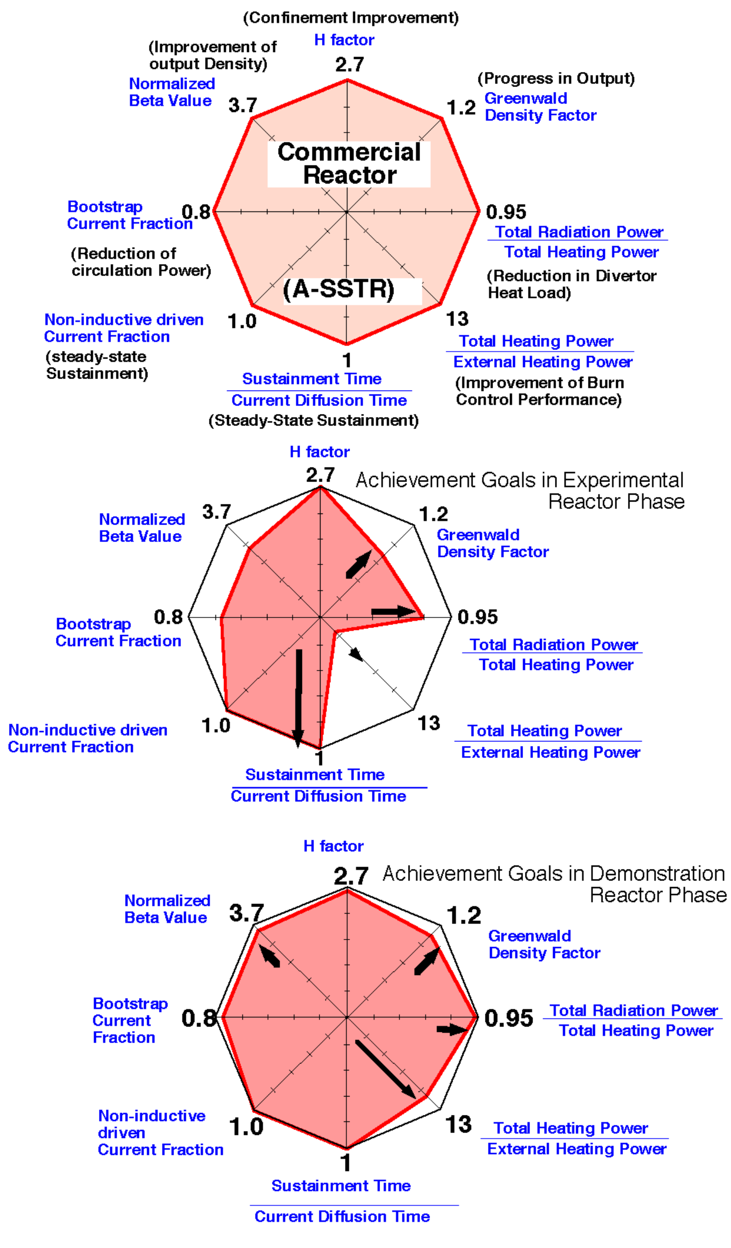

Fig. 3.1.3-2 Required integrated plasma performance in each phase, based on the steady-state reactor SSTR

References

[3.1.3-1] Y. Kamada, M. Kikuchi, H. Ninomiya, "Issues of Fusion Plasma Research towards Realization of Fusion Reactor," this sub-committee Document No. 5-4.

3.1.4 Issues and prospects for confinement performance

(1) Confinement Scaling

More than several seconds of confinement time of the thermal energy is necessary in ITER and relevant fusion reactors to achieve the reactor condition (in the self-ignition condition or Q larger than 10-20) in the Lawson diagram shown in Fig. 3.1.1-1. The prediction of this energy confinement time has been performed by using empirical scalings that were developed based on the experimental data compiled in various tokamaks worldwide. The scaling of the energy confinement time is intended to express the energy confinement time (= plasma stored energy / heating power) with engineering parameters, such as the plasma density. For plasma confinement quality, several confinement modes of operation have been discovered, such as the L mode (Low confinement mode) that exhibits degraded performance under auxiliary heating, the H mode (High confinement mode) that has about twice as much confinement time as the L mode [3.1.4-1], and other improved confinement modes (see Fig. 3.1.4-1).

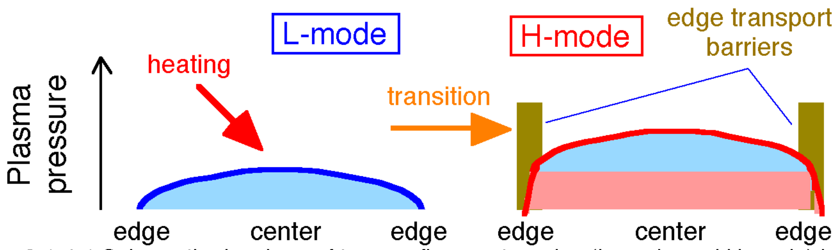

Fig. 3.1.4-1 Schematic drawings of two confinement modes (L mode and H mode) in tokamaks. When the plasma heating power exceeds a certain value (L-H transition threshold power), a transport barrier appears at the plasma periphery and the high confinement state (H mode) appears. Scaling laws of the confinement time have been developed for these two modes of confinement.

Scaling of L mode confinement time:

The ITER89-P law that scales the L mode confinement time was developed based on the confinement database accumulated during the ITER CDA activity [3.1.4-2].

$$\tau_E^{ITER89P} = 0.048\, M^{0.5}\, I_p^{0.85}\, B_t^{0.2}\, R^{1.2}\, a^{0.3}\, \kappa^{0.5}\, n_{20}^{0.1}\, P^{-0.5} \qquad (1\text{-}1)$$

In this expression, τ_E is in the units of s, M denotes the ion mass number, Ip is the plasma current in MA, and B_t represents the toroidal field at the plasma center in T, whereas the major and minor radii are expressed in R and a both are in m, κ stands for the ellipticity, n_{20} is the line-averaged density (n_{19} in $10^{19}\ m^{-3}$ and n_{20} in $10^{20}\ m^{-3}$), and the heating power P is in the units of MW. Here, the degradation of energy confinement time with the heating power is indicated as the $P^{-0.5}$ dependence.

The energy confinement time is a useful index to describe the heat transport property in a plasma interior, and it has a relation with the heat diffusion coefficient χ expressed as $\chi \sim a^2 / \tau_E$. When the heat diffusion coefficient is written in a form $\chi \sim \chi_{Bohm} (\rho^*)^{\mu} F(\beta, \nu^*)$, the index of ρ^* equal to zero, i.e., $\mu = 0$ is relevant to the Bohm type transport, whereas $\mu = 1$ is called as the gyro-Bohm type transport. Here, $\chi_{Bohm} \sim T / B_t$: Bohm diffusion coefficient, T : plasma temperature, e : elementary charge, $\rho^* \sim T^{0.5} / B_t R$: normalized Larmor radius, $\beta \sim nT / B_t^2$: beta value, $\nu^* \sim nR / T^2$: collisionality, and F : dimensionless function. In the L mode confinement, the heat diffusion coefficient corresponding to ITER-89P, Eq. 1.1, is nearly of the Bohm type.

Combining the ITER-89P law shown in Eq. 1.1 as the energy confinement scaling with the power-balance equation for a burning plasma, which includes the fusion-power multiplication factor Q, we can estimate the necessary condition, related to the confinement performance [3.1.4-3].

$$H\, I_p\, A\, 4\, C_f\, C_{eff}^{0.5} / (3 + C_f) \geq 90\, (1 + 5/Q)^{-0.5} \qquad (1\text{-}2)$$

Here, $H = \tau_E / \tau_E^{ITER89P}$ denotes the confinement improvement factor, $A = R / a$ is the aspect ratio, $C_f = 2 n_{D,T} / n_e$ represents the fuel dilution rate, while $C_{eff} = P / \{P_\alpha (1 + 5 / Q)\}$ is the effective heating rate. (P_α: total effective heating power with the radiation loss power subtracted, and P_α : alpha heating power). In order to increase the Q value in a fusion reactor with a reasonable value of the product of Ip and A, it is necessary to sustain a steady-state plasma with low dilution ($C_f \approx 1$) and low radiation ($C_{eff} \approx 1$), in addition to achieving an improvement in the confinement time (H > 1). From this point of view, the ELMy H-mode is considered to be appropriate as a standard mode of operation in ITER. The ELM is an unstable MHD mode localized near the plasma edge and breaks out intermittently in short period of time ($\tau_{period} << \tau_E$). In the Q = 10 operation of ITER, provided that the He content be as low as about 5% and other impurity contents are suppressed as well, i.e., $C_f = 0.8$ and $C_{eff} = 0.8$ are realized, Eq. 1-2 is reduced to H Ip A ≥ 90 MA. Therefore, the required improvement factor H in ITER is 1.94 for the present design parameters of A = 3.1 and Ip = 15 MA. Such values of H-factor have been obtained for ELMy H-mode plasmas in various tokamaks, and the possibility of achieving Q ≥ 10-20 in ITER seems to be realistic. Thus, on the basis of the above considerations, the ELMy H-mode would be a promising candidate for the standard operation mode in ITER.

ELMy-H mode scaling:

As discussed above, the improvement factor ($H = \tau_E / \tau_E^{ITER89P}$) required in ITER ELMy H-mode plasmas is approximately 2 or larger. Formulating the confinement scaling of the ELMy H-mode plasmas and the direct extrapolation for ITER has been undertaken. A large amount of ELMy H-mode confinement data have been compiled in various devices (Alcator C-Mod, ASDEX, ASDEX Upgrade, COMPASS-D, DIII-D, JET, JFT-2M, JT-60, PBX-M, PDX, TCV) since 1995 and merged in the international database to obtain the following scaling of $\tau_{E,\,th}$ (confinement time of thermal energy) in ELMy H-mode [3.1.4-1].

$$\tau_{E,th}^{IPB98(y,2)} = 0.0562\, M^{0.19}\, I_p^{0.93}\, B_t^{0.15}\, R^{1.39}\, a^{0.58}\, \kappa_a^{0.78}\, n_{19}^{0.41}\, P^{-0.69} \qquad (1\text{-}3)$$

In Fig. 3.1.4-2, the experimentally obtained data is compared with the ELMy-H mode confinement scaling.

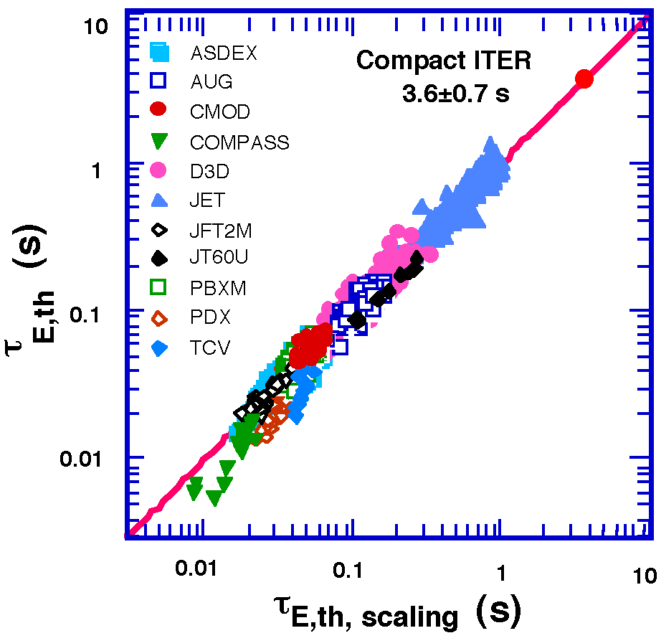

Fig. 3.1.4-2 Comparison of the ELMy-H mode confinement scaling law and the values obtained experimentally in various tokamaks. Here, the predicted value as well as the confidence interval is provided for ITER.

Although the non-dimensional constraint was not applied to derive the above scaling in the regression analysis, Eq. 1-3 satisfies the non-dimensional constraint, and it has the dimension of time. The transport corresponding to this scaling is close to the gyro-Bohm type, which is similar to the previous ELM-free H-mode scaling, and is consistent with the results of dimensionally similar experiments performed to date, i.e., χ / χ_{Bohm} $(\rho^*)^{0.7} \beta^{0.9} (\nu^*)^{0.01}$. If the ELMy-H mode confinement is indeed the gyro-Bohm type, the H-factor of the ITER plasma is expected to become larger than those of present tokamak plasmas, because the ρ^* in ITER plasmas is smaller than those of present devices. The predicted thermal energy confinement time for ITER from the above scaling is $\tau_{E, th}$ = 3.6 s. Here, The design parameters of ITER used for the prediction are R = 6.2 m, a = 2.0 m, κ_a = 1.75, Ip = 15 MA, B_t = 5.3 T, n_{19} = 10 x 10^{19} m^{-3}, and P = 90 MW. From the statistical analysis, the uncertainty in this prediction is estimated as 2.9 s < $\tau_{E, th}$ < 4.3 s. The design activity of ITER was performed by carefully considering such an uncertainty in the confinement performance.

An important issue of concern for the confinement scaling study is to improve the accuracy of extrapolation with the H-mode confinement scaling. Having in mind that the pedestal structure of the pressure profile is formed by the transport barrier at the plasma periphery, it can be more plausible to separate the plasma stored energy into two parts, namely, the pedestal component (offset part) and the core component on top of the pedestal component. The offset type scalings are thus derived under the above considerations. By evaluating the parametric dependencies of pedestal and core components, respectively, and also by clarifying the independence and interdependence between them, the prediction accuracy of the scaling would be improved. As the characteristics of the pedestal are dominated by the MHD stability, the pedestal component of the stored energy can be increased by changing the shape of plasma cross-section, which greatly affects the MHD stability. It has been confirmed in the experimental, theoretical, and numerical studies that the steepness and triangularity of the plasma cross-section are effective in stabilizing the MHD activities. In addition, the interdependence between the pedestal and core has been found; i.e., the core confinement can be improved by the improvement of the pedestal confinement [3.1.4-5]. Therefore, it is very important for the improvement of H-mode confinement to optimize the plasma shape. For the design of ITER, the poloidal coil system has the flexibility to be able to optimize the plasma shape. Therefore, it is possible that the confinement performance could be higher than the present H-mode scaling predictions.

Extrapolation to the demonstration reactor, based on the results of the ITER confinement demonstration:

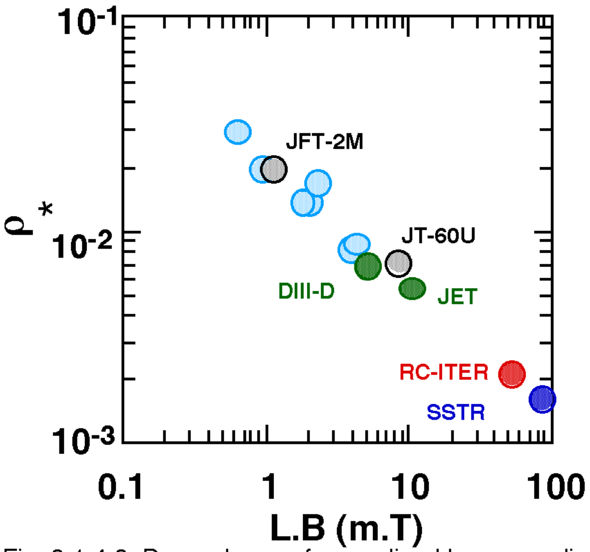

Fig. 3.1.4-3 Dependence of normalized Larmor radius ρ^* on the product of device size and magnetic field L·B

Figure 3.1.4-3 shows the values of ρ^* as a function of L·B for present devices and the experimental reactor ITER as well as the conceptual demonstration reactor SSTR, where L = $V^{1/3}$ denotes the device size, and B is the magnetic field strength. Here, the beta values are similar for all data points. This figure indicates that an extrapolation step from ITER to SSTR is much smaller than that from present devices to ITER. If the ITER performance were within the range of present prediction capabilities, it would be straightforward to assess the confinement properties of the demonstration reactor with significant accuracy by including the ITER results. In other words, extrapolation of the confinement performance to fusion reactors in the L-mode, H-mode and other regimes will be assured by the verification of the confinement performance in the ITER experimental reactor.

Confinement scaling for other improved confinement modes:

Other than the ELMy-H mode, various schemes of operation with improved confinement are presently suggested. Namely, they are the high-poloidal-beta ELMy-H mode, the reversed shear mode, the RI mode, etc. The confinement performances of these modes are expected to be higher than that of the ELMy-H mode. However, confinement scaling for these alternative modes has not been established yet and this remains a future issue of concern.

(2) L-H transition threshold power

To attain the ELMy-H mode, which is considered as a standard operating mode for improved confinement in ITER, the heating power has to be above the threshold, namely the L-H transition threshold power. The scaling of threshold power in terms of the magnetic field intensity, plasma density and shape has been derived by the analysis of the experimental database, which has been compiled from 10 existing tokamaks in the world, and is indicated in the figure below. The scaling obtained from the most recent 1999 database is written as follows [3.1.4-6.1].

$$P_{th} = 2.84\ M^{-1}\ B_t^{0.82}\ n_{20}^{0.58}\ R^{1.0}\ a^{0.81} \qquad (2\text{-}1)$$

Here, P_{th} represents the L-H transition threshold power in MW, and Bt, n_{20}, R, a, and M respectively denote the toroidal magnetic field [T], plasma density [10^{20} m^{-3}], major radius [m], minor radius [m], and the average mass number of the fuel working gas, viz., 2.5 for the DT mixture. Inverse dependence on mass was derived from the recent JET experiment, where different species of working gas, such as H, D, T and their mixtures, were used therein.

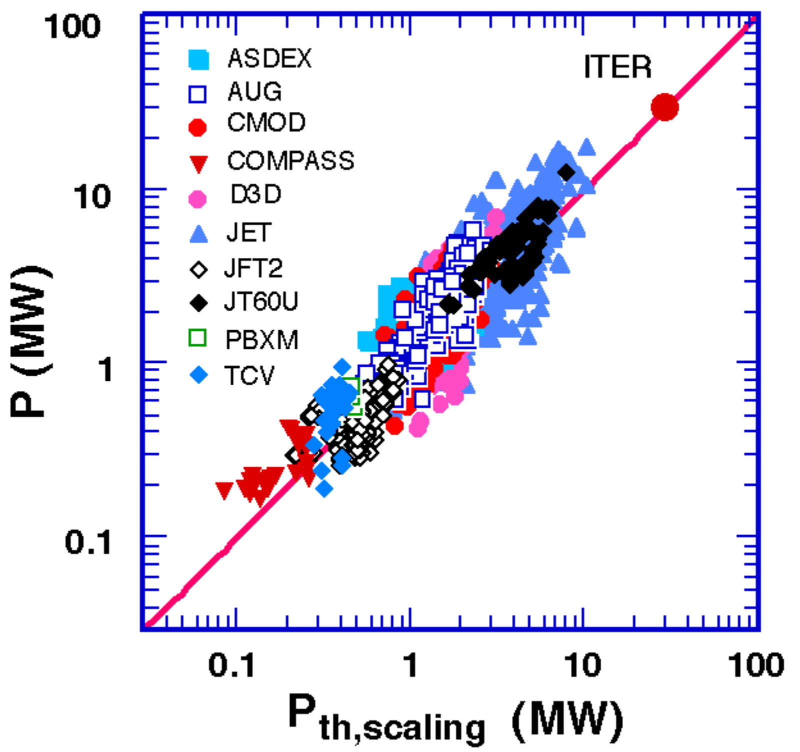

Fig. 3.1.4-4 Comparison between the L-H threshold power scaling and the experimental database, together with the ITER prediction

Based on the above scaling, the heating power necessary to attain the H-mode in ITER, of which the dimensions and operating parameters are R = 6.2 m, a = 2.0 m, Bt = 5.3 T, and n_{20} = 0.5 x 10^{20} m^{-3}, is anticipated to be approximately 30 MW. As the range of uncertainty is around 20 MW < P_{th} < 50 MW, the designed heating power of 50 MW for ITER would suffice for the L-H transition condition. For prediction to SSTR, which is a conceptual fusion reactor with R = 7 m, a = 1.8 m, Bt = 9 T, and n_{20} = 0.5 x 10^{20} m^{-3}, the L-H threshold power is expected to be around 50 MW. Since SSTR would have a capability of 80 MW input for the current drive and alpha heating in the L-mode, it is foreseen that transition to the H-mode would easily be attained. The demonstration of H-mode operation and confirmation of the threshold scaling in ITER would provide confidence for extrapolation to the demonstration reactor, which would actually generate fusion power.

The issue of present controversy, which is related to the L-H threshold power, is that the scatter in the database around the scaling is rather large compared with the confinement database, as shown in Fig. 3.1.4-4. Intensive investigations with experiments and modeling have been performed since the discovery of the H-mode in ASDEX. For example, experimental documentation of the radial electric field formation [3.1.4-9] by a fast potential profile measurement [3.1.4-10] was accomplished in JFT-2M. Accordingly, it has been pervasively recognized that the structure of radial electric field takes a fundamental role in the bifurcation or transition phenomena. Therefore, the penetration of the neutral particles and impurities and magnetic shear near the edge might cause the scatter. Detailed studies on the effect of neutral particle density in JT-60 showed that elimination of the influence of neutrals could reduce the scatter [3.1.4-11]. Another issue of concern is the compatibility of the edge and internal transport barrier, which is prerequisite for the advanced steady-state operation in ITER. It was found in a recent experiment that a threshold power also exists for the internal transport barrier formation. Therefore, it is necessary to quantitatively identify the conditions for which edge and internal transport barriers are simultaneously obtained to assure an adequate confinement margin in ITER.

(3) Improved confinement suitable for the steady-state operation

Several modes of improved confinement so far discovered are shown in Fig. 3.1.4-5. The ELMy-H mode, which is a standard operation mode for ITER, was already discussed in (1) and (2). In the core-improved mode, reduced radial transport is realized in the central region, while reduced transport is obtained in the peripheral region in the H mode. The core-improved mode with an H-mode edge is also observed in various tokamaks.

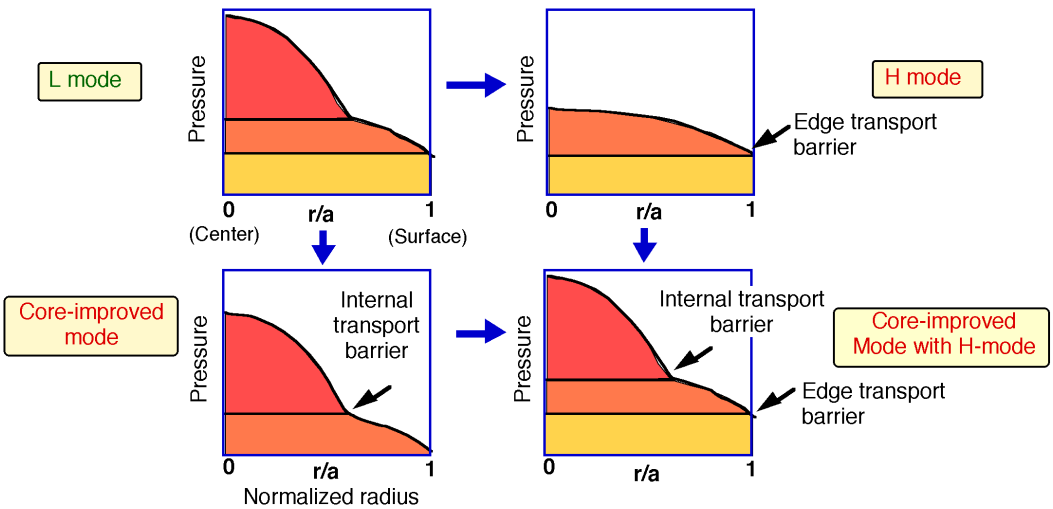

Fig. 3.1.4-5 Improved confinement modes

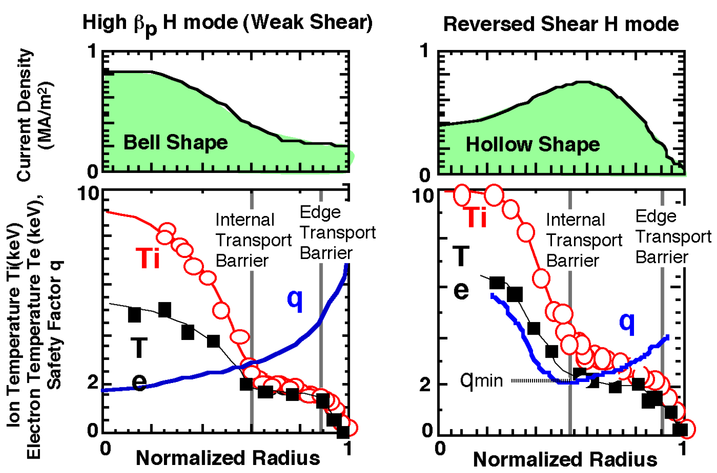

Fig. 3.1.4-6 High-βp mode (weak shear) and reversed-shear mode

Steep temperature and density gradients are formed in the plasma core by the transport reduction in the core-improved mode, which is as if the invisible barrier is suppressing the energy flow from the central region of the plasma. This is denominated as the internal transport barrier, ITB. In the core-improved mode with the H-mode edge, more prominent confinement improvement (3 to 4 times the L-mode confinement) and larger beta values are obtained than in the conventional H mode. Hence, the core-improved mode with the H-mode edge is considered to be a promising operation scenario for a steady-state tokamak reactor, which requires a high fraction of bootstrap current.

Two kinds of core-improved modes have been established in JT-60, a weak shear mode and a reversed shear mode. In both modes, the safety factor on the axis, q(0), is larger than unity, and the sawtooth oscillation is not excited. In the weak shear plasmas, the current density profile has a bell shape (see Fig. 3.1.4-6), and peaked-temperature and density profiles are formed by the intense heating and fueling near the plasma axis. The "high βp mode" [3.1.4-12] in JT-60 and "supershot" [3.1.1-1] in TFTR are typical of the weak shear plasmas. The "optimized shear" mode [3.1.4-13] in JET may be included in this category. The reversed shear mode is characterized by the hollow shape in the current density profile and is observed in TFTR [3.1.4-14], DIII-D [3.1.4-15], JT-60 [3.1.4-16], Tore-Supra [3.1.4-17], and many other tokamaks. Typical profiles of temperature, current density, and safety factor (q) in the high βp mode and the reversed shear mode in JT-60 are shown in Fig. 3.1.4-6. In the high βp mode, the safety factor, q, increases monotonically with the radius. In reversed plasmas, however, the minimum value (q_{min}) exists in the q profile and q decreases with the radius (the magnetic shear is negative) inside the position of q_{min}. In the high βp mode, the peaked-temperature and density profiles are often observed. In the reversed shear plasmas, on the other hand, the steep-temperature and density gradients are localized near the half radius, and flattened profiles are often observed near the plasma center. The confinement improvement is larger in the reversed shear mode than in the high βp mode. The transport is generally reduced to the neoclassical value in the reversed shear mode, while it is not as substantial in the high βp mode. In addition, the transport reduction is seen both for ions and electrons in the reversed shear mode. However, the transport reduction for electrons is sometimes unclear in the high βp mode. As to the stability performance or beta limit (see 3.1.4 (4)), the high βp mode is more advantageous.

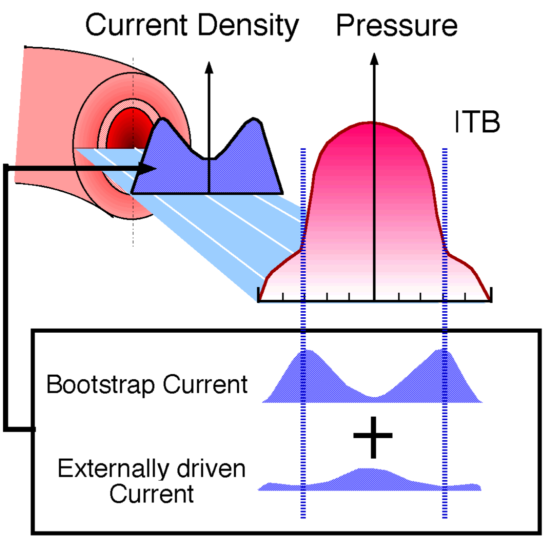

Fig. 3.1.4-7 Sustainment of hollow current density profile with the bootstrap current

In JT-60, the world-record ion temperature (45 keV) and fusion triple product (1.5×10^{21} $m^{-3}\cdot$keV·s) were achieved in the high-βp-H mode operation [3.1.4-18]. In the JT-60 reversed shear mode, the break-even condition [3.1.4-18], the world-record fusion gain (Q_{DT}^{eq} = 1.25), and a confinement improvement of twice that of the conventional H-mode were achieved. These results indicate that the core-improved mode is essential for high plasma performance.

The high bootstrap current fraction provides the sustainment of high q(0) (>1) in steady-state operation and the core-improved mode is readily obtained in the reversed shear configuration. On the other hand, the high-pressure gradient produced in the core is effective in increasing the bootstrap current fraction. Therefore, the core-improved mode is considered to be suitable to increase the high bootstrap current fraction that is required in steady-state tokamak reactors. The concept of steady-state operation with the reversed shear mode is illustrated in Fig. 3.1.4-7. The internal transport barrier is formed in the hollow current density profile and a large bootstrap current is generated in the high-pressure gradient region. In a steady-state reactor, high-beta and high-bootstrap current fractions (above 70%) are envisaged to reduce the current drive power. The total plasma current thereby naturally becomes hollow, and the hollow current profile and the internal transport barrier are sustained stationary state. In the JT-60 experiment, such a stationary state was sustained for 3 seconds.

If the core-improved mode is established in an experimental reactor, higher confinement would be obtained than with the ELMy-H mode and a higher Q value, close to ignition, could be anticipated. However, conditions or scaling of the threshold power to produce the core-improved mode have not yet been established. Therefore, it is not clear if the ITB could be obtained in the present ITER design. As a result, the improvement of the extrapolation capability for plasmas with ITB is an urgent issue of investigation.

In regard to the application of the core-improved mode to the steady-state operation in the experimental reactor and/or the high Q (high-bootstrap-current fraction) demonstration reactor, control of ITB is important. In addition, the evaluation of confinement characteristics in high electron heating and low fueling plasmas has to be investigated. The accumulation and exhaust of impurities, which will be discussed later, are also important issues of concern.

(4) β-limit and optimization of the MHD stability

In magnetic confinement systems, the plasma generally suffers from instability and confinement starts to degrade when the β-value exceeds a critical value. This limit is known as the β-limit. The β-value is defined as the ratio of the plasma pressure P to the magnetic pressure $B^2 / 2\mu_0$, and the critical value is a few percent of P / ($B^2 / 2\mu_0$). The fusion experimental reactor ITER and future DEMO reactors are designed with the β-values lower than the β-limit. To realize a compact and economical fusion reactor, a high plasma pressure is favorable. Since the achievable amplitude of the magnetic field pressure is technically limited, an increase in the β-value within the stability boundary. Therefore, a profound understanding of the parameter dependence of the β -limit is indispensable.

Based on the experimental and theoretical research in the 1980s, it was heuristically found that the achievable β-value in the tokamak is given by (see Fig. 3.1.4-8)

$$\langle\beta\rangle(\%) = \beta_N \, I_p(MA)/a(m) \, B_t(T) \qquad (3\text{-}1)$$

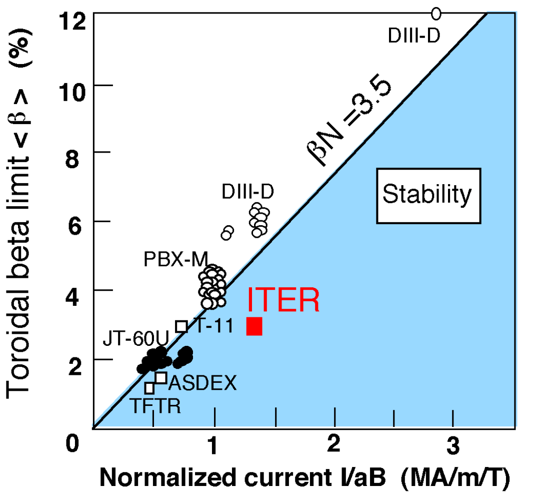

Fig. 3.1.4-8 Although βN ~3.5 is the standard limit, a higher value is achievable by optimization

where the coefficient b_N is denominated as the normalized β-value, which is an important measure of stability. It is known that the achievable β_N –value is typically 3.5. However, in some specific experiments, higher values of β_N have been reported. The fundamental elements that influence the instabilities governing the β_N-limit are 1) the current profile, 2) the pressure profile, 3) the plasma shape, 4) the stabilizing (conducting) wall and 5) the resistive instabilities.

1) Current profile

Theoretical and experimental investigations performed to date have confirmed that the MHD stability is strongly influenced by the current profile, e.g., reported from JIPP-T-II [3.1.4-19]). Experiments performed in DIII-D, JT-60, and TFTR demonstrated that the β_N-limit increases almost linearly with the internal inductance l_i (DIII-D achieved a β_N = 6 at l_i =2 [3.1.4-20]), where l_i is a measure of to what extent the plasma current profile is peaked at the plasma center. These are important initial experiments that demonstrate that high values of β_N are achievable by current profile control.

In the steady-state operation of the ITER and DEMO reactors, the current profile can be flexibly controlled by the combination of externally driven current and bootstrap current, which functions to flatten the current profile. In the course of the SSTR design studies, two types of advanced operation modes have been proposed. One is the weak magnetic shear mode [3.1.4-21], where a hollow current profile is avoided by the centrally peaked current drive with neutral beams. The other is the reversed shear mode with a hollow current profile [3.1.4-22]. In particular, after the theoretical verification of the stable solution for the reversed shear configuration, which had been thought to be unstable, designing a steady-state fusion reactor operating in the reversed shear mode has become popular (ARIES-RS in US and CREST in Japan). In the reversed shear mode, stabilization by the conducting wall plays an essential role for the achievement of the high value of β_N.

2) Pressure profile

Dependence of the β_N-limit on the pressure profile shape was experimentally resolved in JT-60. As shown in Fig. 3.1.4-9, the achievable β_N is limited by the instability at the edge (ELMs: Edge Localized Modes) when the plasma pressure profile is broad, which is relevant to the small values of p(0) / <p>. Here, p(0) is the pressure at the center and <p> represents the volume-averaged pressure. On the other hand, in case the pressure profile is centrally peaked (high values of p(0) / <p>), β_N is limited by the kink-ballooning modes in the central region. Thus, there exists an optimum pressure profile that maximizes the achievable β_N [3.1.4-23].

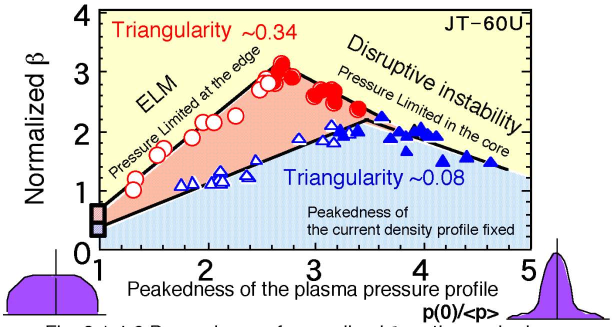

Fig. 3.1.4-9 Dependence of normalized β on the peakedness of the plasma pressure profile
The pressure profile is controlled by the external heating power profile.

In ITER and the demonstration reactor, the pressure profile is determined by the α-particle heating, and the peakedness of the pressure profile is expected to be rather high. The most appropriate pressure profile shape for steady-state operation would have to be investigated in ITER.

3) Plasma shape

Another essential factor that governs the stability is the shape of the plasma poloidal cross section. In particular, the β_N-limit increases with an increase in triangularity, according to the experiments in JT-60 and DIII-D (see Fig. 3.1.4-10). Based on these results, ITER was designed to enable a high triangularity, 0.35-0.4. In addition, it has been theoretically pointed out recently that the sharp bending of the triangle or the elliptization of the cross section can improve the β_N -limit. It is expected that the sustainable β_N -value would possibly be improved by the optimization of shaping in the manner mentioned above.

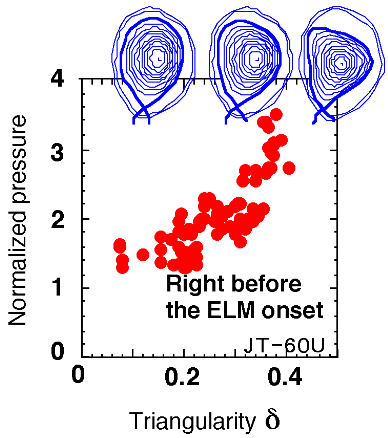

Fig. 3.1.4-10 Increasing normalized pressure gradient with triangularity

4) Stabilizing wall

To achieve the high values of β_N, 3.5-4, required in the high-efficiency steady-state operation in ITER and the demonstration reactor, improved stability using the stabilizing wall is necessary. The practical β_N-limit determined by the ideal MHD stability can be improved by an ideally conducting wall, located close to the plasma surface. The stabilizing effect depends upon the profiles of current and pressure, and it is more influential for unstable modes with broad eigenfunctions spreading to the plasma surface. In particular, it has been theoretically shown that the wall stabilizing effect is remarkable for the reversed shear plasmas (Fig. 3.1.4-11) [3.1.4-24].

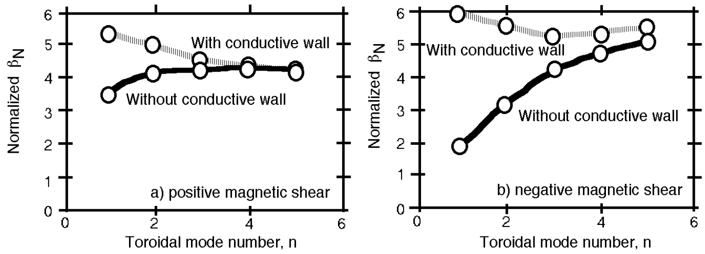

Fig. 3.1.4-11 Significant improvement of β-limit by wall stabilization (left: positive magnetic shear, right: negative magnetic shear). Horizontal axis is the wave number in the toroidal direction for the eigenfunction of the instability.

In case where the β_N-value exceeds the non-wall limit, the stability theory predicts the appearance of a resistive wall mode that grows in a time, relevant to the characteristic time for the magnetic field lines to penetrate into the wall. The DIII-D experiment first documented the resistive wall mode at a β_N lower than the limit expected by the ideal wall stabilization [3.1.4-25]. Theoretically, the resistive wall mode can be stabilized by keeping the plasma rotating between a few to a few tens of kHz or by the application of a corrective magnetic field, which cancels the perturbed magnetic field resulting from the instability. Intensive research to identify the characteristics of the resistive wall mode and development of its stabilization techniques are presently being undertaken in various tokamaks.

Figure 3.1.4-12 shows the classification of the MHD stability regimes and the design parameters of ITER and the demonstration reactors (ARIES-I, SSTR, ARIES-RS), summarized by S. Jardin [3.1.4-26] (with an additional point from CREST). The operating point of the ITER in the inductive operation and ARIES-I do not require the wall stabilization. In turn, stabilization of the resistive wall modes are required against the n=1 mode for SSTR and the n≥1 modes for ARIES-RS and CREST.

Figure 3.1.4-13 shows the waveform of the weak magnetic shear mode discharge in DIII-D [3.1.4-27]. In this discharge, high performance was sustained for 2 seconds with high values of β_N (>3.5), which exceeds the required value in the fusion demonstration reactor.

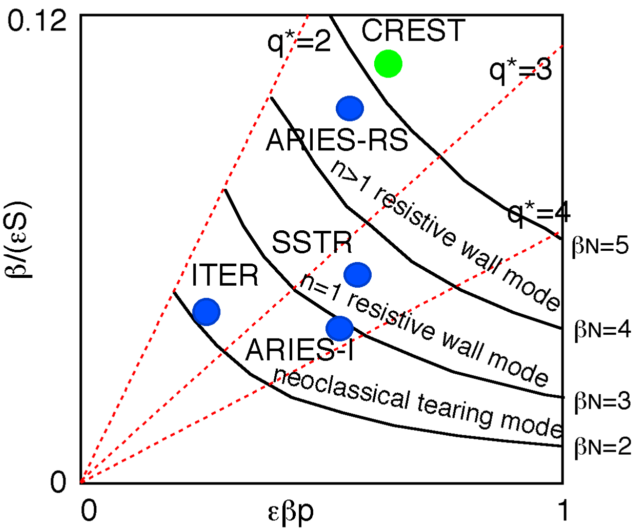

Fig. 3.1.4-12
The tokamak MHD operation region.
(βp: poloidal β value, ε=a/R)

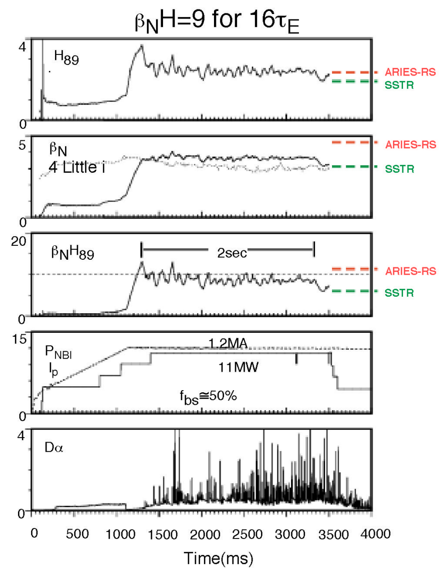

Fig. 3.1.4-13
Waveforms of the weak magnetic shear plasma in DIII-D

5) Resistive instabilities

The previous sections dealt with the ideal MHD instabilities with fast growth time. However, a recent high-beta tokamak experiment indicates that the sustainable β_N value in a long pulse is lower than the β_N–limit determined by the ideal MHD predictions. It has been found that degradation is caused by the appearance of resistive MHD instabilities, which are induced by the finite resistivity of the plasma having a growth rate much less than that for the ideal instabilities. Figure 3.1.4-14 shows the JT-60 results. Intensive investigations performed recently showed that the dependence of the critical β_N value at the onset of the resistive instabilities on the plasma collisionality is consistent with the theoretical prediction for the neoclassical tearing mode. At present, many tokamaks are striving to stabilize this mode by the application of local current profile control with electron cyclotron current drive. Accordingly, a successful result has been obtained in ASDEX-U [3.1.4-28].

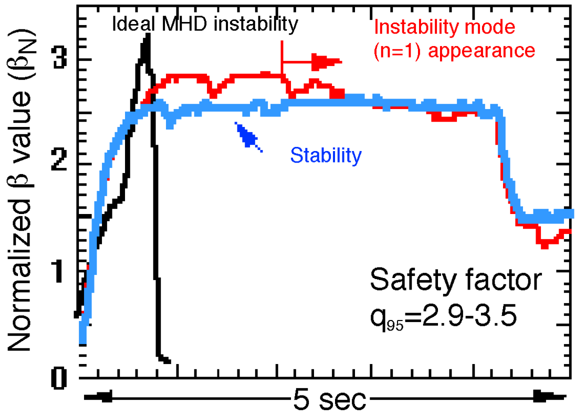

Figure 3.1.4-14
In quasi-steady-state discharges, β_N is lower than the ideal MHD limit due to appearance of resistive MHD instabilities (JT-60).

6) Heat and particle control using the Edge Localized Modes (ELMs)

In ITER and demonstration reactors, an H-mode with ELMs is thought to be the standard operation mode to avoid the harmful impurity accumulations and substantial degradation in energy confinement. Based on the long history of the H-mode research after its discovery in ASDEX [3.1.4-1], it is known that ELMs are benign instabilities driven by the current and pressure gradient, which does not accompany a large scale collapse, and that ELMs are useful for expelling the impurities from the plasma confinement region. There are three types of ELMs, i.e., Type I, II, and III, observed in tokamaks so far. The standard type observed in high confinement plasmas is called Type I. Since a Type I ELM has low frequency but a large amplitude, as shown in Fig. 3.1.4-15, it is apprehended that erosion of the divertor plates may occur due to the large instantaneous peak heat load onto the divertor plates during each ELM event. In the ITER-class devices, the peak heat load of the Type I ELMs does not seem critical. However, for the long-pulse operation, ELMs with small amplitude and a high frequency (Type II) are more beneficial than the Type I ELMs. Investigation of the H-mode operation region with the Type II ELMs has been carried out. In JT-60, the Type II ELMs appear at high triangularity and in a high safety factor regime. It also has been clarified that the confinement performance is not degraded, compared to the Type I ELMs, and the impurity accumulation is also relatively small. An operation region having high triangularity and high safety factor is relevant to the steady-state operation in ITER and thedemonstration reactor, and a good choice from the strategical point of view in the fusion R&D.

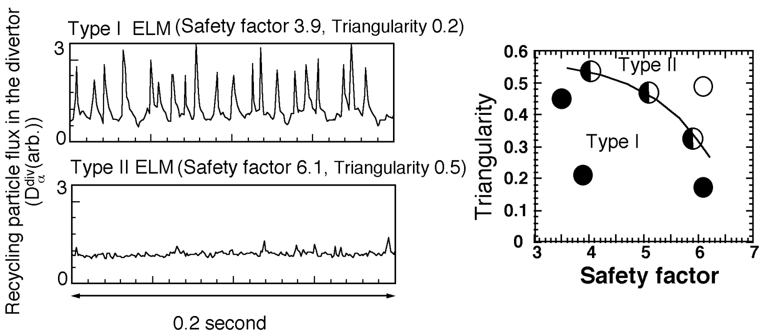

Fig. 3.1.4-15
Discharge regions for Type I and type II ELMs (βp>1.5 – 1.6)

For the β_N-limit and ELMs, the main issues of concern and future prospects have been reviewed as described above. For all of the above issues, promising approaches to resolve the problems have been proposed, and many of them have already been settled although some others are still under intensive investigation.

(5) Confinement of energetic particles

1) Heating by energetic particles (alpha particles)

Fusion reactors such as ITER are supposed to confine the energetic alpha particles in the fusion reaction until the particles are slowed to thermal velocities to allow the self-heating of the plasma and sustain the burning. As many of the experiments performed in the past indicate, the slowing-down time of such energetic ions agrees well with classical estimate, as shown in Fig. 3.1.4-16 [3.1.4-29]. Furthermore, the experimentally evaluated diffusion coefficient of energetic particles, which has provided the key to predicting the confinement of alpha particles, is as low as 0.1-0.01 m^2 / s, being consistent with the neoclassical model. The rather small diffusion coefficient of energetic ions, in comparison with the commonly observed "anomalous" diffusion of thermal ions, may be explained by the "orbit averaging" of small-scale plasma turbulence over the large orbits of the energetic ions. The diffusion coefficient of alpha particles in ITER is estimated to be as low as 0.1-0.01 m^2 / s. Therefore, sufficiently good confinement of the energetic particles can be expected if the increase of transport caused by the magnetic field perturbation described below is removed.

2) Ripple loss

A small-scale loss of symmetry in the toroidal magnetic field leads to the loss of alpha particles, which is dubbed "ripple loss," and results in hot spots on the first wall of a fusion reactor. For this reason, the ripple loss mechanism will have to be understood well for the reactor design. In recent experiments performed in large tokamaks, a theoretical model of the ripple loss has been validated intensively, which is applicable to the design of the experimental and demonstration fusion reactors. Although the ripple loss is enhanced in the advanced operation based on reversed shear, it was demonstrated that the insertion of ferritic steel to the vacuum vessel could reduce the loss of energetic ions to an acceptable level [3.1.4-30].

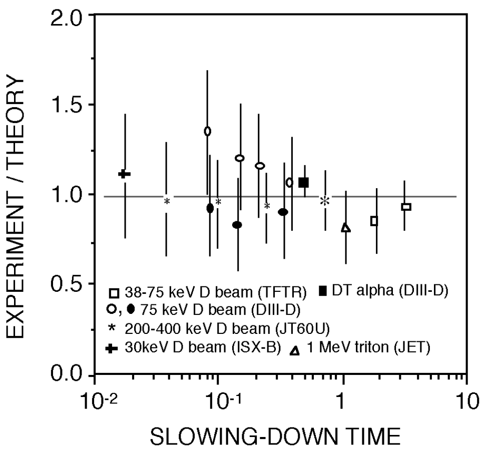

Fig. 3.1.4-16 Experimental and theoretical (classical) Slow-Down Times of energetic particles

3) Alfvén eigenmodes (AE modes)

When the velocity of energetic particles is as high as the phase velocity of shear Alfvén waves, the wave-particle resonant interaction can occur, destabilizing the AE modes under certain conditions. When the AE modes are destabilized, the resulting magnetic fluctuations can possibly expel the alpha particles. In the recent tokamak experiments, it was found that the observed AE frequencies, mode structures, and instability thresholds are consistent with the present linear theory. The nonlinear saturation level of the modes and the interaction between the individual AE modes and the interaction of the alpha particles with the modes are open questions to be resolved. It should be noted, however, that the AE modes in the experimental fusion reactor, ITER, are expected to differ from those observed in present tokamaks with respect to mode stabilization mechanisms, mode structures, etc. According to the linear theory, ITER plasma can be near the destabilization boundary of the AE modes, and it is then likely that several tens of AE modes with short-wave lengths will be destabilized under certain operation conditions [3.1.4-31]. Such multiple AE modes should be extensively investigated in the experimental fusion reactor to confirm their effects on alpha particle transport, i.e., flattening of the alpha particle density profile and the alpha particle loss.

4) Extension of the results obtained in the experimental reactor to the demonstration reactor

Both in the experimental reactor and demonstration reactor built in the next phase, the physics of the energetic particles would be the same. Thus, accumulation of the experimental results on AE modes, sawtooth stabilization by the energetic particles obtained in the experimental fusion reactor will make it easy to predict the energetic particle properties in the demonstration reactor. The nonlinearity of the AE mode characteristics may lead to a different behavior of AE modes in the demonstration reactor, which will operate at higher alpha particle beta value. In order to predict the behavior of AE modes more reliably, a reliable AE nonlinear model should be developed on the basis of observations in the experimental reactor.

(6) DT burning and burn control

As shown in Section 3.1.1, a DT fusion power exceeding 10 MW has been demonstrated in TFTR (US) and JET (EU), and the characteristics of the DT plasmas have been studied. In JET (see (2)), the threshold heating power required for the transition from the L- to H-mode was studied by changing the deuterium and tritium ratio. The threshold power in the DT plasmas was significantly lower than in the deuterium plasmas. Thus, the new scaling of the L-H transition threshold power Eq. 2-1 was obtained, incorporating the dependence on the averaged mass number (M^{-1}). It was also shown that the previous scaling of energy confinement for the ELMy-H mode is applicable also for the DT plasmas. Consequently, the accuracy of the confinement and threshold scalings was improved for extrapolation to ITER [3.1.4-32].

In JET, the maximum DT fusion power reached 16.1MW. As for the α-particle behavior, the confinement property, the slowing-down and heating processes, were shown to be classical [3.1.4-29, 3.4.4-33]. It was also documented that the electron thermal diffusivity in the core plasma region is the same for both DD and DT plasmas, if the effects of α-particle heating are taken into account. The DT fusion power obtained agreed with what was expected from the DD experiments, thus, it is now possible to evaluate the performance of DT plasmas accurately [3.1.4-32].

In the DT experiments, formation of the internal transport barrier has also been examined in the reversed shear plasmas, which is expected to be a promising mode for the steady-state operation of ITER and demonstration reactors. In addition, it has been confirmed that in DT plasmas the internal transport barrier can be produced, and this is accompanied by a large pressure gradient [3.1.4-33].

Following the theoretical prediction of possible destabilization of Alfvén eigenmodes by α-particles produced by the DT reaction, theoretical and experimental research on the AE modes have been conducted extensively. In TFTR DT experiments, the toroidal Alfvén eigenmode (TAE mode) was observed, as predicted by the linear theory. In these experiments, the β-value of the α-particles (β_α) was small and anomalous transport of α-particles was not observed because of the small TAE mode amplitude [3.1.4-34]. However, the β_α-value in ITER is expected to be larger than in TFTR by one order of magnitude, and the spatial structure of the mode may be different from that in the present devices. Therefore, the interaction between the Alfvén eigenmodes and α-particles, which influences the transport of α-particles, should be studied in detail in ITER.

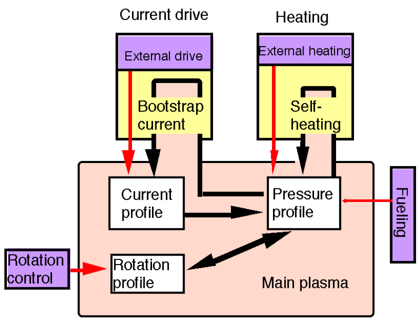

Fig. 3.1.4-17 Schematic of the external controls applied to burning plasmas

The burn control is one of the most important missions in both the experimental and demonstration reactors. In these devices, where the fundamental heating is the 3.5MeV α-particle heating, understanding the energy confinement and spatial distribution of the α-particles is the principal issue. A second important issue is the establishment of the burn control technique. The burn controls will be accomplished by means of the deuterium and tritium mixture fueling, external heating, energy confinement control, and particle confinement control (both fuel particles and impurities, including helium ash). Figure 3.1.4-17 shows a schematic of the external controls applied to the burning plasmas.

In the burning operation in ITER, the external heating power is 1/3 of the total heating power. It is thought that the fueling rate and external heating power will control the fusion power in a steady-state. In ITER steady-state operation, an advanced burn control scheme should be established that sustains a high value of β_N and full non-inductive current drive with a high fraction of bootstrap current. In demonstration reactors, such a burn control would have to be demonstrated with a small fraction of external heating power (<10% of the total heating power).

In the present devices, the total heating power is determined by the external heating, and the pressure profile is controlled by the external power. In the demonstration reactors and the commercial reactors that follow, the heating profile will be determined by plasma itself (self-sustained plasma), as the self-heating by the DT reaction is to be higher than 90%. In addition, the current profile should be controlled by the externally driven current, of which fraction is 20 – 30% of the total current, since the bootstrap current fraction will reach 70 – 80%.

ITER is the first device in which the burn control can be examined for such self-sustained plasmas. From this point of view, the significance of the ITER program is overwhelming.

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3.1.5 Issues and prospects towards steady-state operation

(1) Current drive and current profile control.

In a tokamak, the confinement magnetic field is formed by producing the current in plasma. However, it is necessary to keep increasing the current in the primary coil monotonously in the inductive method, using the principle of an electric transformer. Accordingly, steady maintenance of the magnetic field for plasma confinement is difficult with the inductive method. Therefore, the plasma current must be driven without depending on the inductive method for a steady-state operation, and a noninductive current drive is thereby required. The research on the noninductive current drives in tokamaks has been carried out since the first principle demonstration was performed in JFT-2 (nearly 20 years) [3.1.5-1].

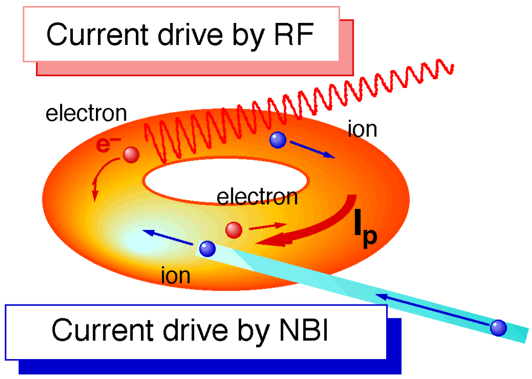

Fig. 3.1.5-1 Current generation mechanism by the particle beam and radio frequency waves in a tokamak

For the noninductive current drive with external input, there are two principal schemes, the use of a particle beam and radio-frequency (RF) waves, as indicated in Fig. 3.1.5-1. On the other hand, the plasma current is spontaneously generated when the pressure of the plasma increases with respect to the magnetic pressure of poloidal field. This current fraction is known as the bootstrap current. The continuous operation of a tokamak fusion reactor is realized by the combination of the bootstrap current and the noninductive current drive, supplied externally as a beam or radio-frequency waves.

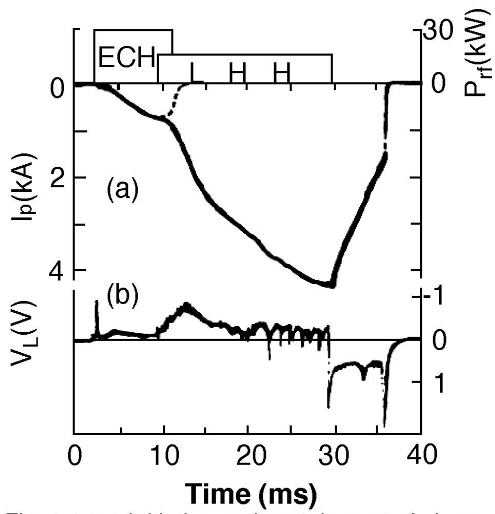

Fig. 3.1.5-2 Initiation and sustainment of plasma current by EC+LH in the WT-2 tokamak

1) Non-inductive current drive with external input

The feasibility of the following four techniques have been investigated to drive the plasma current in the fusion experimental reactor ITER with external input. (a) 1-Mev energy neutral beam injection (NBI), (b) 170-GHz electron cyclotron range of frequency (ECRF), (c) 40 to 70-MHz ion cyclotron range of frequency, (d) 5-GHz lower hybrid range of frequency (LHRF). Scheme (a) is realized by injecting an energetic neutral beam into plasma, while (b)-(d) use radio frequency waves in the frequency ranges stated. The numerical values indicated above are relevant to the beam acceleration energy or RF frequency anticipated in ITER.

At Kyoto University in Japan, leading plasma research on the generation, sustainment, and current stability improvement by means of plasma current distribution has been carried out using RF. In WT-2, startup and sustainment of the plasma current by EC+LH, as shown in Fig. 3.1.5-2, was demonstrated for the first time in the world [3.1.5-2]. The current drive performance was improved on WT-3.

One of the most important figures of merit is the current drive efficiency (η_{CD}) for the noninductive current drive. The efficiency is written using the driven current I_{CD}, plasma major radius R, electron density n_e, injection power P as $\eta_{CD} = I_{CD} \cdot R \cdot n_e / P$. The lower hybrid wave (LHCD) has achieved the highest current drive efficiency so far.

In JT-60, the world's highest current drive efficiency, 0.35×10^{20} MA · m $^{-2}$ · MW $^{-1}$, has been achieved, and the world's highest driven current, 3.6 MA, under a fully noninductive current drive has also been acieved[3.1-5.3].

As shown in Fig. 3.1.5-3, the world's longest tokamak discharge, 2 hours, was demonstrated by means of the LH full-current drive in TRIAM-1M at Kyushu University [3.1.5-4]. In TORESUPRA, full current drive for 2 minutes was also achieved [3.1.5-5].

On the other hand, it is expected that the power injection to plasma will be easier with NBI and ECRF, and that they are more suitable for use in a fusion reactor, where the operating range and engineering constraints are more stringent. NBI especially has been used in high-power heating experiments in many tokamaks and achieved high-confinement performance. Therefore, it is expected to be the principal plasma power injection scheme for ITER and future devices. However, the acceleration energy of the neutral beam must be raised to about 1 MeV for application to high-density and large-scale fusion plasmas. Accordingly, it is necessary to use the negative-ion beams, whose neutralization efficiency is significantly high. The importance of this energetic beam-heating technique has been recognized for many years in Japan, and development of the NBI system using the negative-ion source (N-NBI) has been promoted. As a result, the 0.5-MeV N-NBI system was completed and installed on JT-60, and experiments have followed. To date, the relevant physics aspects have been investigated and the driven current profile by N-NBI was identified, which indicates that it can be reasonably explained by the Coulomb collision theory [3.1.5-6]. Moreover, a current drive efficiency of 0.13×10^{20} MA · m^{-2} · MW^{-1} was also obtained and, an improvement in performance is in progress. Based on these results, the consensus is that the N-NBI current drive is explained well by the existing theories, and its performance in fusion plasmas is predictable. In conclusion, it was confirmed that the current drive efficiency of N-NBI in ITER could satisfy the requirements.

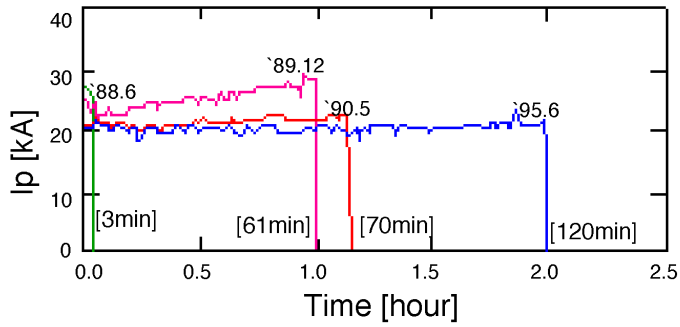

Fig. 3.1.5-3 Discharge waveforms of the world's longest discharge obtained in TRIAM-1M

As for ECRF, its contribution to the stabilization of the resistive modes is expected, as shown in Section 3.1.4, but future relevant research and development is called for. Although experimental use of the ICRF current drive has been carried out in many countries, sufficient efficiency has not yet been documented. However, it was found that the values obtained agree well with the theoretical predictions.

2) Steady state operation utilizing the bootstrap current

The steady-state operation scenario of a tokamak using an external current drive source has been substantially modified after the experimental documentation of the bootstrap current, followed by the proposal to construct a steady tokamak reactor, SSTR, by Japan Atomic Energy Research Institute. The bootstrap current, driven by the plasma pressure gradient, in JT-60 was demonstrated to reach 80% of the total plasma current [3.1.5-7]. The bootstrap current is predicted by the neo-classical theory, and experimental results in many devices agree with the theoretical prediction. Based on these results, the concept of steady-state tokamak operation, so called advanced tokamak operation, where most of the total plasma current is sustained by the bootstrap current, has quickly pervaded. Moreover, the superiority of this advanced mode of operation has been emphasized after the discovery of high confinement weak or negative magnetic shear plasmas in various tokamak devices, such as JT-60 (3.1.4). Since the bootstrap current is driven by the pressure gradient, it becomes zero at the plasma center, where the pressure gradient is zero. Therefore, the spatial distribution of the current density in the plasma becomes flat or even hollow when the fraction of the bootstrap current increases. This is how the weak / negative magnetic shear plasma is formed. In an advanced tokamak with a high bootstrap current fraction operation, based on this scheme, the fraction of plasma current driven by external drivers can be a few tens of % of the total current. The most important role of the externally driven current is to form the stable current distribution profile while supplementing the bootstrap current.

In Fig. 3.1.5-4, the waveform of full non-inductive current drive plasma obtained in JT-60 weak magnetic shear plasma is depicted. In the discharge, the full noninductive current drive condition was achieved by a bootstrap current fraction of approximately 70% and the remainder was supplied by the beam current drive. Here, the normalized beta value was 2.9, and the confinement enhancement factor to the L-mode was 2.5. Therefore, it can be concluded that the possibility of the steady-state operation with weak magnetic shear was demonstrated for extrapolation in ITER and the demonstration reactor [3.1.5-8].

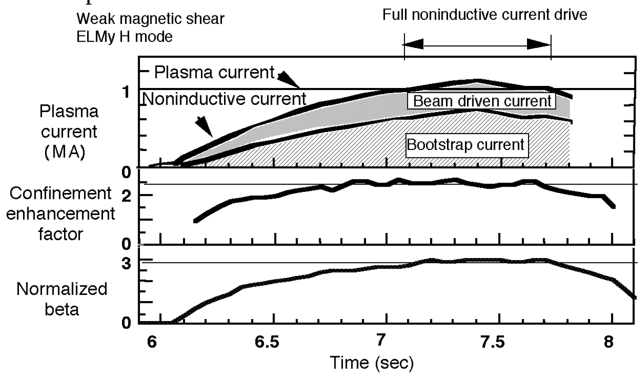

Fig. 3.1.5-4 Waveforms of full noninductive current drive discharge in weak magnetic shear operation

The waveform of the full noninductive current drive plasma obtained in the JT-60 negative magnetic shear plasma is also shown in Fig. 3.1.5-5. In this discharge, the bootstrap current fraction reached 80%, and the remainder was driven by the beams [3.1.5-9]. Although the normalized beta is relatively low, around 2, the confinement enhancement factor to the L-mode was 3.6, and the corresponding facor to the H-mode was 2.2. These results are significant for the steady-state operation scenario based on the negative magnetic shear in ITER and the demonstration reactor.

Controversial issues of research on the current drive are the development of an operation scheme that provides the steady sustainment of improved confinement and high-beta plasma by current profile control (3.1.4). In the advanced tokamak operation, the structure of internal transport barriers, which varies with the current profile, and the bootstrap current profile, determined by the pressure profile, strongly couple with each other. The development of the current profile control scheme necessary for plasma with stable and stationary profiles is an important issue. In JT-60, the quasi-steady sustainment of the negative magnetic shear equilibrium by the LHCD hollow current profile [3.1.5-10] as well as the full current drive by increasing the bootstrap current fraction to 80% in combination with the current drive by NBI to form an appropriate current profile have been demonstrated. In TORE-SUPRA, the negative magnetic shear equilibrium is formed and sustained only by LHRF [3.1.5-5]. The demonstration of longer sustainment, in which current distribution reaches steady state, is an investigation issue for the future.

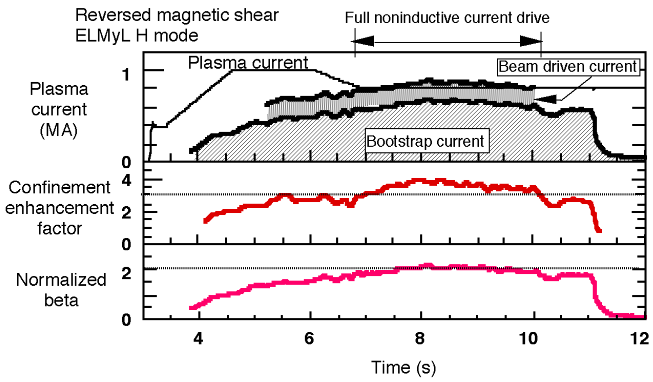

Fig. 3.1.5-5 Waveforms of full noninductive current drive discharge in negative magnetic shear operation.

The stabilization of the resistive mode that accompanies the magnetic island generation, such as the neoclassical tearing mode, is also an important issue. For effective stabilization, locally peaked current drive by ECRF close to the center of the magnetic island is required. Much effort has been devoted in many devices in the world. Accordingly, the stabilization effect was confirmed in a middle-sized tokamak, namely ASDEX-U of Germany. Verification in larger and higher magnetic field tokamaks is presently called for.

In ITER, particularly in the steady-state burning plasma, confirmation of the controllability of the current profile under the high bootstrap current condition as well as the development of the operation scenario relevant to the simultaneous achievement of improved confinement and high-beta steady-state conditions are important issues. In addition, it is prerequisite to acquire the quantitative prediction capability of the NBI current drive efficiency and the driving current distribution when the Alfvén eigenmodes (3.1.4) appear.

(2) Particle Exhaust and Impurity Control

The fusion reaction in a D-T fusion reactor continuously produces alpha particles with 3.5- MeV energy. Alpha particles are trapped in the magnetic field and they heat the plasma (alpha heating) as they slow down through collisions with the background plasma particles. Newly created thermal helium ions (He ash) must be removed from the core plasma. The overall performance of the fusion reactor critically depends on the capability of removing this He ash to prevent dilution of the D-T fuel while maintaining the fusion burning. If the helium ash exhaust is insufficient, helium ash will accumulate and the fusion power will be reduced. Therefore, the control and continuous purging of the He ash are essential in future tokamak reactors. Helium ions in the core plasma experience a charge exchange in the divertor region through plasma-surface interactions, after which the neutral He particles are exhausted by divertor pumping. To examine the He exhaust capability in burning plasmas, intensive studies have been performed, and it was found that successful reactor operation can be maintained only if the He ash is removed from the system within a period of 10 times the energy confinement time for the ITER-FDR design. For ITER-FEAT, which is a compact-sized ITER, the ratio of the global particle confinement time of the He ash inside the plasma chamber, τ^*_{He} to the energy confinement time, τ_E is required to be less than 5. In addition, it is also claimed that the helium fraction in the plasma has to be less than 5% for ITER-FEAT. Accordingly, experimental studies of helium ash exhaust have been performed to evaluate the He ash properties in ELMy-H mode plasmas in DIII-D, JT-60, and other devises.

The JT-60 divertor was modified from open divertor geometry to the W-shaped divertor that has pumping capabilities. The helium exhaust experiments were performed in a way to simulate the He ash behavior in the ELMy- H mode plasmas with the W-shaped pumped divertor. Three units of cryopumps were “argon (Ar) frosted” to effectively exhaust the helium. As a result, favorable He exhaust capability was successfully demonstrated in JT-60 for the first time in the world [3.1.5-11]. The helium particles were continuously supplied to the central region of the plasma for 6 s by injecting 60-keV neutral beams to simulate He ash production by the fusion reaction (Fig. 3.1.5-6 (a)). Here, the He source rate was 1.5×10^{20} /s (equivalent to 85 MW of α-heating, which corresponds to 80% of He ash in ITER-FEAT), and it was balanced by the He pumping to reach the steady-state condition 1.5 s after the start of the He fueling. The He concentration reached 4% of the electron density in the main plasma, and it was held constant for 4 s. Figure 3.1.5-6 (b) compares the measured He density with and without the He pumping.

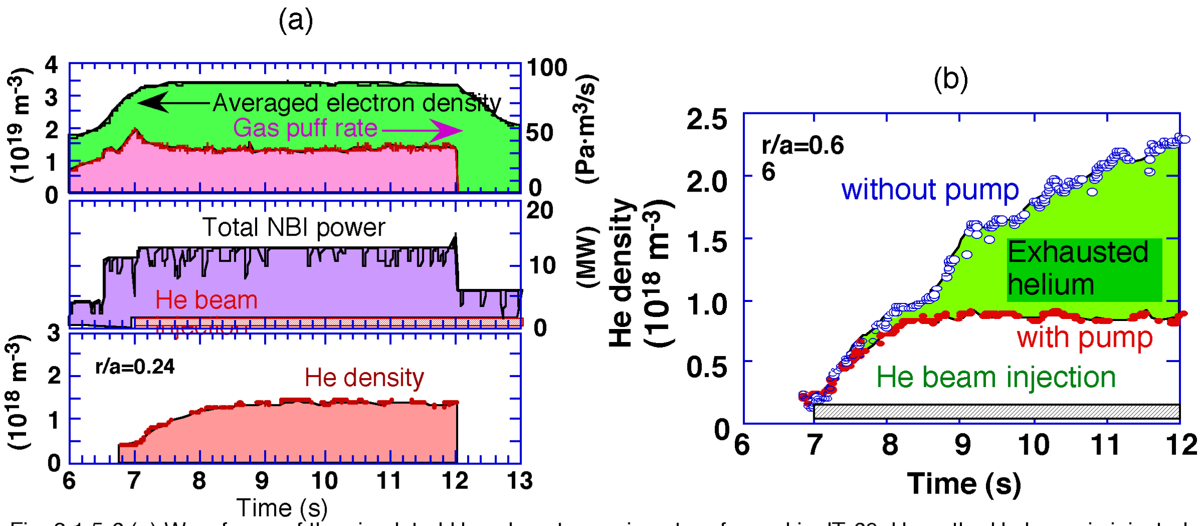

Fig. 3.1.5-6 (a) Waveforms of the simulated He exhaust experiment performed in JT-60. Here, the He beam is injected into the plasma to simulate He ash production by the fusion reaction. (b) The measured He density reaches equilibrium 1.5 s after the start of the He beam injection, being balanced by the pumping.

With He pumping, the ratio of $\tau_{He} / \tau_E = 4$ was achieved, and it was shown that efficient He exhaust is possible well within the range necessary for the successful operation of the ITER-FEAT ($\tau_{He} / \tau_E \leq 5$). Without pumping, the He concentration linearly increased; accumulation of helium was observed. In addition, the enrichment factor of He was evaluated to be around 1.0, which is five times larger than the ITER requirement of 0.2.

The enrichment factor of He, η_{He} was defined by $\eta_{He} = [P_{He}/2P_{D2}]_{div} / [n_{He}/n_e]_{main}$, where $[P_{He}/2P_{D2}]_{div}$ is the ratio of the He neutral pressure to the deuterium neutral pressure in the divertor, and $[n_{He}/n_e]_{main}$ is the ratio of the He density to the electron density in the main plasma. The key issue of concern in the future fusion reactors in regard to the He ash pumping is to reduce the pumping speed in the divertor as well as the ratio of τ_{He} / τ_E, as both helium and fuel particles are simultaneously exhausted by the divertor pumping. The higher He enrichment factor indicates that the required pumping speed can be lower and the tritium inventory in the torus can be reduced. Therefore, the above results strongly support the present ITER design, and predict that He accumulation does not present an obstacle to the sustainment of burning Q=10 plasmas in ITER.

In the fusion reactors, removal of the impurities from the first wall and seeding impurities (neon, argon, etc.) to enhance radiative cooling in the divertor is necessary, in addition to exhausting the He ash. The requirements for the plasma purity for ITER in terms of the effective ionic charge, Z_{eff} , is $Z_{eff} \leq 1.8$. The following two approaches are presently proposed and were examined in JT-60 to reduce the impurity generation from the first wall, where divertor plates are, in particular, the main impurity source. (1) Reduction of the heat flux on the divertor plates to suppress impurity generation by the physical sputtering. (2) Introduction of the dome structure in the divertor to suppress chemical sputtering by neutral particles. In practice, reduction of the CD-band intensities emitted from hydrocarbon molecules by the chemical sputtering was measured with a spectrometer, after the introduction of the dome structure in the W-shaped divertor of JT-60. As a result, the concentration of carbon impurities in the main plasma was successfully reduced [3.1.5-12]. On the other hand, impurity (Ne, Ar) seeding is considered to be effective for the radiative cooling of divertor plasmas. The increase of edge and divertor density provides the enrichment of seeded impurities in the divertor and enables the effective pumping of seeded impurities.

The helium exhaust for the advanced tokamak operation of future steady-state tokamak reactors is an issue of serious concern. The experimental investigation of He exhaust in JT-60 reversed shear plasmas indicates that helium removal inside the ITB was about twice as difficult as outside the ITB [3.1.5-13]. It was found that τ^*_{He} / τ_E is approximately 10 in the reversed shear plasma with an H-factor ($= \tau_E / \tau_E^{ITER89P}$) of around 1.5. In this case, the He exhaust capability is not sufficient to remove the helium ash inside the ITB. Improvement of the He exhaust efficiency in the reversed shear discharge with high-edge density and high recycling flux, with improved confinement (H-factor ≥ 2) is a key issue for the success of the advanced tokamak operation scenarios (non-inductive current drive and Q =5) in ITER.

In ITER, investigation of the He ash exhausting capabilities and establishment of its control method are necessary in the high fusion power output operation, in the long burn operation and in the steady-state operation. In addition, investigations related to the generation, back flow, and penetration of impurities under the intense divertor conditions with the large heat flux is necessary in reactor relevant plasmas. Furthermore, intensive studies on transport of the seeded impurity and development of its control scheme have to be carried out.

(3) Divertor heat flux control

To suppress the heat flux onto the divertor targets, it is necessary to produce radiative divertor plasmas and to dissipate the power that comes from upstream of divertor. The most important issue is to satisfy the integrated performances required for ITER, namely the high main plasma density (80-110% of the Greenwald density), radiative divertor plasma (80-95% of the input power), improved energy confinement (H-factor ~ 2), and the low Z_{eff} ($Z_{eff} < 1.9$).

From the recent divertor experiments, a subset of the ITER requirements relevant to an H-factor ~ 2, a total radiation power fraction of ~ 0.8, and 80% of the Greenwald density is expected to be achieved in the ELMy-H-mode plasmas. At present, there seems to be a density limit as high as 80% of the Greenwald density, and above that the improved energy confinement starts to degrade.

Recent progress in the divertor simulation code and the divertor design by making full use of the code resulted in a radiative loss as large as 80% of the input power in ELMy-H-mode plasmas without strong gas puff with the Lyla II type divertor in ASDEX-U[3.1.5-14]. Since reduction of the heat flux was achieved experimentally, as predicted by computer simulation, we have had confidence in the divertor design for fusion reactors. On the other hand, it was demonstrated in DIII-D that the SOL (Scrape-Off-Layer) flow, produced by combining the intense injection of fuel gas and Argon gas with strong divertor pumping, suppressed the impurity accumulation in the main plasma to realize Z_{eff} = 1.8. Here, the radiation loss fraction to the total input power of 0.72 and an H-factor of 1.7 were obtained [3.1.5-15]. It was also demonstrated that the SOL flow was beneficial to increase the enrichment of Argon in the divertor and to enhance the radiation loss in the divertor with a lower Z_{eff}.

It is known that central fueling by pellet injection is more suitable for high density than the edge fueling by gas puffing. In addition, it was confirmed both experimentally and theoretically that the pellet infection from the high magnetic field side had a higher particle-fueling rate than that from the low magnetic field side. In DIII-D, high-density plasma with 150% of Greenwald density was obtained with an H-factor of 1.8 in a pellet fueled and strongly pumped discharge [3.1.5-16].

In the effort to produce a radiative divertor in the reversed shear plasmas, it was first demonstrated in JT-60 that the divertor detachment, accompanied by a high radiative power loss rate, as large as 0.8, can coexist with the internal transport barrier [3.1.5-17]. Based on recent progress in providing even better target plasma in the reversed shear plasma operation, a high confinement plasma with H-factor of 1.8 and a radiation loss fraction of 0.66 was obtained by argon injection at 82% of the Greenwald density [3.1.5-18]. An issue for future investigation is to suppress the accumulation of Ar in the main plasma.

In an effort to produce high plasma density and a high radiation loss fraction by impurity gas injection, the RI mode and the CDH mode were discovered. The former was found in the limiter discharges in TEXTOR. The latter is characterized by the peaked density profile and good energy confinement. It is triggered when the radiation loss in the edge plasma reaches a certain amount due to the impurity injection, such as Neon. In the RI mode plasmas in TEXTOR, a radiation loss fraction as large as 0.8 and 140% of Greenwald density were simultaneously obtained with the good energy confinement [3.1.5-19]. In order to reproduce the RI mode in the pumped divertor configuration, various experiments have been undertaken in large- and medium-size tokamak devices. The RI mode in the divertor configuration was first obtained in DIII-D, where the plasma density was as low as 40-50% of the Greenwald density [3.1.5-20]. Therefore, high-density operation is desired in the future. The CDH-mode, found in ASDEX-U, obtained a total radiation fraction of nearly 0.95. It is noteworthy that as the radiation loss increased, the density profile became peaked and then 90% of the Greenwald density was obtained.

As described in this section, the ITER requirement of an H-factor around 2, a total radiation power fraction of approximately 0.8, and 80% of the Greenwald density seems to be accessible. In ITER, burning plasmas coexisting with the radiative divertor plasmas must be demonstrated. Therefore, an operation regime with a high radiation loss ratio of > 0.8 must be sought, since a higher radiation loss fraction of > 0.9 is required in the demonstration reactor.

It is generally difficult to achieve low Z_{eff} criterion in the radiative reversed shear, RI-mode and CDH-mode plasmas, since large amounts of impurities would have to be injected to sustain those plasmas. In the CDH-mode, Z_{eff} < 3 has not been obtained and reduction of contamination due to the seeded impurities remains an important challenge. Long sustainment of these radiative discharges is also important.

In the demonstration reactor, the plasma density is expected to exceed the Greenwald density by 50%. Therefore, discharges with high-energy confinement in such a high-density regime must be developed. The pellet injection is also promising, since a plasma density much higher than the Greenwald density has been achieved in medium-sized tokamaks. Such high-density plasmas must be demonstrated in the large-size tokamaks.

When the ELM frequency is too low in ELMy-H-mode plasmas, divertor targets will be damaged due to the pulsed heat fluxes at each ELM and the resulting significant rise in temperature. Accordingly, H-mode plasmas with a high-ELM frequency, which are described in Section 3.1.4, must be studied.

(4) Disruption Control

In major plasma disruptions, instantaneous release of the plasma thermal energy occurs in the first phase. This is followed by the release of the plasma magnetic energy, which is characterized by the rapid termination of the plasma current, in the second phase. The former and the latter phenomena are respectively called thermal quench and current quench. The disruption phenomena observed in major plasma disruptions are briefly illustrated in Fig. 3.1.5-7.

At the thermal quench phase, the plasma thermal energy is deposited on the first wall, in particular on the divertor plates. The first wall materials thereby suffer substantial erosion, especially in case where excess plasma energy is deposited. For plasmas with a vertically elongated cross section, vertical displacement of the plasma column is often observed at the plasma current termination phase. In such a case, the plasma boundary is pushed onto the first wall, and halo current is induced. The halo current, interacting with the toroidal magnetic field, produces a localized electromagnetic force on the in-vessel components, and this may result in damage to the first wall. In case where the plasma displacement is small at the current quench phase, runaway electrons with energies of several tens of MeV are generated. The localized high heat load produced by these runaway electrons can damage the first wall. Accordingly, major plasma disruptions have been considered to be one of the important issues to be solved. Remarkable progress in the disruption studies was made in the last 5 years at JT-60 and other tokamaks in the world. The demonstration of the mitigation scheme of various disruption phenomena mentioned above was performed, and prediction capabilities related to the counteractions to ameliorate the disruption effect have been improved. The major issues of concern at present are thereby moving towards engineering research and development to mitigate the disruption phenomena in more reliable way, as well as to avoid the disruption itself.

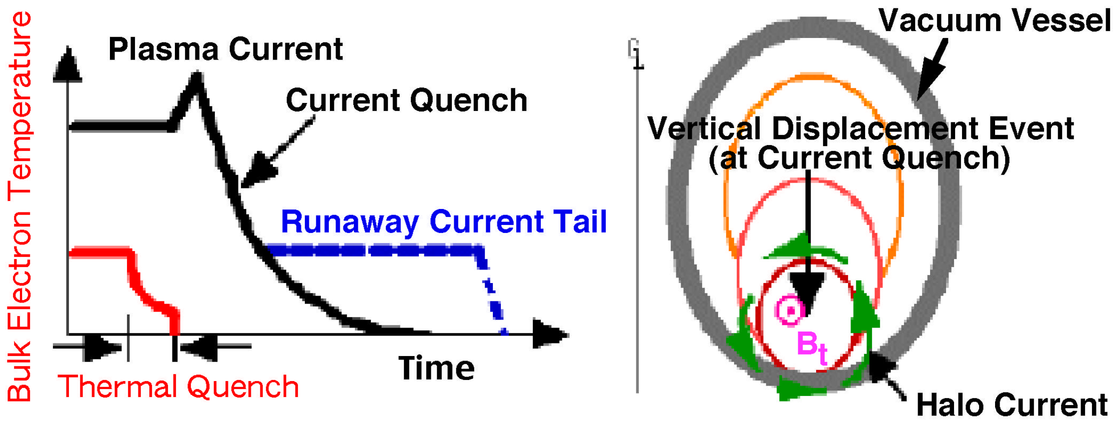

Fig. 3.1.5-7 Disruption schematics. Thermal quench, followed by plasma current termination and generation of runaway electron current tail (left hand figure), Vertical displacement event and generation of halo current (right hand figure).

1. Present status on the mitigation of major disruptions

Thermal Quench:

At the occurrence of a thermal quench, plasma thermal energy is deposited onto the first wall, in particular heavily on the divertor plates, within the very short duration of about 1 ms. As the number of replacement of the first wall tiles in a fusion reactor will be limited, the erosion of the tiles should be minimized. The effectiveness of a vapor shielding mechanism is theoretically predicted to suppress the erosion. It is claimed that the heat load on the first wall could be reduced by the shielding effect of the vapor from the first wall. However, experimental confirmation of the theoretical model is not straightforward in the present tokamak devices, as the thermal energy is much too low.

On the other hand, erosion of the first wall could be remarkably reduced if the plasma thermal energy could be quickly diminished. In JT-60, reduction of the heat load on the first wall has been demonstrated by the injection of a neon ice pellet into the plasma, which has lead to the fast conversion of thermal energy to the radiation energy [3.1.5-22]. The effectiveness of such an impurity injection method has been confirmed by reproducible tests in other devices, such as ASDEX, DIII-D.

Plasma Current Quench:

The characteristic time of the current quench is determined by the electron temperature. The predicted electron temperature from the compiled disruption database, where the time constant of the current quench is evaluated by L/R, is around 3 eV. Here, L and R respectively denote plasma inductance and toroidal resistance. Based on this estimate, the minimum termination time of plasma current, i.e., the maximum current quench rate in ITER, can be estimated to be about 50 ms.

Vertical Displacement Event and Halo Current:

The vertical displacement event (VDE) is often observed at plasma current quench. The magnitude of VDE is governed not only by the vertical instability but also by the vertical force induced by the vertical asymmetry of the toroidal eddy current, which is generated in the vacuum vessel at the plasma current quench. It was found in JT-60 that VDE does not occur when the center of the plasma column is adjusted to form a symmetrical eddy current [3.1.5-23]. The significance of this finding is that the plasma displacement can be determined by the geometrical vessel design. In case where the time constant of current termination is relatively long, degraded accuracy in the detected vertical position may cause the VDE. However, avoidance and suppression of the VDE have been experimentally demonstrated by control based on the accurate detection of the vertical position [3.1.5-24].

The halo current flows in the scrape-off layer of the plasma, the first wall and the in-vessel structure with the helical components in toroidal direction. The component of halo current in the poloidal direction interacting with the toroidal magnetic field results in localized electromagnetic force induced on the in-vessel components. Therefore, it is quite important to estimate the magnitude of the halo current in the reactor design. The characteristics of halo current are presently predicted by the experimental data compiled in ALCATOR-Cmod, ASDEX-U, Compass-D, DIII-D, JET, and JT-60 [3.1.5-25]. The electromagnetic force induced on the in-vessel components is scaled in terms of the amplitude, the peakedness of the toroidal inhomogeniety. The Halo current is characterized as follows.

- Maximum halo current reaches 50% of the plasma current before the major disruption.
- Maximum toroidal peaking factor is about 4.
- Maximum halo current is inversely proportional to the toroidal peaking factor.

The halo current is relatively low in large tokamak devices. Therefore, it is anticipated that the ratio of halo current to plasma current tends to decrease in devices as they increase in size. The interpretation of such a tendency is presently under investigation. The reduction of halo current by the injection of a neon ice-pellet or an impurity gas puff, observed in the several tokamaks, such as ASDEX-U, is taken into account as a promising method to suppress the halo current [3.1.5-26].

Runaway electrons:

When the electron temperature is rapidly reduced at a thermal quench, a substantial electric field is induced to conserve the poloidal magnetic flux, and this occasionally generates runaway electrons with energies up to several tens of MeV. The runaway electrons could produce an intense local heat flux of more than 100MJ / m ^{2} on the first wall. Thus, the development of a scheme to avoid or suppress the runaway electrons is demanded. In JT-60, it was found that the generation of runaway electrons could be avoided in the presence of large magnetic fluctuations, which appear at a major plasma disruption [3.1.5-27]. Furthermore, it was heuristically documented that the runaway electrons are not produced if the surface safety factor is below 2 [3.1.5-28]. The theoretical analysis and numerical simulation performed indicate that the runaway electrons are lost by the collisionless scattering in the presence of large magnetic fluctuation with low-m/n mode instabilities, as the toroidal kinetic momentum of electrons is not conserved [3.1.5-29]. In conclusion, it is suggested that the runaway electron tail can be spontaneously terminated by the growth of the instability when the decrease of the plasma radius and the resulting decrease of the surface safety factor are induced by the VDE.

2. Present Status on the avoidance of major disruptions

A major disruption is induced by several different causes, which are classified into the following categories, namely, density limits, impurity accumulations, too low or too high internal inductance, stability limits of plasma pressure, external error field, vertical instability, plasma surface safety factor around 2 or 3, and the minimum safety factor around 2 or 3 in reversed shear plasma configuration. It is expected that disruptions caused by the above mechanisms can be avoided through the deliberate engineering design of the device and optimization of the operating scenario [3.1.5-30]. Therefore, possible occurrences of disruption in the fusion reactor could be considered as possible events in the process of the optimization of operation scenarios, fault operation, failures in hardware, or an emergency interrupt triggered by the safety interlock.

3. Issues of future investigation related to the avoidance and mitigation of the disruption events

Larger thermal and magnetic energy will be released at a disruption in ITER, compared to present large-scale devices. The essential issue is if the predicted disruption effect is within the engineering allowance in the present ITER design. In the demonstration reactor, disruption should be minimized to as little as 0.5 disruption event per year. In order to minimize the disruptions, sufficient physics margins against the operating boundaries, such as the β-limit, the density limit, and an internal inductance limit, should be retained. In the demonstration reactor, since β-value, density, and radiation fraction have to be increased, probability of disruptions will increase. Therefore, it is an urgent issue of investigation to establish the scheme to avoid disruptions near the operation boundaries.

The disruption phenomena could possibly be predicted by detecting the precursor signals, by the probability analysis near the operation boundary in terms of the safety margin, or by their combination, since the disruption is often observed near the operational limit. In case the growth of relevant instability is relatively slow, real-time control to avoid the disruption is possible after the precursor is detected. The feedback response does not have to be extremely fast for resistive MHD instability, such as the density limit. On the other hand, substantially fast feedback stabilization would have to be performed for ideal MHD instabilities even though the precursor could be detected well before the disruptions. Therefore, it is necessary to always bear in mind how close the plasma is to the operational boundaries. Effective prediction of disruptions by using a neural network has been experimentally demonstrated in DIII-D [3.1.5-31]. It is anticipated that the advancement in computing technology could enable the plasma operation with real-time detection of MHD instability and its feedback suppression, which would contribute to a remarkable improvement in disruption control.

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3.1.6 Plasma control issues in a steady-state fusion reactor

(1) Load following operation

Normally, fusion reactors would supply constant electrical loads. Should they be able to vary their output power easily, however, their value as attractive power sources would be enhanced. At present, varying electrical loads are accommodated by load-following operation performed by thermal power plants. If the fusion reactors had such load-following capability, they could be main power sources in the future

A steady-state fusion reactor has a finite energy amplification factor and requires external electric power for driving the plasma current. It is suggested that in such a steady-state reactor, the fusion output can be varied substantially by controlling the current drive power, plasma density, and thus the plasma current, keeping the confinement improvement factor constant [3.1.6-1]. Figure 3.1.6-1 shows a result of computer simulation of the output control in a steady-state fusion reactor. Here, the fusion power output is reduced from 3.8 GW (100%) to 1.3 GW (34%) by reducing the plasma current (rating 12 MA) to 75%, the electron density to 60%, and the external electric power for the current drive (rating 60 MW) to 75%. In this example, during the 200 seconds of output control period, the neutral beam driven current and the bootstrap current sustain the discharge, while the confinement improvement factor is kept almost constant.

To realize such an output control scheme in a steady-state fusion reactor, research and development of the control of burning plasma should be conducted in an experimental reactor. Based on the results of prior investigations, optimal engineering design and a demonstration of the ability of the blankets to withstand the varying neutron and heat loads would have to be performed. In principle, the steady-state fusion reactor can reduce its output to zero. Therefore, it can be a flexible power plant that can deal with emergency occasions, such as the trouble in power transportation (distribution grid)system.

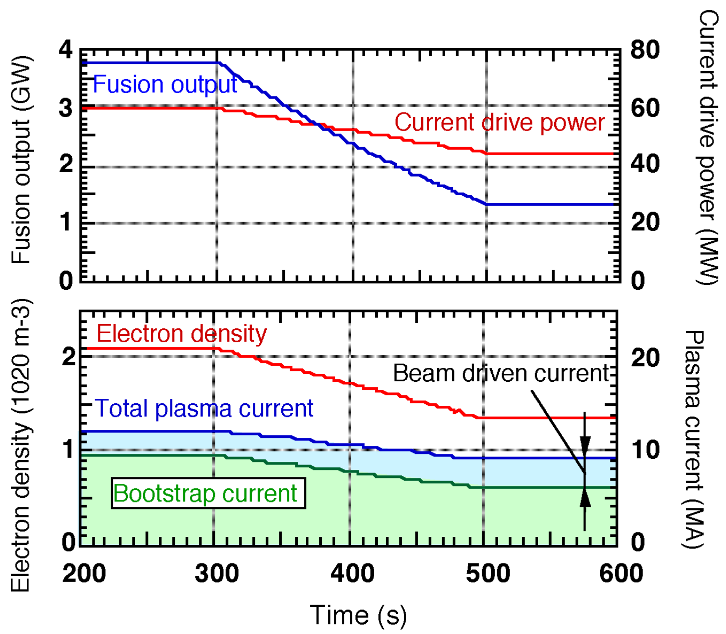

Fig. 3.1.6-1 A computer simulation of the output control operation in a steady-state fusion reactor. Here, A-SSTR with a confinement improvement factor of 1.05±0.05 and a helium concentration ratio of 8% is considered.

(2) Control of high bootstrap current plasma

As described in Section 3.1.5, full noninductive current drive with high bootstrap current fraction has been demonstrated in JT-60 and other devices. In ITER and the DEMO reactors, alpha particle heating is dominant

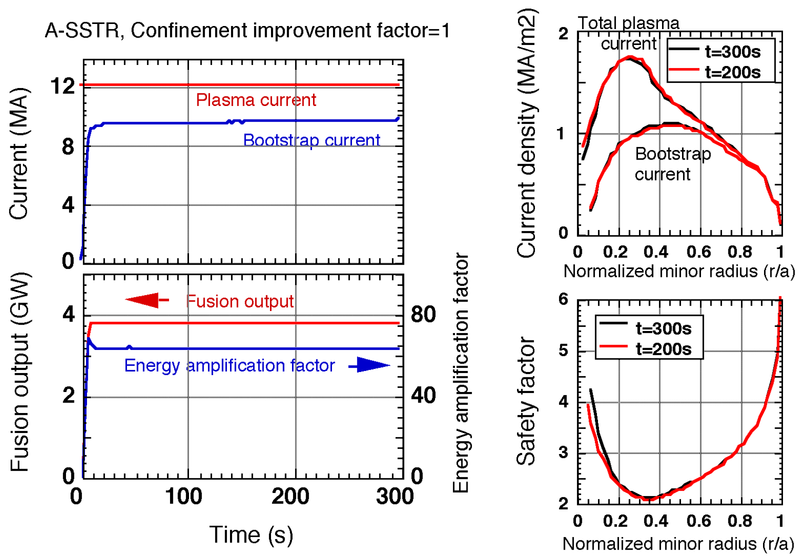

Fig. 3.1.6-2 Results of a computer simulation of the current profile during the plasma current ramp-up phase in the A-SSTR

and it is expected that the coupling between the plasma heating, plasma pressure, and plasma current will become stronger. Hence, the control of these elements is an important issue of concern.

Figure 3.1.6-2 shows the results of computer simulation of plasma current profile during the plasma current ramp-up phase in the A-SSTR. As seen in the figure, the current profile is maintained in an almost stationary state. This means that the high bootstrap current fraction can be sustained stably. For the plasma control, the above mentioned load-following operation is an issue of concern. Because if the load-following operation should reduce the plasma current below its rated value, a variation in the plasma current profile will simultaneously be induced. Fig. 3.1.6-3 shows an expected time evolution of current profile in a load-following operation.

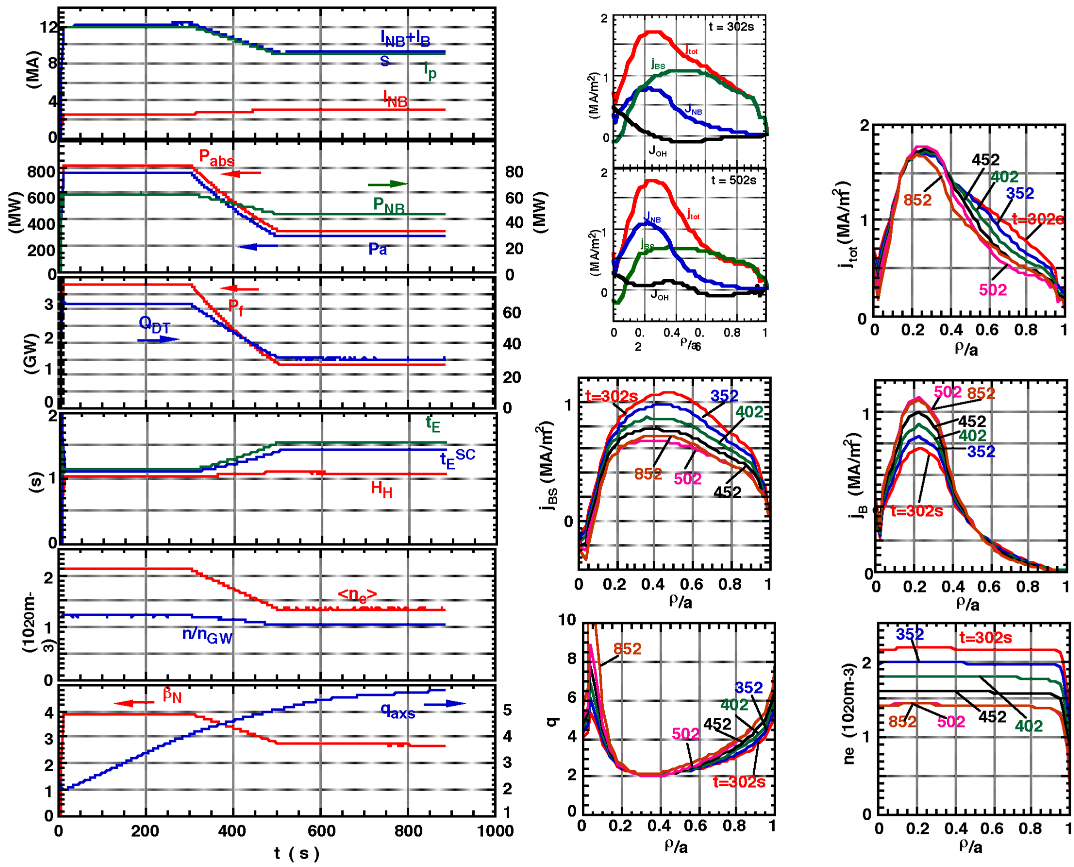

Fig. 3.1.6-3 A scenario of load-following operation in the A-SSTR and the expected evolution of current profiles

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3.1.7 Advanced material and plasma wall interaction

In D-T fusion reactors, the structural materials are radio-activated by the 14-MeV neutrons produced in the plasma. Therefore, it is important to develop materials that will have low activation under this intense fast neutron irradiation. Among the other components, the blanket component materials inside the vacuum vessel have to endure the most rigorous neutron flux. Therefore, appropriate materials having low activation, high heat conductivity, and minimal swelling have to be developed. The reduced-activation ferritic steels, vanadium alloys, and SiC / SiC composites have favorable characteristics, and they are expected to be the leading candidate materials for the fusion reactor structural components. Here, we will discuss the experimental status and future issues, related to the compatibility of advanced materials (reduced-activation ferritic steel) and the next generation of advanced materials (vanadium alloys and SiC composites) with the required characteristics to have no harmful influence on the plasma performance.

(1) Reduced-activation ferritic steel

The composition of reduced-activation ferritic steel is chosen to reduce activation while retaining the inherent favorable characteristics. Accordingly, high-activation elements, like molybdenum, are replaced with alternative elements, like tungsten. The reduced-activation ferritic steel is the principal candidate material for the blanket structural material for the steady-state tokamak reactor (SSTR) being proposed by JAERI.

1) Present status, related to the application of reduced-activation ferritic steel to the tokamak experiments

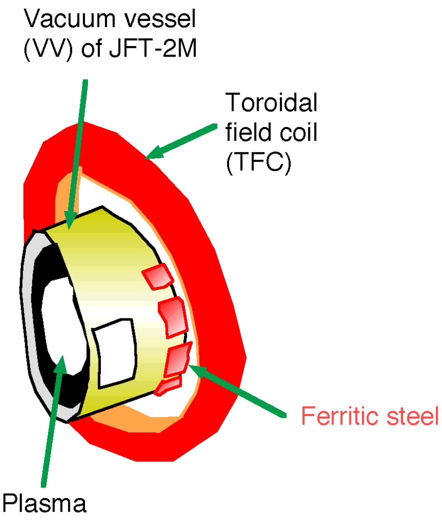

Fig. 3.1.7-1 Schematic diagram of ferritic board for ripple reduction in JFT-2M.

Since the reduced-activation ferritic steel inherently has ferromagnetic properties, it is apprehended that this steel may produce adverse effects on plasma production, plasma control, and confinement performance. However, it was proposed that the magnetic field structure could be improved by taking the advantage of the ferromagnetic properties of the steel. The basic idea is to install the ferritic steel in a way that strengthens the magnetic field between the toroidal field coils (TFC), where the toroidal magnetic field is weaker. This will reduce the TF ripple fraction. The energetic particle heat flux to the wall, which is induced by the ripple loss, is enhanced in a negative shear discharge. The reduction of TF ripple is a critical issue for the steady-state operation in ITER so application of such steel may be beneficial for ITER. The proof of principle experiment on the ripple loss reduction was performed in small- to medium-size tokamaks in Japan. In the HT-2 tokamak, ferritic steel plates were installed inside the vacuum vessel (VV), and successful plasma equilibrium formation was demonstrated [3.1.7-1]. In order to investigate the feasibility of reduced-activation ferritic steel (F82H) in a tokamak device, the Advanced Material Tokamak Experiment (AMTEX) has been performed at JFT-2M. The ripple reduction experiment was carried out as the first step [3.1.7-2]. The schematic diagram of the configuration of the ferritic board outside the

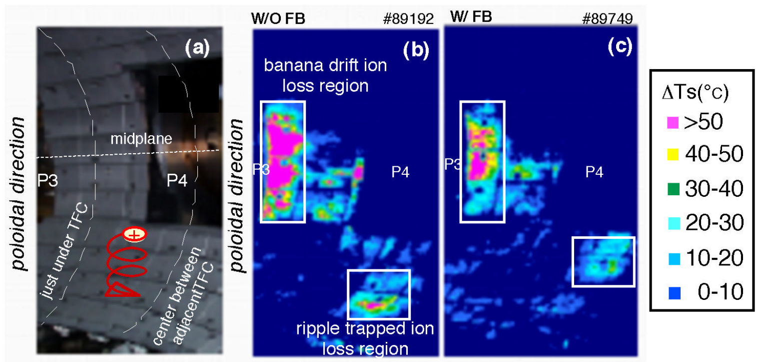

Fig. 3.1.7-2 (a) View of out-board wall inside the vacuum vessel in IRTV measurement. (b)(c) Increments of the wall temperature in degrees during NBI heating (Vacc = 36 keV, PNBI ~ 0.54 MW), (b) before and (c)after the ferritic board is installed. The direction of ion ∇B drift is downward.

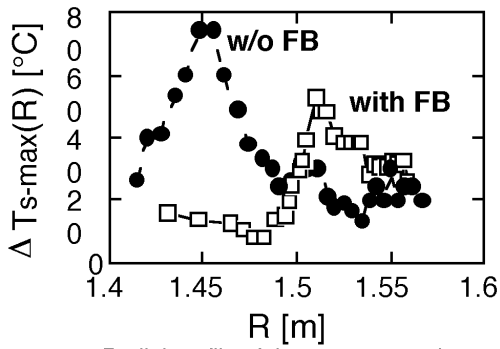

Fig. 3.1.7-3 Radial profile of the temperature increment in degrees, produced by ripple trapped ion losses before (broken line, l) and after (solid line, q) the ferritic board installation.

VV is depicted in Fig. 3.1.7-1. According to the simulation results, the ripple rate is reduced from 2.2% to 1.1% at the plasma periphery by the installation of the ferritic board. The reduction of the ripple loss rate was hereby confirmed experimentally by magnetic probe measurements, and the ripple loss reduction of energetic particles by the ferritic board insertion was demonstrated effectively. The results obtained are shown in Fig. 3.1.7-2 and 3.1.7-3.

2) Future issues

The effect ferromagnetic ferritic steel has on the plasma performance would have to be investigated further. In addition, ferritic steel is easily oxidized and emits a large amount of gas under ultra-high vacuum conditions. Therefore, the compatibility of the ferritic steel under high-performance plasma conditions should be demonstrated to establish the suitability for the structural material of the future fusion reactor. The anticipated research subjects are as follows.

- Investigate the effect of the ferritic steel on the equilibrium magnetic field and plasma performance, and as necessary, establish counter-measures for detrimental effects.
- Investigate the plasma-wall interaction of the ferritic steel
- Demonstrate the high-performance plasma in a vacuum vessel wall made of the ferritic steel.

(2) Advanced materials for the next generation tokamaks

In addition to the ferritic steel, the feasibility of other advanced materials for the next generation tokamaks should be examined in a tokamak experiment. These materials, i.e., vanadium alloy and silicon carbide compound materials (SiC / SiC), have many favorable characteristics regarding activation and thermal properties.

1) Vanadium Alloy

Vanadium alloy is a candidate material for the principal structural material of the ARIES-RS reactor, which is the conceptual design demonstration reactor proposed by the United States. Experimental documentation is necessary to address the issues of utmost concern, including those related to embrittlement caused by the occlusion of hydrogen or helium during the tokamak operation. In practice, relevant investigations have been hitherto performed in DIII-D and JFT-2M. In recent US / Japan collaboration studies, performed in JFT-2M, it was confirmed after a nine-month exposure to the tokamak environment that the hydrogen content in the test specimen decreased from 18 ppm to 8 ppm at high temperature (300 degrees) and the absorption of fuel deuterium was only 2 ppm.

For the next step, planning us underway to acquire the data at different fluence and energy of action particles. This will be accumulated in the database for extrapolation for the demonstration reactor.

2) Silicon carbide composite

Silicon carbide composite is considered a potential structural material for the conceptual design of the commercial reactor DREAM, proposed by JAERI, and the demonstration reactor ARIES-I, proposed by the US. Investigations focused on the characteristics of silicon carbide composite, relevant to the plasma wall interactions, have not been performed. Therefore, extended research on the sputtering and heat load characteristics would have to be carried out, preferably under the tokamak environment, in order for this material to be qualified as a desirable structural material.

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3.2 Current Status and Future Subjects of Fusion Reactor Technology Development

In this section, concerns with the overview items of fusion reactor technology development for the experimental reactor, the demonstration reactor and the commercial reactor are enunciated. The expected achievements and difficulties as well as the development perspectives are described.

3.2-1 Development of Component Technology for Tokamak Type Fusion Reactor

Development of component technology for tokamak type fusion reactor are shown in Fig. 3.2.1-1.

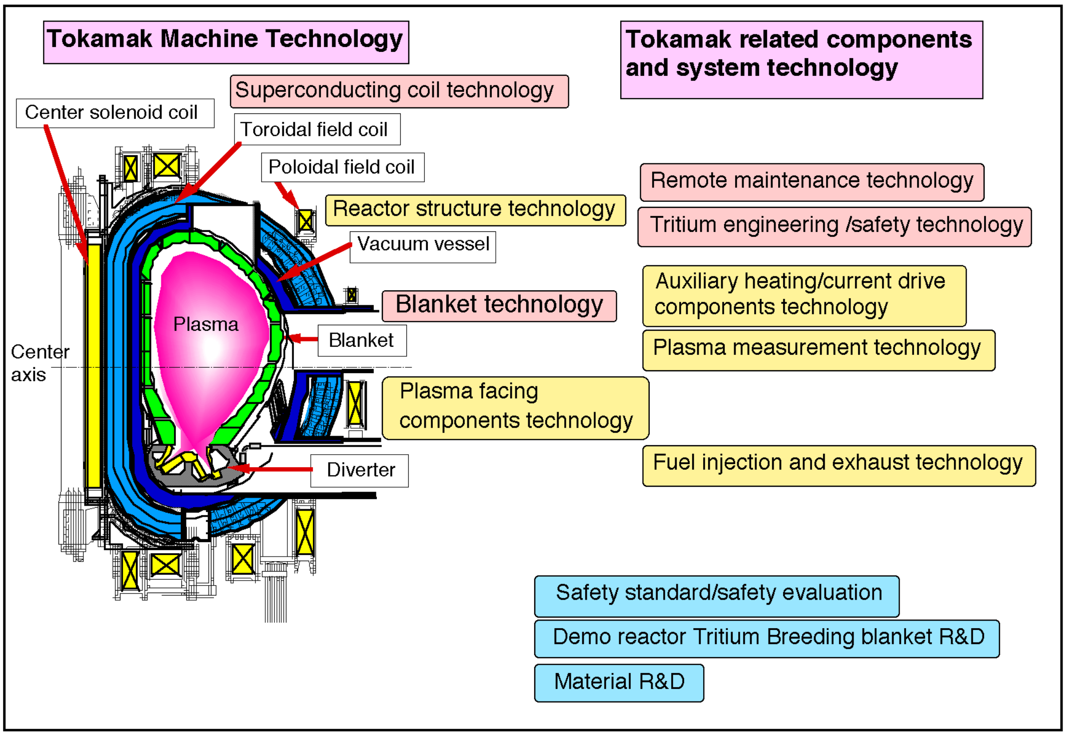


Fig. 3.2.1-1 Development Issues of Component Technology for Tokamak Type Fusion Reactor

<u>Technologies of Tokamak Components are;</u>

- Blanket Technology: to develop the blanket that surrounds plasma and converts the kinetic energy of neutrons and other particles into heat and also shields the superconducting magnets from radiation
- Plasma Facing Components Technology: to develop the divertor that captures the high-energy particles and absorbs the heat load from plasma
- Reactor Structural Technology: to develop the vacuum vessel and support structures that will sustain the high vacuum for generation of plasma and contain the blanket and divertor
- Superconducting Magnet Technology: to develop the superconducting magnet that provide magnetic field to confine plasma, which is a magnetohydrodynamic fluid, and induces a current in the plasma by varying the magnetic field

<u>Component technologies related to the tokamak are;</u>

- Auxiliary Heating and Current Drive Equipment Technology; to heat the plasma and drive the plasma current
- Plasma Measurement Technology: to measure the temperature and density of plasma to form and control the plasma
- Fuel Injection and Exhaust Technology; to inject and exhaust fuel
- Tritium Engineering /Safety Technology: to recycle tritium safely, which is radioactive and do not exist naturally

• Remote Maintenance Technology; to remotely maintain and repair the components that are radio-activated by neutrons generated from the plasma

Furthermore, toward the safety review for licensing and the future power reactor development;

• Preparation of safety standards required for safety review for licensing, and data and evaluation methods required for safety evaluation
• Research and development of the Tritium breeding blankets for the future power reactors
• Development of first wall materials are required.

3.2.2 Phased Integration of Reactor Technology Development

Fig. 3.2.2-1 Current Statues and Future Subjects of Reactor Technology Development

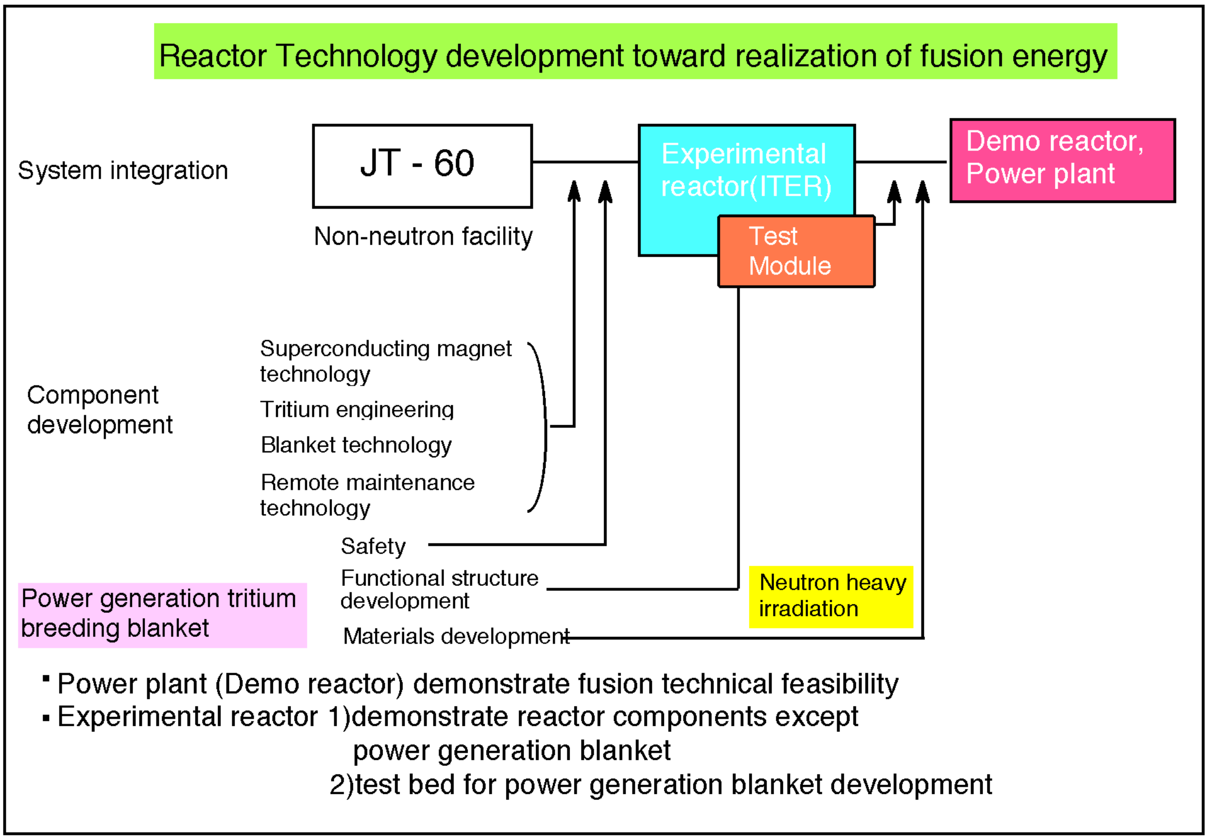


Fig. 3.2.2-1 Current Statues and Future Subjects of Reactor Technology Development

Figure 3.2.2-1 shows development steps of the reactor component technology that will be needed to realize ITER and DEMO reactor.

JT-60 has been operated a long series of experiments more than 15 years and has performed pulsed operation of 10-second maximum duration using normal (not superconducting) copper coils, and has developed the system integration technology for a tokamak facility that can provide deuterium plasmas.

The technologies JT-60 embraces are magnetic confinement, plasma control, heating, vacuum pumping, cooling, and so on, which are characteristics of a fusion facility. Further, it can said that the prototype engineering base described in Fig. 3.2.1-1 has already been integrated through the construction of integrated devices such as JT-60. However, neutron fluence generated from deuterium plasma is incredibly low in comparison with that from DT burning plasma, so it is difficult to claim that JT-60 is a real nuclear device.

In future efforts to realize fusion energy, component technologies will be integrated into the core device. And, each component technology must be designated as being for the experimental reactor phase, ITER, or the next step, the ‘demo reactor phase.’ This discrimination is decided from the judgment of which step is the most effective for system integration of each component toward a fusion power plant, which will involve a balance between the maturity of integrated technology and the capacity of core device.

The experimental reactor (ITER) requires that superconducting magnet technology be advanced for reliable long-pulse operation and tritium technology must be able to safely use tritium fuel. Furthermore, blanket technology for shielding against neutrons resulting from the D-T fusion reaction and remote maintenance technology to remotely maintain equipment that has been radio-activated by neutrons are also required. Accordingly safety technology is required, and that should safely manage the increase of radioactive waste and materials in comparison with present ones.

These technologies are being integrated into the core device in the ITER phase for the first time. In this case, these have made steady development progress with the planned integration into a Tokamak device and cannot be suddenly integrated in ITER. For instance, superconducting magnet technology, in international collaboration under the International Energy Agency (IEA), during 10 years following 1977, six large superconducting magnets, which were fabricated by different worldwide national institutes and private companies, were gathered and tested in a toroidal layout in the Oak Ridge National Laboratory in the US. (9T, 1GJ). Superconducting magnets have been applied as small- and middle-scale devices in Tokamak facilities such as the TRIAM-1M of Kyusyu University and the Tore-Supra in France, and moreover, the Large Helical Device (LHD), with a more complicated coil shape, in the National Institute for Fusion Science of Ministry of Education, Science, Sports and Culture. For Tritium engineering technology and safety technology, loop tests and combined tests in a test circulation system with a Tokamak facility, as shown in Fig. 3.2.2-2, have been already carried out in TSTA, in the US, and TPL, in the Japan Atomic Energy Research Institute, and tritium has been used as fuel in DT experiments in TFTR and JET. On remote maintenance technology, the remote maintenance system could be developed in its practical scale through the ITER engineering R&D. In addition, though it involved small-weight loads, remote-handling technology has been used for maintenance in vacuum vessel components in JET. As mentioned above, these technologies will be integrated through the needed development steps before integration in ITER.

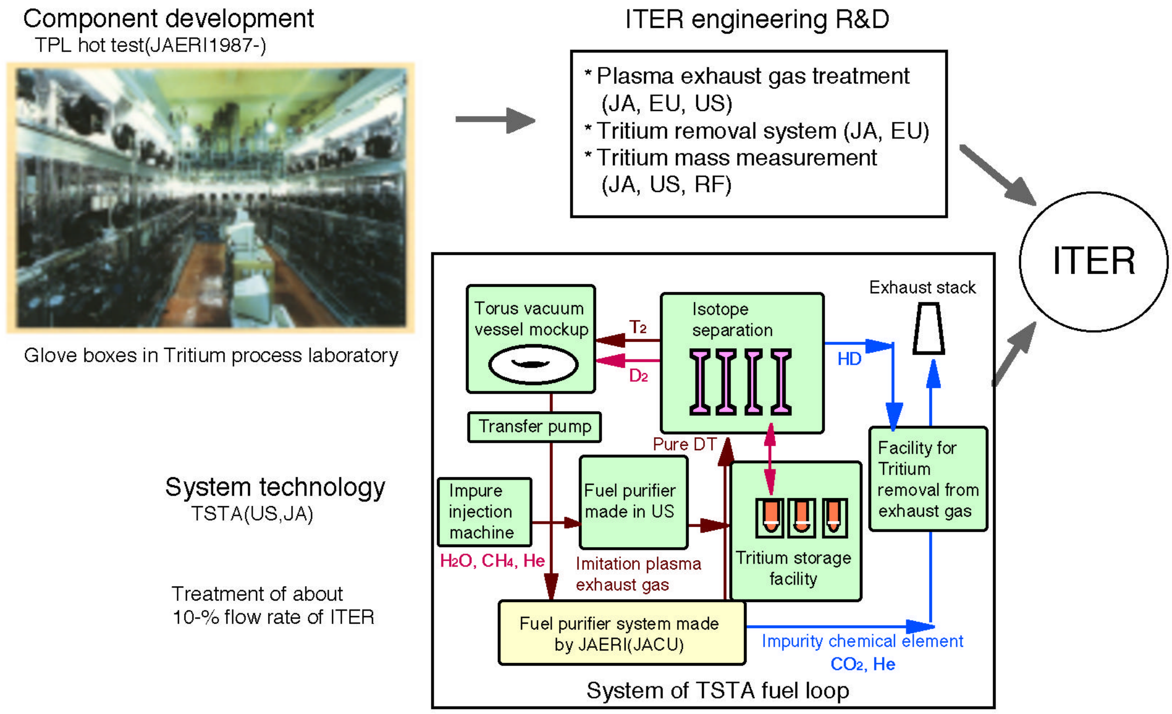


Fig. 3.2.2-2 Tritium Engineering and Safety Technology

As mentioned above, the experimental reactor ITER combines all elements of reactor components except those of material development including the power generation blanket. ITER aims for a demonstration of the integrated technology of these components.

The ITER Engineering Design Activity included seven especially large components. Technology R&D had never been performed on such a large scale before. It was accomplished cooperation with 4 parties, each taking a leading part in ITER Engineering Design Activity and each having made significant breakthroughs. By these achievements, the technology required to construct the experimental reactor has reached almost 80% as a whole, and the remaining achievements, which are expected to be attained during remaining ITER/EDA, will provide adequate technical data to start the construction of ITER. The details are described in Section 3.2.3.

The DEMO reactor, which is the next step after the ITER experimental reactor, is the final target in the development phase of a fusion power reactor. It includes all components of a future power reactor and aims to demonstrate power generation on a practical power-plant scale. Reactor technology, which is integrated in this phase, will include material development including the power generation blanket.

As stated, materials development and structure module development are needed for development of the power generation blanket. To develop the blanket structure module, the removal of high-temperature heat and tritium breeding performance of the module, developed by the neutron irradiation database at the material test fission reactor, should be tested in the environment of high-energy neutron flux, at a high-heat load, and with a large electromagnetic force, as close as possible to a practical fusion reactor. ITER itself will function as a fusion environment test bed. On the other hand, for the development of materials, data to estimate neutron irradiation degradation is important to determine the lifetime of materials. Until now, performance degradation has been estimated using a material test fission reactor and a small-scale 14-MeV neutron source. However, that is inadequate for the work ahead so construction of a strong 14-MeV neutron source and a materials testing facility are needed to perform the heavy irradiation tests in strong neutron fluence required for fusion reactor materials. In parallel with ITER construction and operation, these material development and blanket functional tests should proceed and both successful achievements must be integrated and reflected into the design of the blanket module for the DEMO reactor.

At present, according to this plan, development of reduced-activation ferrite F82H material as blanket structure material for the DEMO reactor and a lithium-titanium compound as a tritium breeding material take a leading part of development. Moreover bonding technology for development of the heat removing structure casing is proceeding.

In above basic system integration, technologies for the DEMO reactor will be developed. In addition, further research of each component is required to improve reliability, the economy, and safety to elevate the public acceptance of fusion power. Typical studies include, an increase of the magnetic field generated by superconducting magnets, improvement of refrigeration efficiency, development of structural materials such as vanadium alloys and SiC, development of high-heat flux components with long lifetimes, improvement of remote maintenance and exchange technology, and measures toward a steady operation of the tritium plant. In this research for improvements in superconducting magnet technology, there are developments that include a superconducting magnet using niobium-aluminum alloy conductor and, in addition, a high-temperature superconductor based on bismuth alloys, which is in early development at present.

3.2.3 Outline and current status of reactor technology required for each phase of the experimental reactor and demonstration reactors

The development target and current status of achievement of each component item for ITER and an outline of reactor technology targets for the DEMO reactor are described below. Current achievement for each ITER technological target is also shown in each field. Furthermore, an overall achievement, estimated by taking importance of each achievement, simultaneousness of achievement, and various small subjects not described in this report, is also described for each field. Although the research and development activities aimed for the DEMO reactor components are just beginning, except for the materials and blanket development, the current achievements are also shown for these items. In these fields, great progress will be obtained if ITER is completed. If improvements follow this progress, completion of the DEMO reactor can be foreseen. Accordingly, the long-term R&D items, which should be performed in parallel with ITER, are limited to development of reactor materials, the blanket, superconducting conductors, and so on.

(1) Superconducting Magnet Technology (Fig. 3.2.3-1)

In ITER, a high-magnetic field, about 13 T, is required and the usage of Niobium-3 Tin as a superconductor is indispensable. This material has been already used in the tokamak device TRIAM-1M of the Kyusyu University and results of more than 15 years of operation have been obtained. ITER requires pulsed operation with a center solenoid coil with a 13-T magnetic field and 42-kA operating current conductor, which are expected to be more difficult to attain. Fabrication and assembly of a model coil has been completed and experiments with it have started. The overall achievement to date is 80%. Its significance is reaching and utilizing the ultimate limit of low-temperature technology, 4 K, in the academic field of cryogenics and superconducting technology , and is making an academic field that forms a bridge to the next high-temperature superconducting technological breakthroughs. As for the former low-temperature technology, realizing superconducting coil that could produce more than 16 T by using a low-temperature superconductor, and a refrigeration system, which could achieve higher operation efficiency (1/200) are the target of technology development required for the DEMO reactor. As shown in Fig. 3.2.3-2, development of high performance Niobium-Aluminum conductor and advancing development of 16-T magnet technology using Niobium-Aluminum are planned in parallel with the basic technology developments of magnets using high-temperature superconductors. The development of large-scale and high-efficiency components of the refrigeration system is aimed for the DEMO reactor. Research will continue to target further cost reductions of superconducting strands and also to develop high - performance high-temperature superconductors, such as alloys of bismuth.

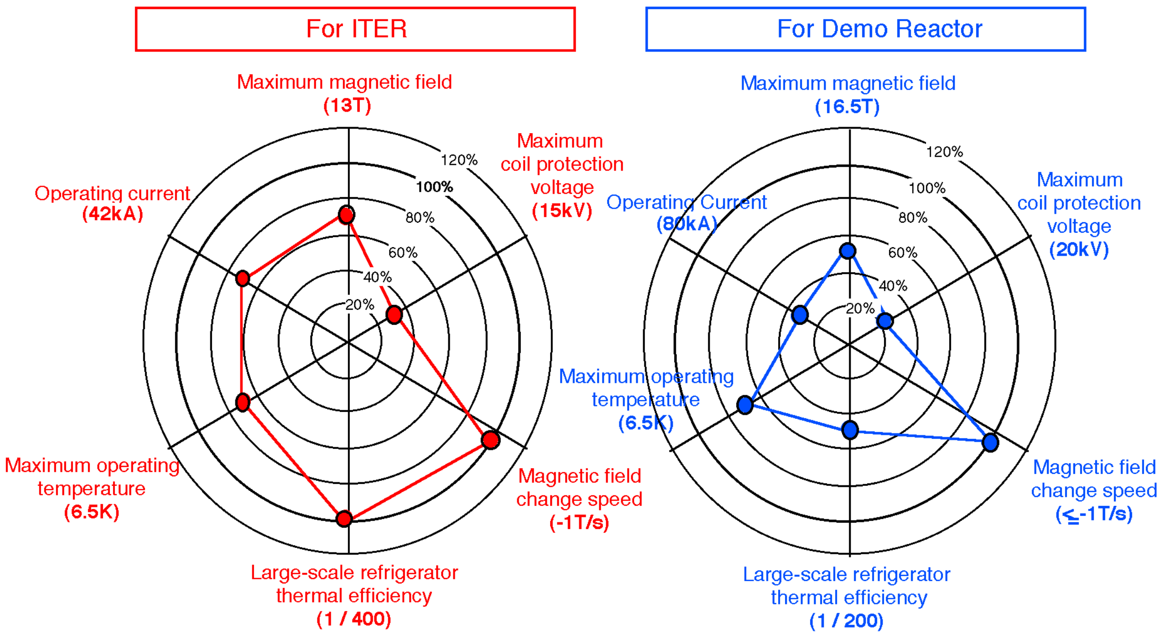


Fig. 3.2.3-1 Goals and Present Achievement Levels of Superconducting Magnet Technology

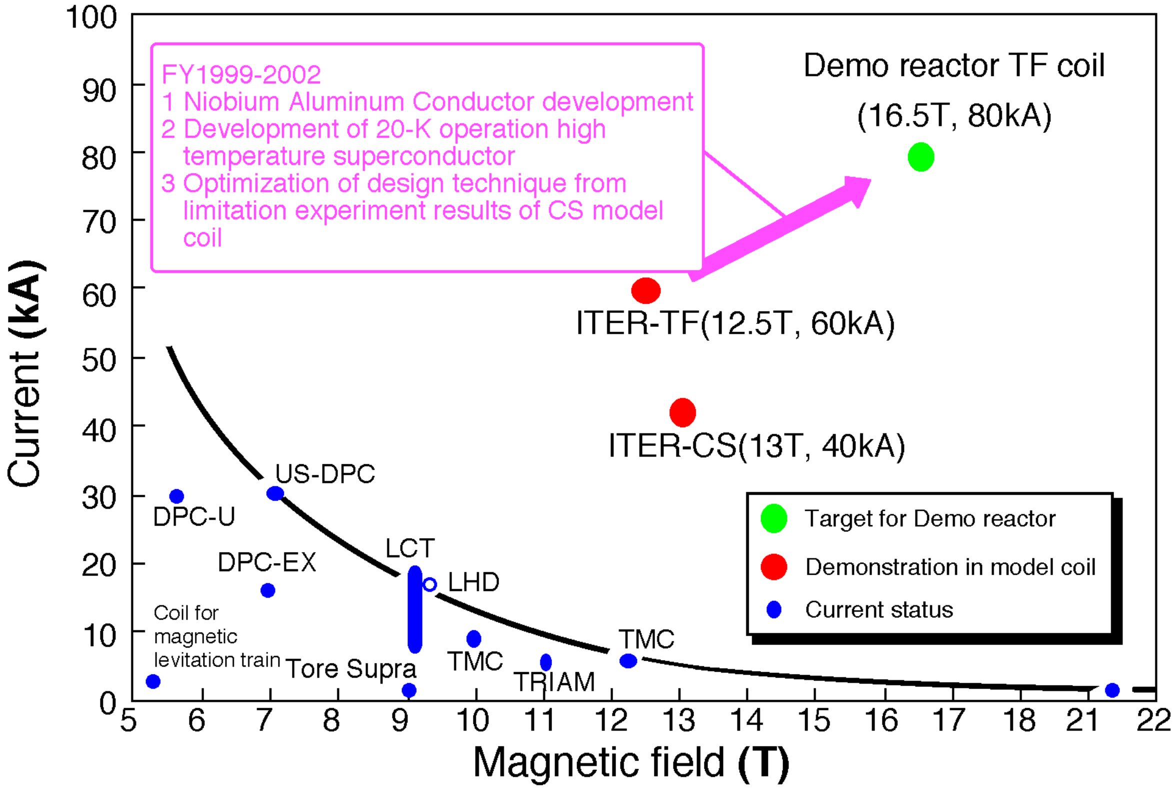


Fig. 3.2.3-2 Development Steps for the DEMO Reactor Coil Conductor

(2) Vacuum Vessel Technology (Fig. 3.2.3-3)

Using the technology developed for ITER, a full-scale sector model (9 m wide and 15 m high) that corresponds to a 1/20 sector of the ITER vacuum vessel, has been fabricated. The dimensional accuracy of within ± 3 mm and ± 10 mm has been successfully achieved for the sector fabrication and for the sector assembly, respectively. The remote welding and cutting test of the sector model and port extension with the full-remote welding/cutting tool are planned as further activities. The present achievement of the vacuum vessel R&D to the ITER requirements is estimated to be 80%, based on the R&D results.

In addition to austenitenitic stainless steel, ferritic materials, which have high mechanical strength at high temperature and have low-activation characteristics under neutron irradiation, can be employed for structural material of the vacuum vessel for the demonstration reactor. In any case, the reduction of electromagnetic (EM) force by the development and use of an electrically insulated structure is one of the critical issues of the vacuum vessel. For the reduction of EM force, the development of a function gradient material (FGM), which has high mechanical strength obtained by the bonding of a ceramic insulator and ferritic steel, is an essential R&D task. The insulation structure can be applied to the field joints of a vacuum vessel and cooling pipes for the reduction of the EM forces. Regarding the development of the insulated joint, the trial fabrication of a joint made of stainless steel and zirconia has been successfully completed, and it showed no significant degradation of insulation characteristics after neutron irradiation. For the development of an insulated joint made of ferritic steel and zirconia, the optimization of bonding conditions and an evaluation test under neutron irradiation should be done.

Advanced ferritic steel and SiC/SiC composite material are promising candidates for the structural material of the fusion reactor vacuum vessel. For advanced ferritic steel, the reduction of fabrication costs can be realized by using a shielding structure fabricated by cold isostatic pressure (CIP) bonding of chipped material. For the SiC/SiC composite material, which has good electrical insulation qualities and low-activation characteristics under neutron irradiation, the consistency of the structural design with the blanket structure should be considered to satisfy the required toroidal electrical resistance.

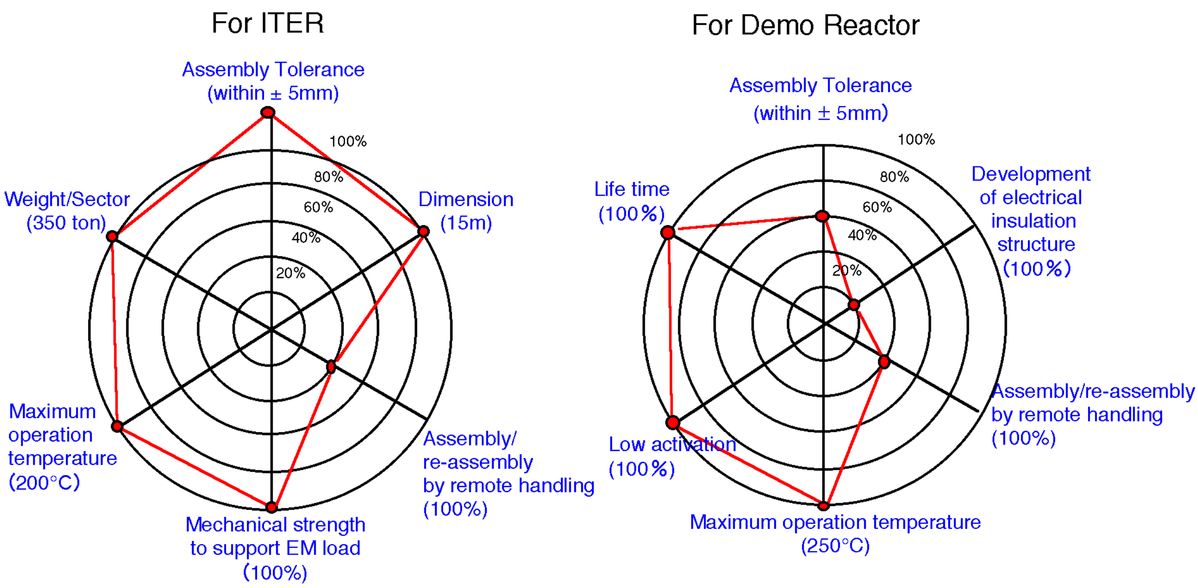


Fig. 3.2.3-3 Goals and Present Achievement Levels of Vacuum Vessel Technology

(3) Divertor and high heat-flux component technology (Fig. 3.2.3-4)

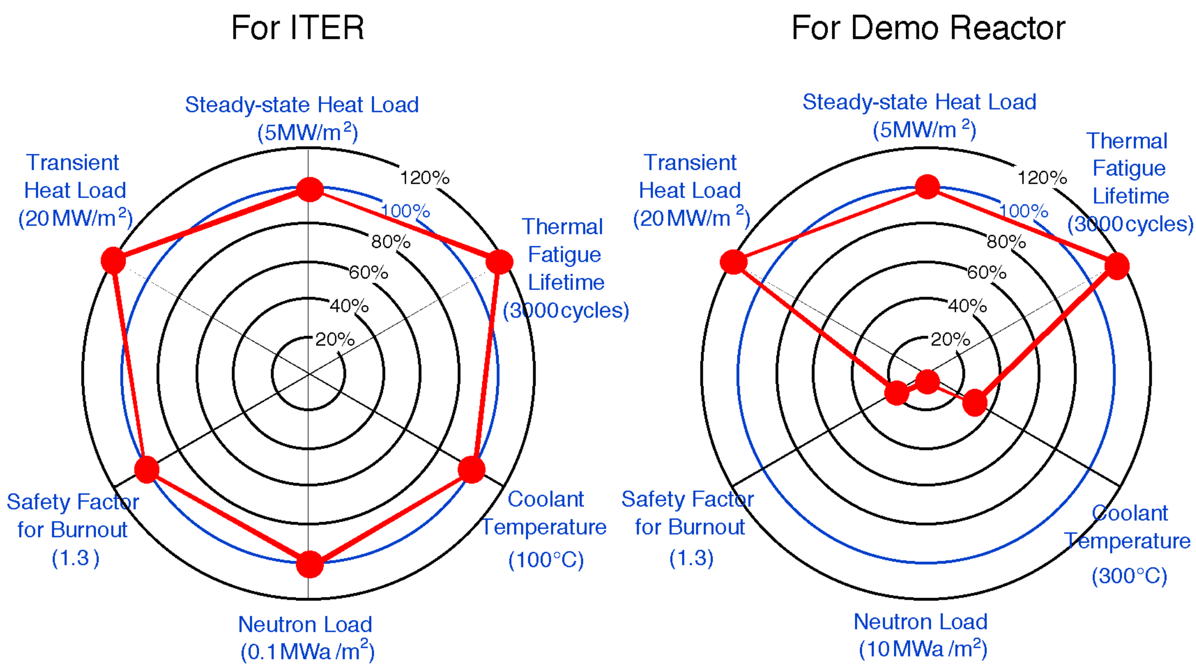


Fig. 3.2.3-4 Goals and Present Achievement Levels of Divertor Technology

The ITER divertor has been successfully developed with elemental components that can withstand heat loads of 5 - 20 MW/m2 with a coolant temperature of 100 - 150ºC at a neutron fluence of 0.1 MWa/m2. Initially, this development was considered to be difficult. Components that can endure the cyclic high-heat loads expected in the ITER divertor have been successfully demonstrated. An overall demonstration test is going to be carried out with a divertor cassette body, which will include thermal flow tests. The degree of the overall achievement can be evaluated as 85%.

In a demonstration reactor (DEMO), the neutron load will be ten times higher than that in the ITER. In addition to this, the divertor, which is a part of the energy generation system as well as the blanket, is required to be able to remove high heat loads at a coolant temperature of more than 300ºC. From the standpoint of the neutron load, reduced-activation ferritic steel is the most promising structural material. Heat removal technology with this material should be developed to accommodate divertor heat loads. As a buck-up of the reduced-activation steel, the development of a copper alloy, which has better durability against the neutron irradiation is being considered.

For an advanced divertor system, a vanadium alloy with a liquid-metal cooling system or the SiC composite with a helium-gas cooling system should be developed, as an option. The steps of divertor development are shown in Fig. 3.2.3-5.

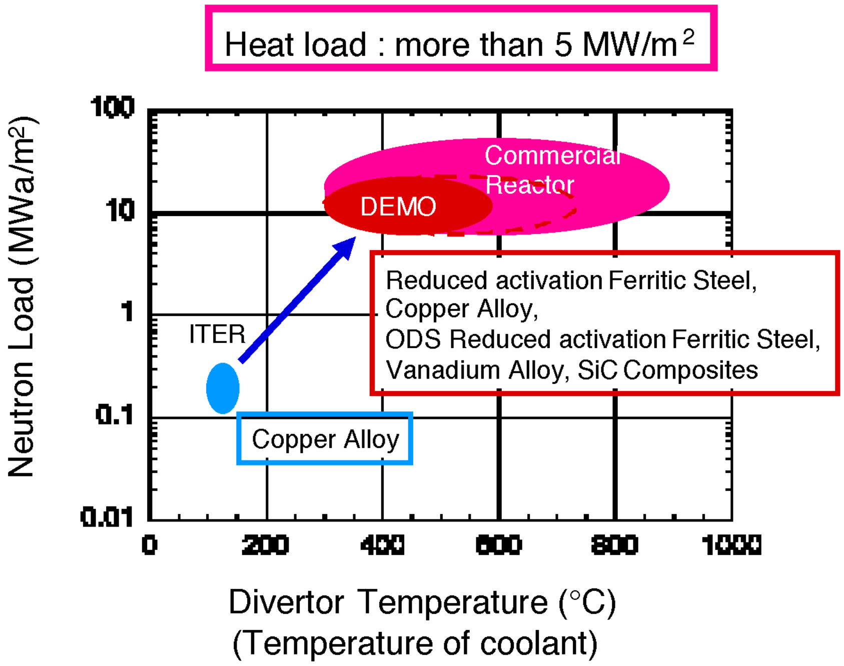


Fig. 3.2.3-5 Divertor Development Plan

(4) Blanket technology (Fig. 3.2.3-6)

The fabricability of a full-scale ITER Shielding Blanket has been demonstrated by using the high-temperature iso-static pressure (HIP) bonding technique. Also, structural integrity tests of a medium-scale shield blanket mockup were performed. The mockup withstood the surface heat-flux condition of ITER. With respect to the breeding blanket, neutron irradiation and tritium release tests were performed with one candidate breeder material, Li_2TiO_3. Sound tritium release behavior was obtained. With respect to the neutronics research for blanket development, the neutron streaming effect through various openings in the blanket and the peaking of the neutron flux due to neutron permeation effect through the non-homogeneous structures of the blanket have been evaluated. The overall achievement is 75%.

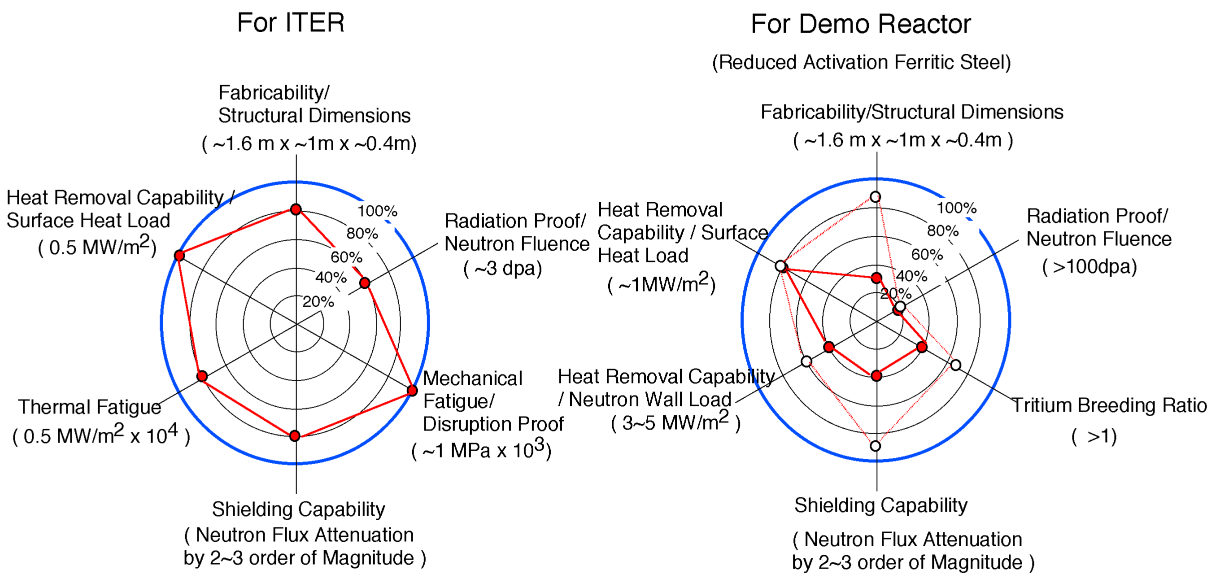


Fig. 3.2.3-6 Goals and Present Achievement Levels of Blanket Technology

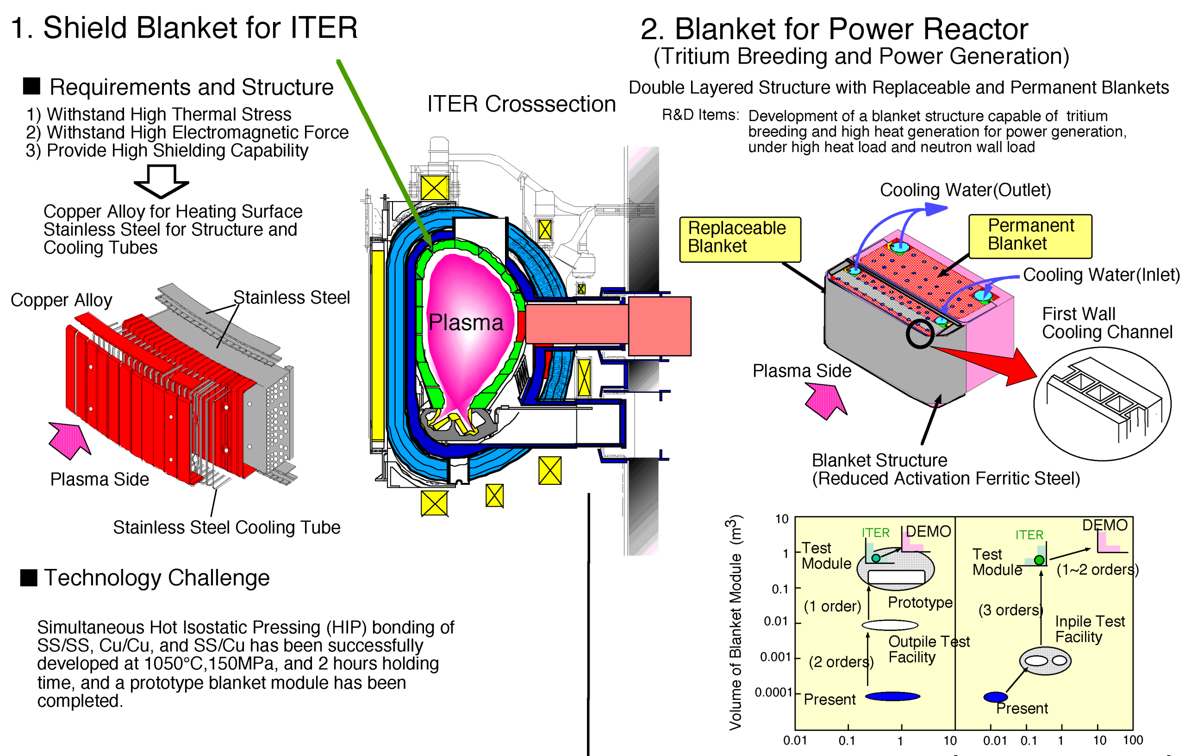


Fig. 3.2.3-7 Blanket development steps

In the DEMO reactor, it can be expected that the high heat flux and neutron wall load can be withstood by applying the reduced-activation ferritic steel (dedicated structural material for high-temperature use) and swirl tubes for the cooling channels of the first wall. Also, it is an important issue to develop the blanket structure, which withstands severe electromagnetic loads and heat flux during plasma disruptions, while incorporating the plasma control technology. Development strategy of the blanket is shown in Fig. 3.2.3-7. Specific issues of blanket technology for the DEMO blanket are discussed in the Section 3.3 along with the development of materials.

(5) Remote Handling Technology (Fig. 3.2.3-8)

With regard to the ITER blanket and divertor remote handling R&D, the full-scale remote handling systems, i.e., the “rail-mounted vehicle manipulator system” for blanket maintenance and the “floor vehicle transport system” for divertor maintenance, have been developed. By using them, the remote replacement of the weight of 4 tons for the blanket and 25 tons for the divertor have been successfully demonstrated. Welding, cutting, and inspection tools and the in-vessel viewing system also have been developed, and their performances have been verified. In addition, improvement of the components and elements the remote handling system, such as optical components, cables, and insulators, which lacked strength against radiation, has progressed significantly. The present achievement of the remote handling R&D to the ITER requirement is estimated to be 70%, based on the R&D results.

The rail-mounted vehicle manipulator system for blanket maintenance, which was developed as ITER technology R&D, can be applied to blanket maintenance of the DEMO reactor. The maintenance time for all blanket modules using the vehicle manipulator system is estimated to be approximately 50 days, based on expected future progress of the technology. In order to increase the availability of the reactor, it is necessary to decrease the maintenance time and increase the reliability of the remote handling system. Therefore, as an additional approach to minimize the maintenance time, the development of radiation-resistant components, such as the radiation-resistant battery and signal transmitter for wireless control, is also required to begin so these advances can be applied to wireless maintenance operations. Wireless control does not require control cables interfaces for blanket maintenance of the ITER, so both reliability and safety can be improved. For improvement of the durability and reliability of the remote handling systems, the development of rescue equipment to cope with accidents and troubles of the remote handling systems and development of radiation-resistant components will be continued.

After development, the technology described above can be applied as a common technology to the maintenance of the fission reactor and remote handling tasks in space. A scheme for the horizontal replacement of the blanket modules by extracting them together as a torus sector, which contains a large number of blanket modules between the TF coils, may be a candidate to reduce maintenance time. Therefore, this scheme will be also studied to compare with the present ITER scheme of the module replacement by the vehicle manipulator.

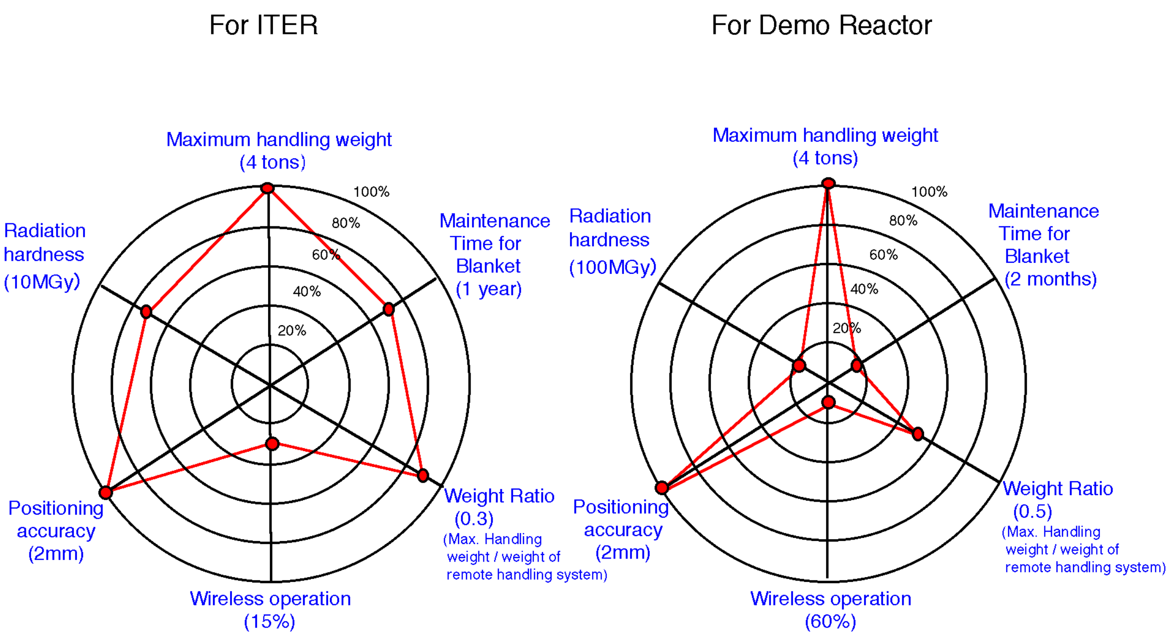


Fig. 3.2.3-8 Goals and Present Achievement Levels of the Remote Maintenance Technology

(6) Heating and Current Drive System

The heating and current drive (HCD) system for ITER is required to inject 50-MW of power. The system consists of a radio frequency (RF) HCD system and a neutral beam injection (NBI) HCD system. It is essential to develop a 170-GHz, 1-MW RF source, i.e., gyrotron, for the RF HCD system. A high-power gyrotron, which uses a synthetic diamond window, has produced 170-GHz, 500-kW RF power for 6 seconds. The output power of the gyrotron used in JT-60 has reached 1 MW for 2 seconds. The window made of artificial diamond has sustained a 10 bar pressure difference, although it has not been irradiated (by radiation). The overall achievement factor is 70%.

In the development of the NBI system, negative ions are accelerated to an energy of 1 MeV, which is the required energy in ITER, by using a multistage electrostatic accelerator. The high-power negative-ion source of JT-60 has produced 400 keV, 13.5A deuterium negative-ion beams. The overall achievement factor is 85%.

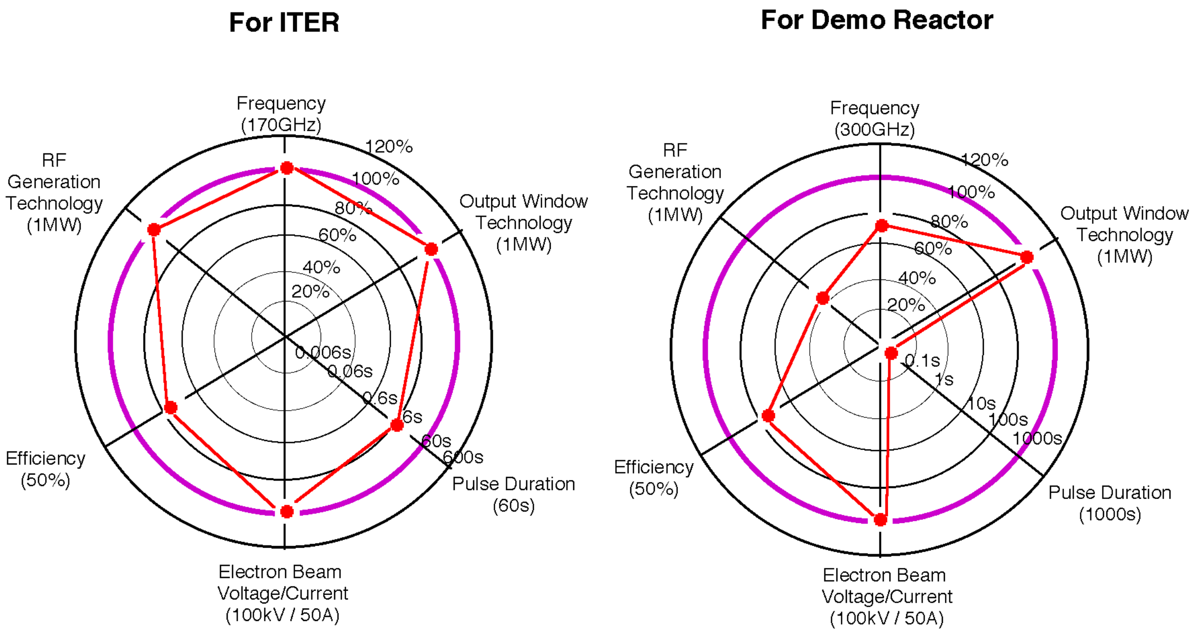


Fig. 3.2.3-9 Goals and Present Achievement Levels of the RF Heating / Current Drive Technology

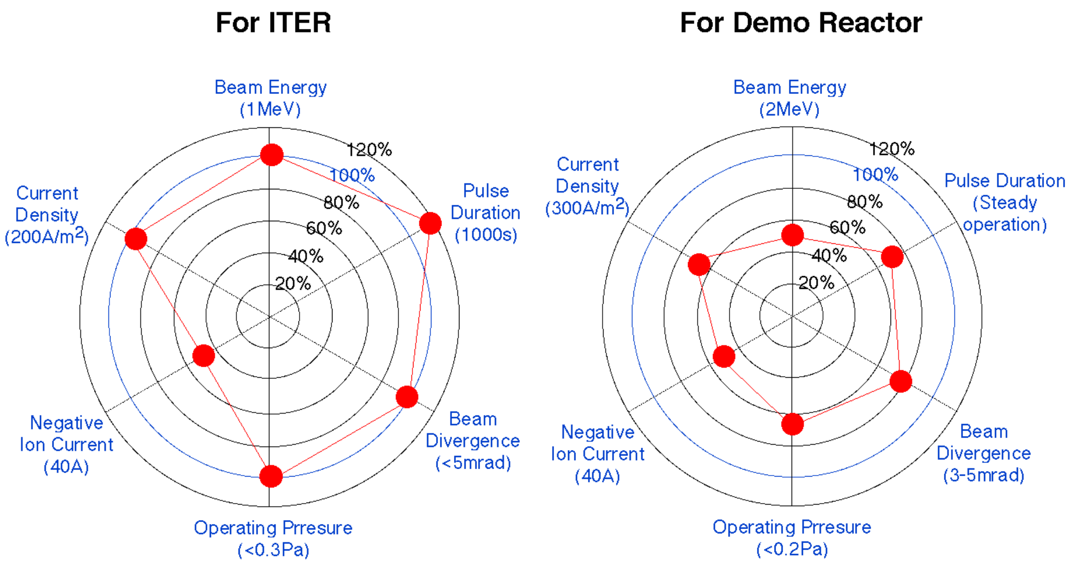


Fig. 3.2.3-10 Goals and Present Achievement Levels of NBI Heating / Current Drive Technology

One of the major issues for the future DEMO reactor is to increase the frequency to 300 GHz, corresponding to an increase of the magnetic field for the plasma confinement. To realize the 300-GHz RF HCD system, it is necessary to develop a resonator that enables higher frequency oscillation and to improve the diamond window. At the same time, it is necessary to develop a frequency variable oscillator and a coupling system to simplify the system and to improve the reliability, the efficiency, and the lifetime.

In the NBI system, the major issues for the future fusion reactors are to develop a maintenance-free negative-ion source to improve the reliability and the availability of the system, to increase the beam energy to 2 MeV, and to develop a plasma neutralizer that gives higher system efficiency.

(7) Tritium Processing and Safety Technologies

In tritium processing technology, a fuel clean-up system for the plasma exhaust gas has been developed. The system was demonstrated to successfully meet the ITER requirement of 10^7 as the detritiation factor. A 1/30-scale ITER fuel circulation model system, which includes the major fuel processing systems needed for ITER, was constructed and its successful cyclic processing operation proved the feasibility of the ITER fuel circulation system concept. To complete the design of the ITER fuel circulation system, an extended experiment campaign is being carried out with this model system to investigate the performance of components and systems in detail and to examine the control technique. Figure 3.2.3-12 shows the development status of tritium processing technology for ITER. Concerning the transportation container for a large quantity of tritium, a design study concluded that the container for 250 g of tritium could be realized on the basis of the present 25-g tritium container.

In tritium safety technology, as part of the necessary system for tritium accountancy in ITER, a process gas analysis system that can measure the composition of gases more than 10 times faster than usual, and a self-accounting tritium storage bed were developed. Satisfactory performance of both devices was demonstrated. Investigation of the performance of these devices under ITER operational conditions is being carried out for the completion of the ITER design. For the processing of components and parts contaminated with tritium, research and development work on an efficient detritiation method is in progress. To complete the design of ITER tritium confinement and removal systems, research on making clear the detailed tritium behavior in the atmosphere and on the wall and verification tests on the performance of confinement and advanced tritium removal systems are underway. An overall achievement status is 80%.

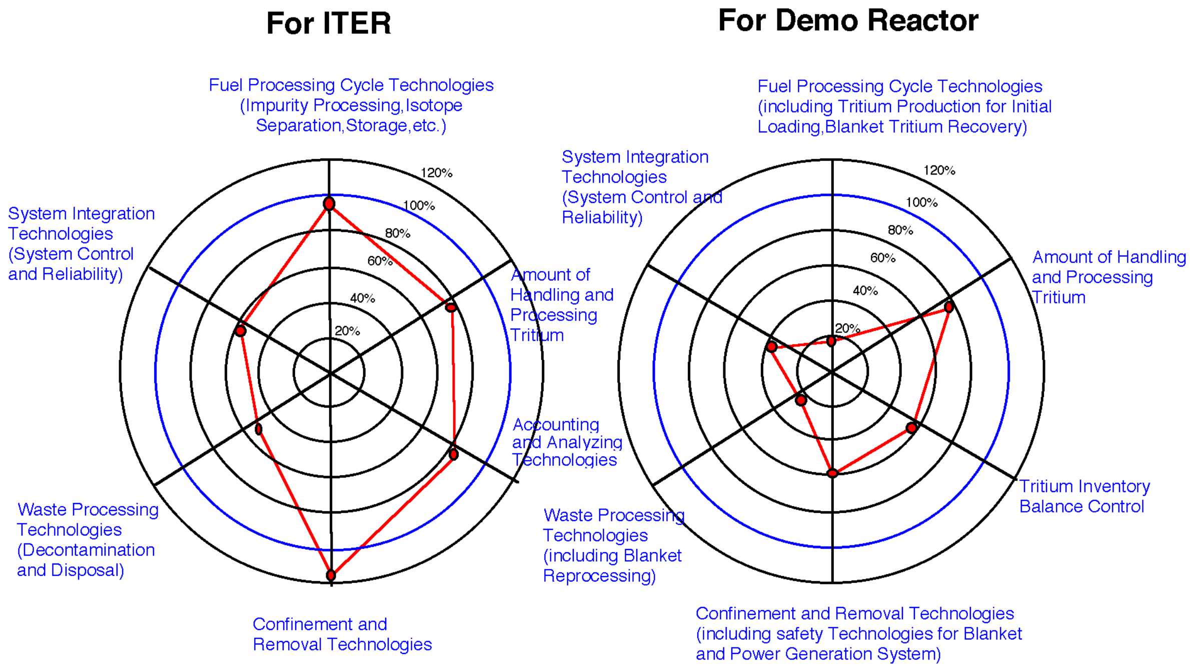


Fig. 3.2.3-11 Goals and Present Achievement Levels of Tritium Technology

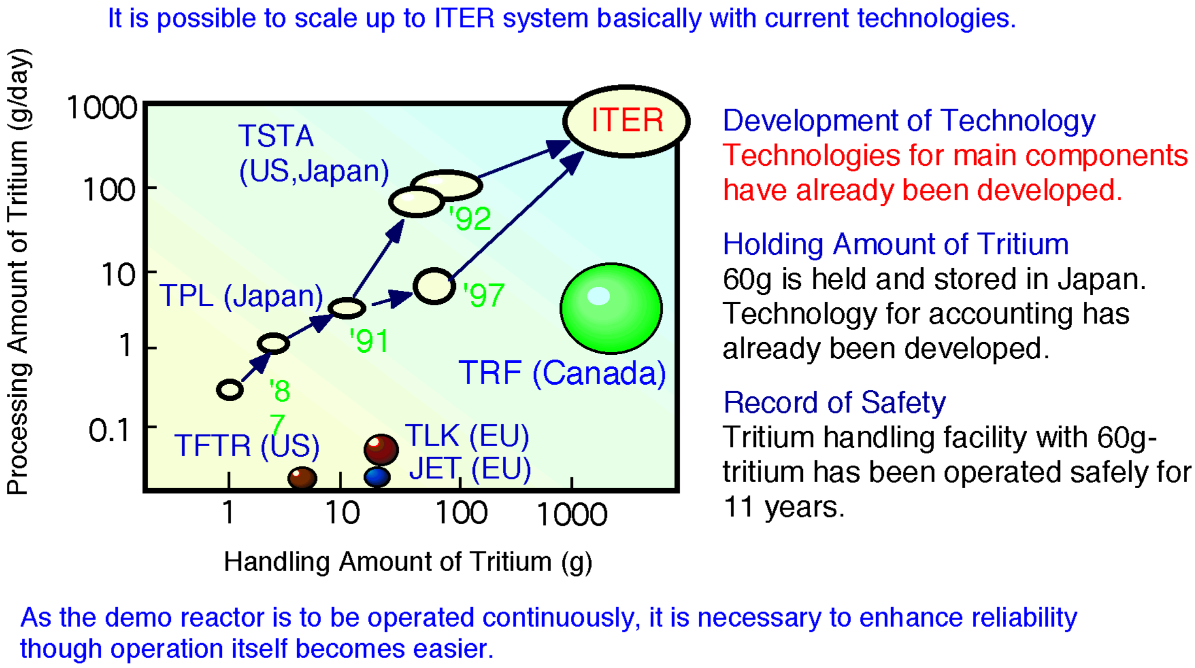


Fig. 3.2.3-12 Steps in development of tritium supply and handling technology

For the development of the DEMO reactor, key issues are reliable tritium processing for steady and continuous long-term reactor operation, safety for the power generation plant, production and security of necessary amounts of tritium, and efficient tritium removal/recovery from contaminated wastes.

For steady and continuous long-term operation, a fuel circulation system for steady continuous tritium processing and a continuous tritium removal/recovery system from the coolant are newly required. To establish technologies for these systems, basic research and development work is being continued. Necessary information for development of actual systems, i.e., long-term reliability and so on, is to be obtained through the development and the operation of the fuel circulation system and the tritium removal/recovery system in the ITER project.

Tritium safety technologies for components and systems, which are new requirements for power generation in the DEMO reactor, are to be developed in the ITER blanket module test program.

To produce and secure the necessary amounts of tritium, technologies for tritium recovery from blanket and of tritium production for initial loading must be established. Technology to recover tritium from the blanket is to be developed in the ITER blanket module test program. For tritium production, basic research and development work is to be continued on a small scale, but larger-scale work will commence when needed.

Due to the increase of tritium contaminated wastes, research and development work on the technology of tritium removal/recovery from wastes will be continued.

(8) Fueling and Vacuum Pumping Technology

In a fusion reactor, it is necessary to develop a technology for the peripheral injection of fuel, i.e., gas puffing and injecting pellets at slow speed. Moreover, the development of central fueling by pellet injection at high speed and a compact-toroid-plasma injection at ultra-high speed to supply fuels (deuterium and tritium) effectively and continuously. The gas puffing technique for ITER has satisfied requirements (response time, supply pressure, etc.) of ITER fueling. In the pellet injector, technologies of continuous pellet-production and successive acceleration of pellets are very important issues of development. Up to now, continuous production of pellets has been realized for 3000 sec by the screw extruder device, but the required pellet speed, repetition rate, and injection time have not been achieved yet.

In vacuum pumping technology, which continuously exhausts impurity gases such as helium from core plasmas, the development of vacuum pumps having robustness to environmental high-temperature, high-magnetic fields and radiation is indispensable. In addition, the development of a vacuum leak detection method is necessary. The vacuum pumps for fusion reactors are classified as mechanical pumps and cryopumps (cryogenic pumps). The mechanical pump with metallic rotors requires a magnetic shield, which is indispensable for operation under a high-magnetic field. There is, however, some merit for the use of mechanical pumps; gases can be continuously exhausted and the tritium inventory is significantly lower than that of the cryopump. On the other hand, the cryopump has merit in that it is not disturbed by a magnetic field. However, due to its pumping nature and gas accumulation, the pump operation has to be stopped periodically for regeneration of the cryopanel.

In ITER vacuum pumping technology, both the cryopump and the mechanical pump have been developed. Especially in the development of the helical-grooved pump, one type of mechanical pump, pump performance of the hydrogen outlet pressure (1000 Pa) has greatly exceeded the ITER goal value (200 Pa). The development of a vacuum leak detection method that can be used in a fusion reactor is being implemented as an ITER-R&D task. The overall achievement level of fueling and vacuum pumping technology is 65%.

Since the fusion reactor after the DEMO reactor will be operated continuously, uninterrupted fueling and vacuum pumping are also required. To meet this requirement, the performances mentioned above should be further improved for the fueling and vacuum pumping systems. In fueling technology, the advanced electromagnetic-acceleration technology for injecting fuel pellets and compact-toroid-plasma at high speed to a core plasma should be developed to reduce the tritium inventory of the reactor wall and the nearby components, and to improve the fuel injection efficiencies. Besides, an overall control technology of the fueling system should be established through improving the reliability and durability of the ITER related technology. On the other hand, in the vacuum pumping technology, a large throughput, ceramics turbo pump containing a ceramic (Si_3N_4) rotor, gas bearing, and gas turbine should be developed to reduce the tritium inventory. Furthermore, reliability and durability of the pumping system should be improved by simplifying the system.

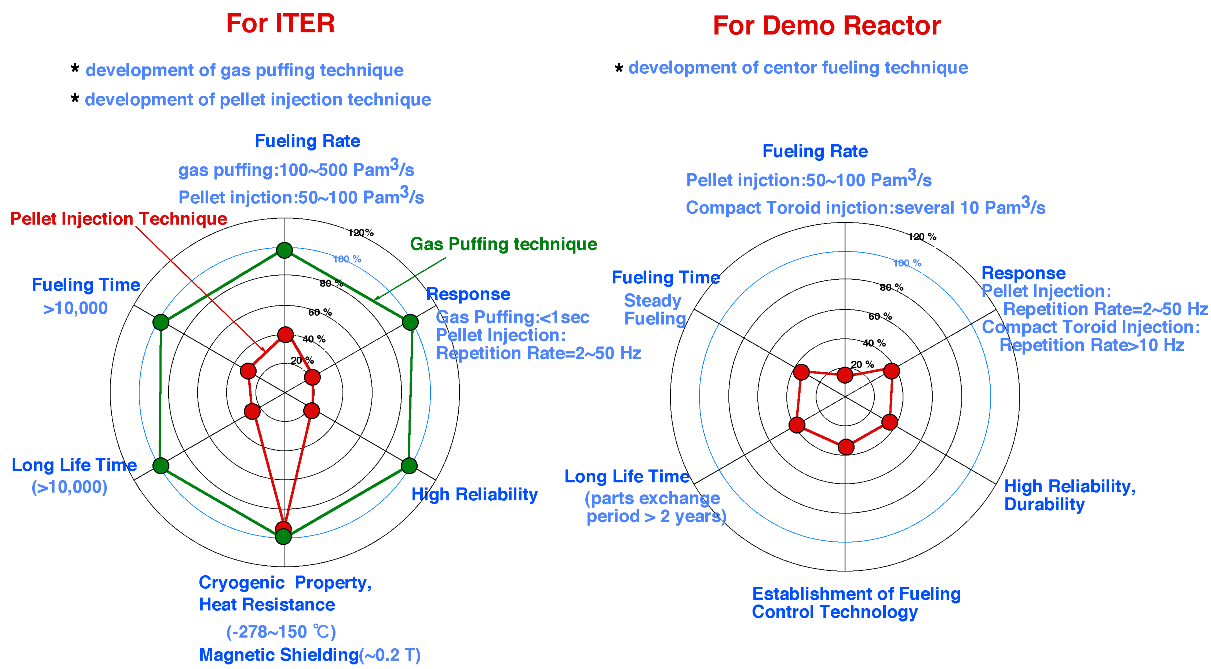


Fig. 3.2.3-13 Goals and Present Achievement Levels of Fuel Injection Technology

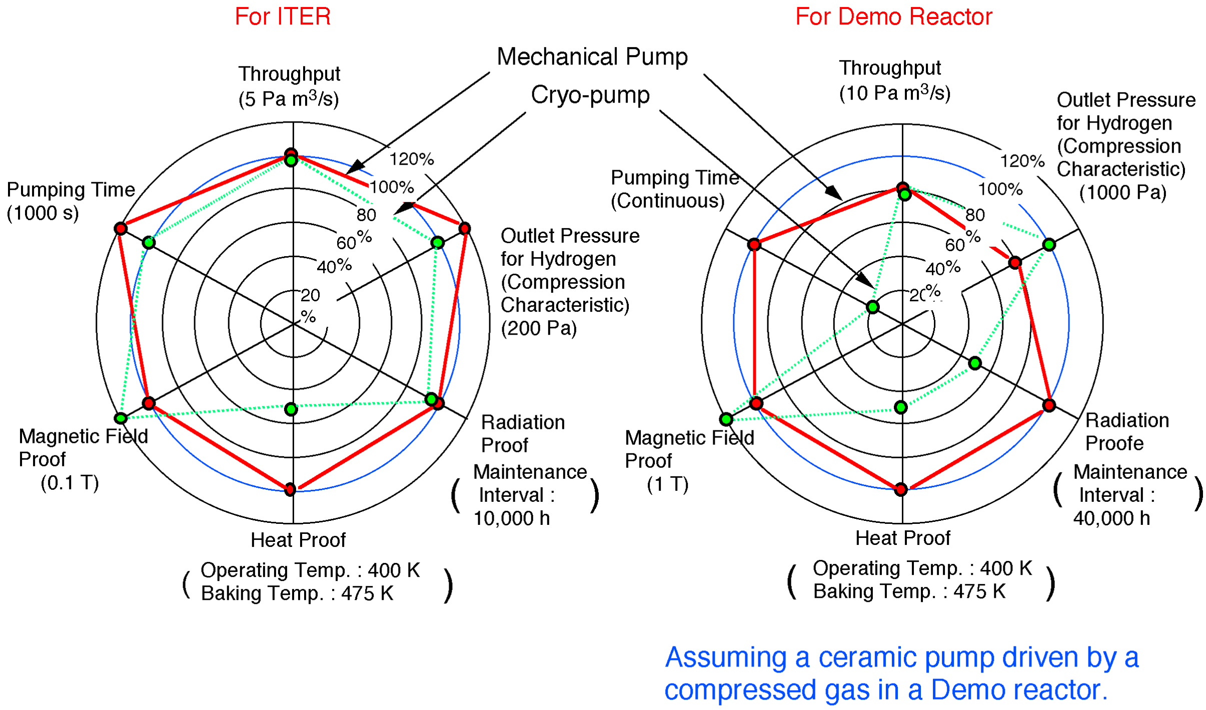


Fig. 3.2.3-14 Goals and Present Achievement Levels of Vacuum Pumping Technology

(9) Diagnostics Technology

A diagnostics technology is essential and important for safe operation of a fusion reactor by monitoring fusion power and plasma-position/shape control, etc. It is necessary to develop diagnostics technology that has high reliability, long life, and radiation resistance in an environment with high-level neutrons and gamma rays. Issues of development are diagnostic elements such as ceramic insulators, optical elements (reflectors, windows, optical fibers, etc.), electric cables, sensors (magnetic probe, bolometer, etc.). The development of prototype diagnostic sensors, vacuum seals for diagnostic windows, optical fiber feedthroughs, electric cable feedthroughs, etc., are also very important.

Neutron and gamma-ray irradiation tests of ITER diagnostic elements have been carried out. Except reflectors and optical fibers in the ultra-violet wavelength region, most diagnostic elements can be used without being exchanged in the basic performance phase of ITER (the number of discharges--10,000 shots). The prototypes of vacuum seals for diagnostic windows and 52-channel optical fiber feedthroughs have been developed, all of which successfully withstood a pressure difference of 5 atm., a temperature of 220°C, and an acceleration of 15 G. A synthetic diamond detector with high energy resolution for neutron measurement has also been developed. The development of each diagnostic system will be executed during the ITER construction phase because there are many issues related to the development of diagnostic. The overall achievement level is about 65%.

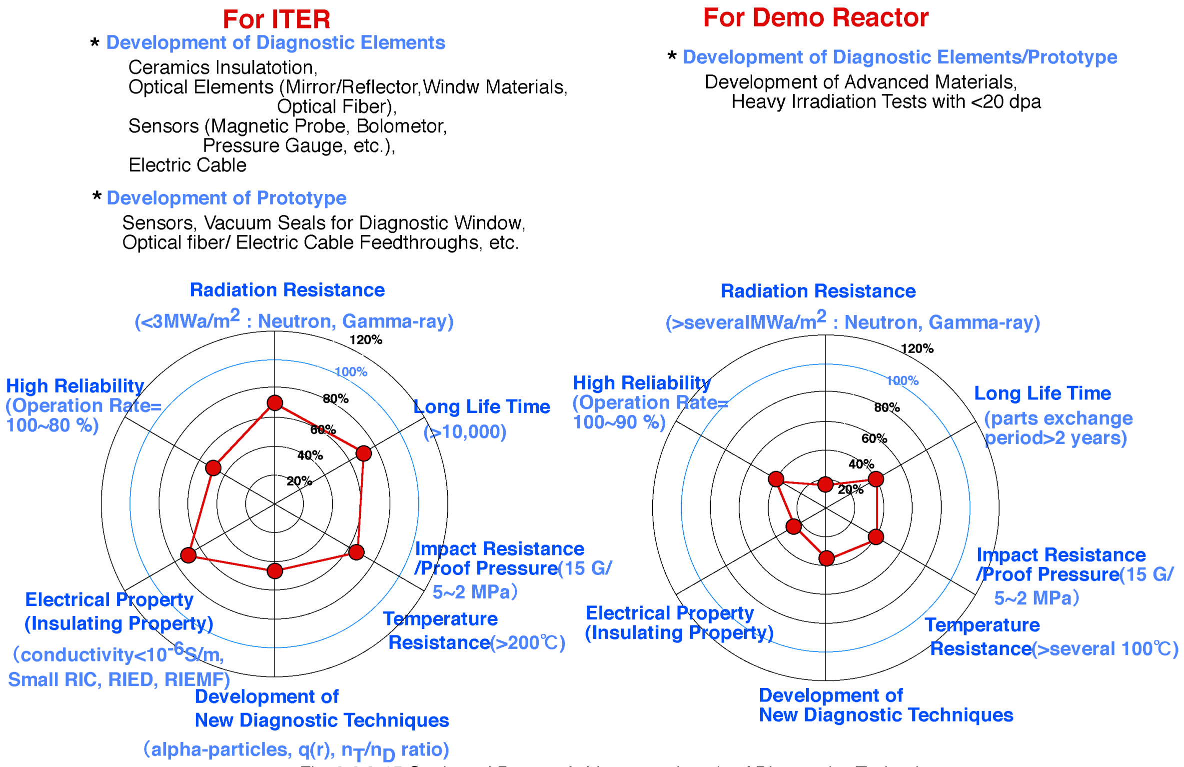


Fig. 3.2.3-15 Goals and Present Achievement Levels of Diagnostics Technology

For and after the DEMO reactor, it is necessary to develop new advanced materials and to carry out the heavy irradiation tests of conventional and advanced materials to an atomic displacement of 20 dpa (goal of reactor) using a fission reactor and a strong neutron source. The diagnostics and control systems with long lifetime and high reliability will be realized by development of diagnostic elements and prototypes which are able to resist neutron fluence of several MWa/m^2. By accumulation of ITER plasma operation experiences, the composition of diagnostics systems for the DEMO reactor will optimized and effective operation/control techniques of the reactor plant will be mastered using a minimum of diagnostics systems with high-radiation resistance and reliability.

(10) Safety Technology (Fig. 3.2.3-16)

Regarding coolant ingress or the loss of vacuum, which would result in the release of the radioactive materials contained in the plasma vacuum vessel, an analysis model has been validated through extensive scaled experiments simulating the ITER vacuum. Based on this basic study, a possible concept for radioactive dust removal has been proposed. The structural design guidelines including welding and inspection methods have been developed through a review by design code experts to assure the structural integrity of the vacuum vessel will remain intact, since the vacuum vessel is the most important structure of the primary containment boundary of radioactive material. In parallel, the engineering data for qualification of the vacuum vessel integrity is also being accumulated from mechanical tests and structural analyses. Furthermore, the design guideline for seismic isolation has been outlined, together with the engineering data from sub-scale experiments and analyses of rubber bearings and the dynamic responses of the tokamak components. In accordance with the above achievements, applicable design codes would be finalized for application to the actual ITER component/facility fabrication. As a whole, the level of technology development concerning safety and regulation would be considered to be 60%.

For the DEMO reactor following ITER, further technology developments are required, including improvement of the safety system reliability for abnormal events of the cooling system due to high coolant temperature, high heat flux, and high neutron flux for power generation, and the improvement of social receptivity of this program by rationalization and passiveness. Details of these items are described in Section 3.4.

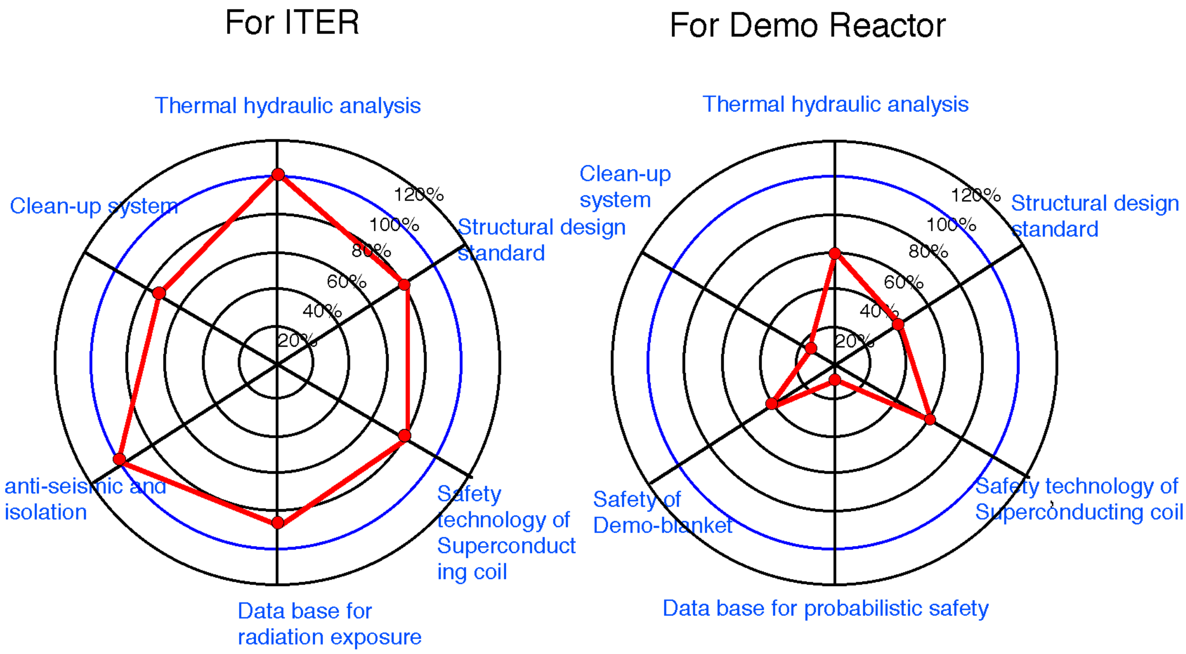


Fig. 3.2.3-16 Goals and Present Achievement Levels of Safety Technology

(11) Materials development (Fig. 3.2.3-17)

For the experimental reactor ITER, 316LN has been selected to be the structural material for the first wall of the blanket. An important issue of the degradation of ductility by irradiation is expected to be managed by the accurate evaluation of the irradiation effects and the improvement of the design method. Evaluation of failure by stress corrosion cracking and thermal fatigue are being carried out. The re-welding conditions for maintenance and exchange of components have almost been established.

Austenitic stainless steel similar to that for the blanket is planned for use in the vacuum vessel. Evaluation of the radiation effects to the basic properties of the alloy has been finished, and the effects of the irradiation on weldments and stress corrosion cracking are now being evaluated.

Tritium breeding materials lithium oxide (Li_2O) and lithium titanate (Li_2TiO_3) have been examined, as well as beryllium as a neutron multiplier. Fabrication methods of the pebbles for both breeding and the neutron multiplier have almost been established, and the irradiation properties of the pebbles are now being evaluated. The level of the overall achievement of the materials development for ITER is evaluated to be 80%.

Development of reduced-activation ferritic/martensitic steel, SiC/SiC composite, vanadium alloys, and other materials for the DEMO reactor are now in progress. Figure 3.2.3-17 shows the current level of development and the expected level of achievement for the reduced-activation ferritic/martensitic steel and for an austenitic stainless steel for ITER structural material. The materials required for and after the DEMO reactor will be discussed in Section 3.3.

Service temperatures for the materials of the vacuum vessel differ depending on the coolant and the blanket temperature. The reduced activation ferritic/martensitic steel will be used for the blanket structural materials of the DEMO reactor. Materials for the vacuum vessel of this reactor will be selected to be the reduced-activation ferritic/martensitic steel or a low-nickel austenitic alloy (major components would be Fe, Cr, and Mn). The structural design of the vacuum vessel will likely be based on the method used for the blanket design, since the latter seems to be applicable to the former.

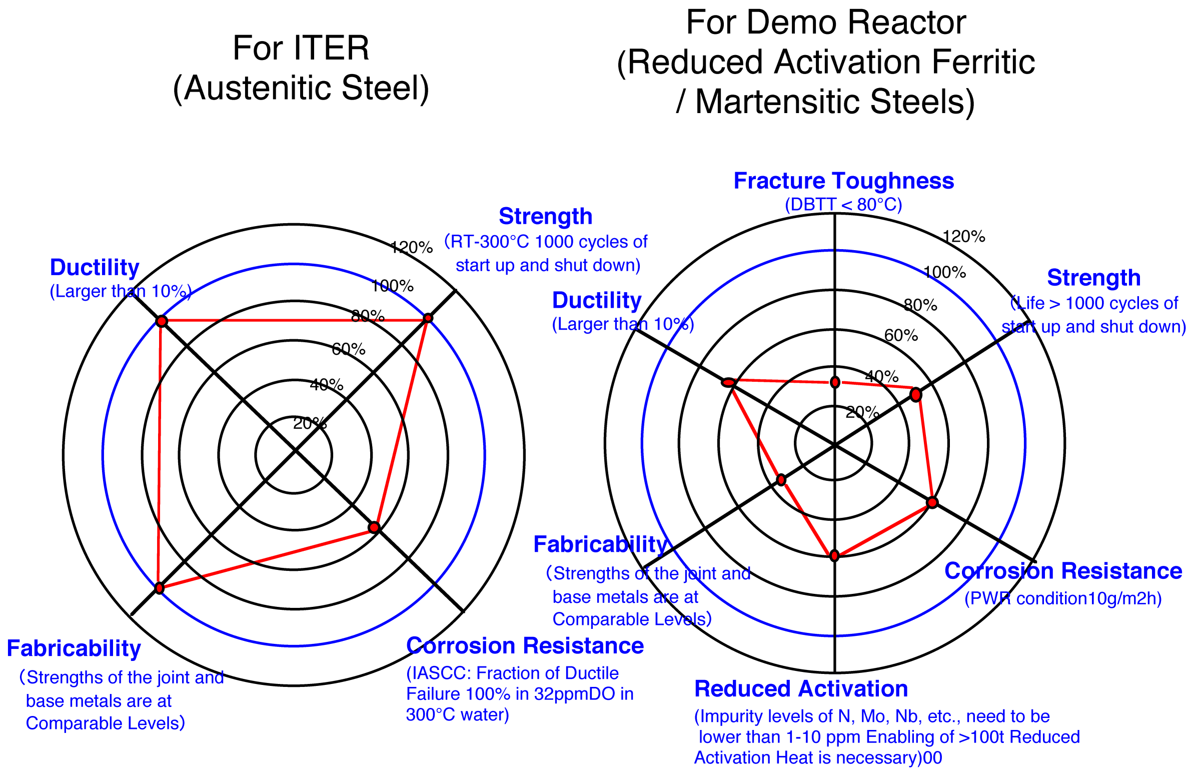


Fig. 3.2.3-17 Goals and Present Achievement Levels of the Structural Materials Technology

3.3 Current status and future issues of blanket and material development

3.3.1 Status and issues of blanket development

The power generation blanket, which is a main component in the DEMO reactor (the DEMO breeding blanket), is required to have excellent characteristic for high tritium production and tritium release, temperature control by high-temperature coolant, and to provide neutron and gamma ray shielding. In addition, it is also required to have high reliability and safety, environmental compatibility, and economic attractiveness. The following is a detailed discussion of the blanket's specifications and characteristics.

1) Tritium production and release characteristics

The power generation blanket needs to generate and release fusion fuel, tritium, to allow the DEMO reactor to self-supply itself with fusion fuel by using reasonable scale tritium recovery system.

2) Temperature control characteristics by the high-temperature coolant for electricity generation

The power generation blanket needs to have the ability to convert the kinetic energy of fusion neutrons to thermal energy and transfer this energy to the coolant with high efficiency and reliability. In particular, a higher temperature is desirable to achieve a higher efficiency of electricity generation.

3) Sufficient shielding characteristics

The power generation blanket needs to have sufficient shielding capability for the protection of the vacuum vessel, for the protection of superconducting magnets and surrounding components and for the bio-shield.

4) Long term durability of the blanket structure

The power generation blanket needs to withstand a high surface heat flux, the neutron wall loads, and strong electromagnetic loads. It also needs to provide long-term durability by withstanding high irradiation fluence, many operation cycles, and exposure to chemicals during operation.

5) High safety, reliability, and environmental susceptibility

The power generation blanket needs to have high reliability and a design margin so that it doesn't trigger an initiating event for accidents in an off-normal condition. At the same time, special effort should be directed at minimizing the potential hazards, such as the chemical energy and radioactivity contained in the power generation blanket. It is expected to reduce the amount of radioactive waste after the end of life by minimizing the radio-activation.

6) High economic factors

The power generation blanket needs to be operated with the coolant at the highest temperature possible to achieve high efficiency electricity generation, and to eventually become an economically attractive fusion reactor, i.e., an economically competitive source of electricity. At the same time, it is necessary to reduce fabrication costs, to recycle used breeder material, and so on. Also, it is important to reduce the duration of remote handling blanket maintenance to a minimum to increase the availability of the fusion reactor and, by so doing, to promote economy.

As stated above, high performance for the power generation blanket are required to be achieved under lots of hard conditions.

(2) Blanket type and the development status

Various types of power generation blankets have been proposed in past years. They are categorized into two types, one is a solid breeder blanket (water or helium cooled blanket), and the other is a liquid breeder blanket (liquid LiPb - water cooled blanket, liquid Li - self-cooled blanket, molten salt - self-cooled blanket) following R&D and design studies.

Past R&D included materials research including irradiation experiments, research on the kinetics of tritium release from the breeder material, and basic research on the fabrication techniques of breeder pebbles and multiplier pebbles. Thus, the R&D status has advanced to the engineering development phase in preparation for the coming testing of the ITER test blanket module. JAERI has placed priority on the solid breeder blanket type, the first candidate type, because of its higher reliability and safety, readily available database information, the certainty of achieving the DEMO blanket, and possibility of upgrading the performance in the future, and in consideration of the limited resources and time allowed by the ITER TBM testing schedule. The solid breeder blanket type has unique merit as follows.

1) Selection of solid breeder type

Basically the thermo-chemical activity of elemental material is small. Also, the tritium inventory can be kept relatively low. The database contains many applicable items of data related to tritium release on the tritium inventory and release behavior or irradiation performance of the solid breeder materials. At the same time, the basic technology for tritium recovery from a solid breeder blanket is already established.

2) Pebble bed utilization for breeder and multiplier layers

An issue of concern for the solid breeder blanket was potential fracture of the breeder and multiplier structure by irradiation and swelling. It can be expected that application of a pebble bed structure may reduce fractures by reducing the influence of the degradation of the thermo-mechanical properties. A pebble bed is an accepted structure for a solid breeder blanket as the EU applied the pebble bed type a few years ago.

3) Pressurized water as the candidate coolant

Commercial power stations have sufficient experience with pressurized water. Thus, the basic technology is already established.

4) Reduced-activation ferritic steel as the candidate structural material

Ferritic steel has superior characteristics for both irradiation and a wide range of high-temperature usage in industry. It has been demonstrated that radioactive waste management efforts will be reduced by the use of material elements having low-activation characteristics.

5) Possibility of performance upgrade

The solid breeder type blanket has potential for higher electricity generation efficiency and higher safety by upgrading to helium gas coolant and innovative structural materials, such as ODS reduced-activation ferritic steel, SiC/SiC composites. Such innovative materials have also the merit of reducing induced activation. Such upgrades do not require major changes in the design of the blanket.

Table 3.3.1-1 Major blanket types under development

type		Solid Breeder Blanket		Liquid Breeder Blanket	
Material	Breeder	Ceramic Breeder	Ceramic Breeder	LiPb	Li
	Structure	Ferrite	Ferrite	Ferrite	V Alloy
	Coolant	Pressurized Water	Helium	Pressurized Water	Liquid Li self-cool
Advantages		Safety Sufficient database Wide base of industrial technology	Safety Sufficient database High electricity generation efficiency	Less irradiation damage on breeder Breeder multiplies neutron	No irradiation damage on breeder Simple blanket geometry
Disadvantages		Irradiation damage Complicated configuration	Irradiation damage Complicated configuration Shielding performance	Tritium permeation Heavy breeder mass and high power for forced-flow Safety concern for liquid metal Less database	MHD Pressure drop Uncertain tritium recovery technology Safety concern for liquid metal Less database
Working Party		Japan	Japan, EU, RF, US	EU	RF, US

With respect to the liquid breeder blanket, the liquid Li self-cooled blanket and LiPb blanket (structural material: reduced-activation ferritic steel, pressurized water cooling) are the main options worldwide as shown in Table 3.3.1-1. Such a liquid breeder blanket has a major merit of not having irradiation degradation in the breeder material. (The LiPb breeder material has the issue of Po generation by nuclear transmutation and the Li-Pb fraction change in the course of breeding tritium from Li.) Also, a merit is that there is a less stringent high-temperature limit for the breeder material, which can result in a simple blanket structure. On the other hand, molten salt blanket has the similar merits to the liquid metal blanket. Additionally, a molten salt blanket has the advantage that it can reduce the MHD pressure loss, which is the major issue for a liquid blanket. Also, there is the possibility of reducing the chemical reactivity compared to the liquid Li blanket. For these reasons, a molten salt blanket is under intense development in preliminary tests and system design work by the NIFS of the Japanese Ministry of Education, Science, Sports and Culture.

(3) Major R&D status and future issues

Development issues of the breeding blanket are 1) fabrication technology development, 2) tritium breeding and recovery technology development, 3) cooling technology development, 4) durability development, including irradiation characteristics, 5) safety and environmental susceptibility development, and 6) economic enhancement through reduced cost development. The following are a description of issues on the solid breeder blanket and the liquid breeder blanket.

1) Fabrication technology development

- Fabrication technology of structural material:

  Ferritic steel has wide industrial usage and a candidate composition of reduced-activation ferritic steel (JLF-1, F82H, etc.) is already optimized. Ingot production of 5 tons has been demonstrated. Further development is needed to adjust the composition to meet specific mechanical strength requirements. At the same time, the accumulation of mechanical strength data with and without irradiation is expected.

- Blanket box structure fabrication including the first wall:

  Application of diffusion bonding to the fabrication of the blanket structure by F82H is under development. The major bonding parameters were already screened out and a demonstrative fabrication of the first wall panel mockup was done. High heat flux tests were performed using the fabricated mockup and these tests showed good thermal cycle fatigue characteristics. Further effort is needed for the optimization of bonding conditions and the accumulation of mechanical data on bonded materials. Also, various specific destructive mode tests such as the thermal creep tests for structure are needed. Development of reduced cost fabrication techniques is also needed.

- Breeder and multiplier pebble mass fabrication technology:

  Past research achieved the selection of the agglomeration method and the sol-gel method fabrication techniques for breeder pebble fabrication, and the rotating electrode method for multiplier pebble fabrication. Especially, the sintering density increase without grain growth by doping TiO_2 is a further issue for Li_2TiO_3 pebble fabrication technology development. Cost reduction technology needs to be developed.

2) Tritium breeding and recovery technology development

- Thermo-mechanical characteristics research for breeder and multiplier pebble bed:

  It is most important to maintain the breeder and multiplier temperature in the appropriate range not only from the view of proper tritium release but also to preserve the mechanical integrity of the pebble bed. Past pebble bed experiments have resulted the data accumulation for effective thermal conductivity of a pebble bed as well as the wall heat transfer coefficient. The mechanical characteristics of a pebble bed are a new area of research so the data accumulated and the theoretical research on the pebble bed Young's modulus, the Poisson ratio, and on the modeling of the mechanical behavior of pebble bed are all significant. The combined behavior of the thermal and mechanical characteristics and irradiation effects are further issues.

- Tritium generation and release characteristics:

  Tritium generation and release kinetics research are investigating purge gas conditions and the temperature dependence for tritium release by BEATRIX-II experiments under the framework of the IEA-IA, and breeder irradiation experiments in JMTR. Sound tritium generation and release characteristics have been demonstrated up to 5% Li burn-up (i.e., DEMO: 10 - 15%). Non-steady-state data is needed by simulated pulse irradiation. Model development is further needed for evaluating tritium release behavior. Higher Li burn-up experiments are needed to demonstrate the feasibility of tritium production in DEMO conditions.

• Tritium recovery and fuel cycle technology:

The technology of tritium recovery from large amounts of He purge gas has already been established by technological research and operational experience of the tritium systems test loops in TPL of JAERI and TSTA of LANL. Further development is necessary for scale-up testing, high efficiency process development, and so on.

3) Cooling technology development

• Coolant handling technology:

Pressurized water and helium cooling technology are already established by experience with PWR, BWR, and high temperature gas test reactors. There is no critical issue such as the MHD effect in liquid metal technology.

• First wall cooling technology:

For the first wall cooling channel structure, a mockup panel (about 10 cm × 20 cm) was fabricated by using reduced-activation ferritic steel F82H. This was tested using high-heat flux cycles of about 2.7 MW/m^2 (the surface temperature of ~500 °C corresponds to the operation temperature in the DEMO reactor) for the DEMO power generation blanket. The test successfully demonstrated a 5,000-cycle operation, including registering the same fatigue characteristics as the base material. Further experiments are planned to address the thermal creep performance using the specific structure used for destructive mode clarification. Also, the heat transfer of the built-in cooling channel of the first wall panel needs further improvement.

4) Durability development, such as irradiation characteristics

• General aspects:

To maintain the integrity of the power generation blanket in assumed operational conditions, it is necessary to certify the irradiation performance of materials, degradation of materials by thermal cycles and long-term operation, first wall durability in high-heat flux, and chemical effects (corrosion, mass transfer, and so on).

• Structural material:

Irradiation experiments in HFIR showed that the tensile strength could be maintained in more than 30-dpa irradiation (DEMO > 100 dpa). In low irradiation conditions, fracture toughness showed sound performance. Further investigation is needed for clarification of the He production effect and the hydrogen embrittlement effect.

• Breeder material:

By the BEATRIX-II experiment, the irradiation durability of Li_2O was demonstrated in 5% Li burn-up. By the out-pile thermal cycling tests, candidate breeder pebbles have withstood up to 10,000 operational cycles. Corrosion by contact with structural materials has also been investigated and shows acceptable corrosion rate data for blanket applications. Further investigation is needed for Li_2TiO_3 on the same research issues. It is necessary to investigate the irradiation effects on the thermo-mechanical characteristics further.

• Multiplier material:

As part of basic durability evaluation research, the Be oxidation rate and corrosion rate of contacting structural materials are being measured and formulated for design criteria clarification.

• Durability demonstration:

It is necessary to perform total integrity testing by using full-scale mockups in out-of-pile testing.

5) Safety and environmental susceptibility development

• General issue:

Important issues for this power generation blanket design are tritium inventory reduction, evaluation of off-normal performance, development of reduced-activation materials, and reduction and recycling of radioactive waste.

• Tritium inventory:

By adjusting breeder temperature within the proper range, the tritium inventory in the breeder material can be reduced to less than 1 kg, which is relatively small compared to the net tritium inventory.

• Reduction of induced activation:

Long-term activation can be reduced by the development of reduced-activation ferritic steel, as compared to austenitic steel.

• Off-normal performance evaluation:

A preliminary safety analysis for ITER Test Blanket Module was performed. The largest impact was caused by loss of coolant in TBM box; however, it was shown that the pressure suppression tank with the proper volume could mitigate the consequences. Further investigation is needed on hydrogen generation reaction between Be in contact with water in a high temperature environment. Also it is important to establish the safety assessment scenario for the DEMO rector.

• Innovative material development:

It is important to develop hazard resistive materials such as Be based intermetallic compounds or innovative structural materials.

6) Economically reduced cost development

• Remote handling technology:

Remote handling technology development is very important to increase the reactor availability, and this will affect the design of the hot cell facility, reactor building, and so on.

• Blanket replacement strategy:

Basic blanket technology was established by ITER engineering R&D. Further refinement of this technology is needed for application to the DEMO reactor. It has been proposed that “whole sector” replacement is a timesaving replacement method, however, assessment by detailed design is needed.

As stated above, many R&D results have been obtained to show the feasibility of a solid breeder blanket based on past material developments and irradiation experiments. Now is the turning point to launch engineering R&D or a demonstrative test program for ITER Test Blanket Module testing. With respect to the liquid breeder blanket, research work is in progress in the fields of development of V alloys as innovative structural materials, evaluation of the MHD pressure drop, development of tritium permeation barrier coatings, and a basic investigation of the compatibility between the breeder materials and the structural materials. On the other hand, the liquid breeder blanket has the following technical issues, therefore, systematic engineering R&D is indispensable for ITER TBM testing.

1) Liquid Li self-cooled blanket:

- Development of an electrical insulation coating to reduce MHD pressure drop
- Evaluation of heat transfer and hydraulic characteristics of liquid Li in a strong magnetic field
- Evaluation of compatibility between liquid Li and structural materials
- Establishment of safe handling techniques for liquid Li
- Development of industrial bases for V alloys and box structure fabrication technology
- Heavy irradiation data for V alloys

2) LiPb blanket

- Development of tritium permeation barrier coatings
- Evaluation of the corrosion effect of LiPb on structural materials

• Establishment of tritium recovery technology

3) Molten salt (FLiBe) blanket

• Development of tritium safe confinement technology
• Development of corrosion resistance technology
• Development of tritium and chemical stability control technology
• Development of molten salt (FLiBe) handling technology and F chemical potential control technology

3.3.2 Current status and issues of materials development

Materials for (1) blanket structures, (2) diverters, and (3) tritium breeding will be treated as "blanket materials" here. Figure 3.3.2-1 indicates expected service conditions for the blanket first wall and the diverter plate. In view of the damage caused by neutron irradiation and the heat flux from the plasma, the conditions for the DEMO reactor components are much more severe than for the ITER experimental reactor. Tritium breeding materials that perform a vital role in the tritium fuel cycling are one of the most unique materials required by the fusion reactor system. Below, the current status, issues, and the prospects for the development of these materials will be introduced with regard to experimental (ITER), demonstration (DEMO), and the later commercial reactors. Also, the schedule of material development and the role of the D-Li fusion neutron source will be described.

(1) Structural Materials

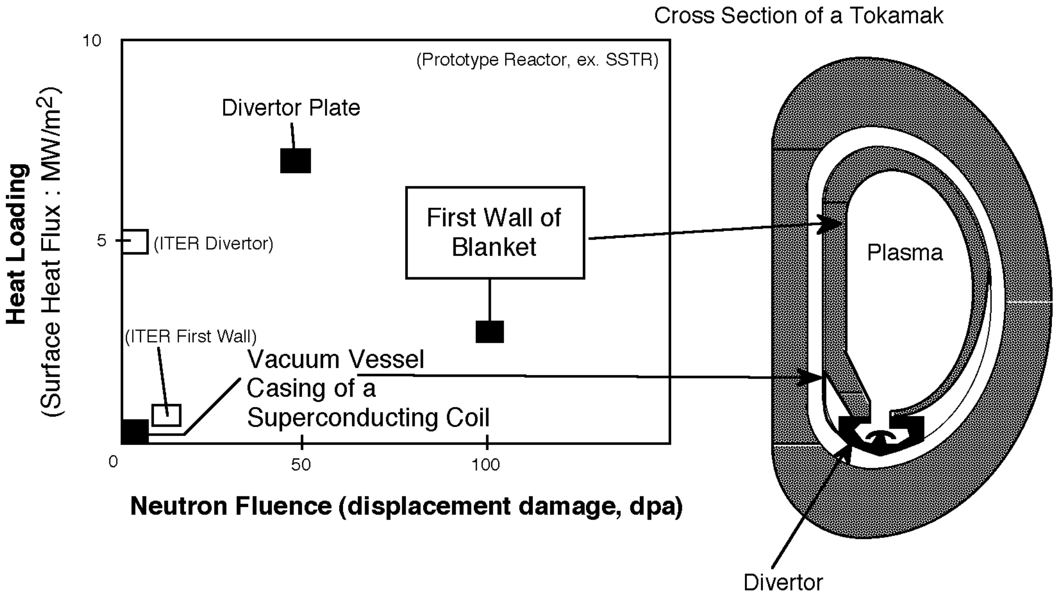


Fig. 3.3.2-1 Major Structures of the Fusion Reactors and their Operating Conditions

Development of structural materials compatible with the high-heat loading at elevated temperatures and to the high neutron damage levels is one of the key factors to construct safe, available, and economically competitive reactors. Moreover, reduction of the residual activity induced by high-energy neutron irradiation will make it possible for the fusion reactor as a next generation large-scale energy source to be harmless to the environment. The requirements for "reduced activation" include the low decay heat during maintenance and the low-induced activity acceptable for the shallow land burial and materials recycling. Alloy development without using the elements (nuclides) of slow induced activity decay is essential for this. For economic competitiveness, improvement of materials is needed to raise the upper bound of the service temperature taking the high-heat flux from the plasma and the reduced activation into account. Since it is necessary to manage the property changes during service for extending the lifetime of the power plant, application of the alloy designing method based on the knowledge of the radiation induced microstructural change is quite important.

The service condition of the structural materials will be quite severe. D-T fusion reaction 14-MeV neutrons introduce displacement damage, transmutation produced gas atoms (hydrogen and helium), and solid transmutation elements in considerable amounts. Materials are expected to retain enough strength to maintain the integrity of the component under the radiation damage by both the high-energy neutrons and the effect of high thermal stress at elevated temperatures. Radiation damage for the materials of the first wall of the blanket is expected to attain levels of about 100 dpa and several thousand appm of helium in the demonstration reactor. Because of this expected severe service condition, a rather long time will be needed for the development and this program should be carefully planned and managed.

Ferritic/martensitic steels, vanadium alloys-refractory alloys, and SiC-fiber/SiC-matrix composite-ceramic composites are potential leading candidate materials and are expected to have the potential of "reduced activation," high thermal performance, and a high neutron wall load. These materials are expected to provide suitable superb performance with water/water vapor, liquid Li, and helium gas coolants. FLiBe is also recognized as a liquid coolant with breeding capability.

Reduced activation is one of the important factors for the performance of maintenance and to minimize radioactive waste. Figure 3.3.2-2 shows the time dependence of the contact dose after shut down for the above materials. Shallow land burial is supposed to be utilized after 100 years of cooling, and reduction of the amount of waste exceeding the allowable level for shallow land burial is essential for the reduction of the load to the environment. Replacement of alloying elements by "reduced activation elements" is one method and is required to reduce the induced activity below the acceptable level for shallow land burial. Elements of "reduced activation" are chosen for the alloying elements of ferritic steels and vanadium alloys. Reduction of the amount of the slow decay elements (nuclides) is also needed for SiC/SiC composite.

For economic competitiveness, the thermal efficiency of the system is a key factor. Figure 3.2.2-3 shows the coolant temperature and the candidate structural materials of the systems together with the thermal efficiency goals; a saturated water vapor system of about 30% thermal efficiency with reduced activation ferritic steel, a supercritical water system of about 40% thermal efficiency with reduced activation ODS steels and vanadium alloys, and a helium gas turbine system of about 50% thermal efficiency with SiC/SiC composites. Both high temperature strength and the compatibility with the coolant are the crucial factors for the development of structural materials.

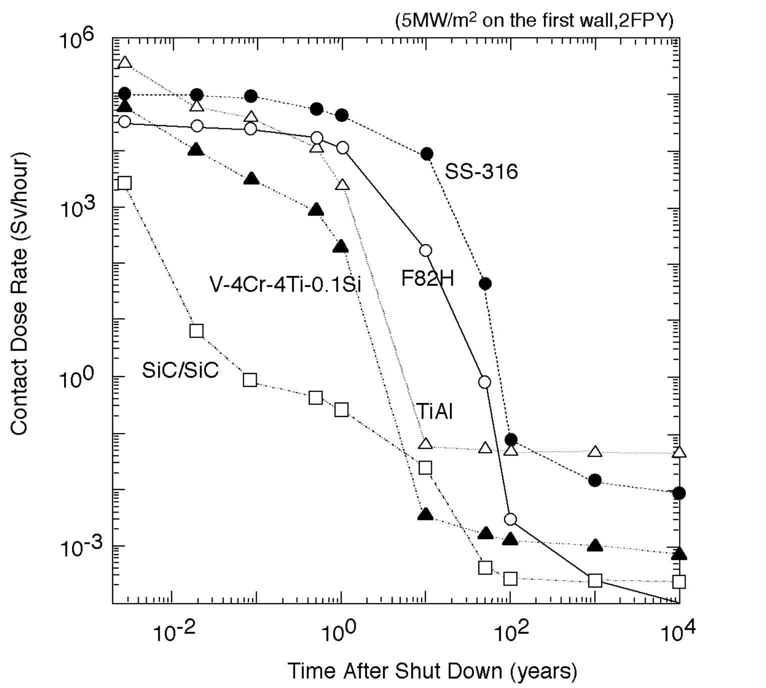


Fig3.3.2-2 Time Evolution Curve of Contact Dose Rate in the First Wall Assuming the Periodic Displacement at a Fluence of $10 MWa/m^2$

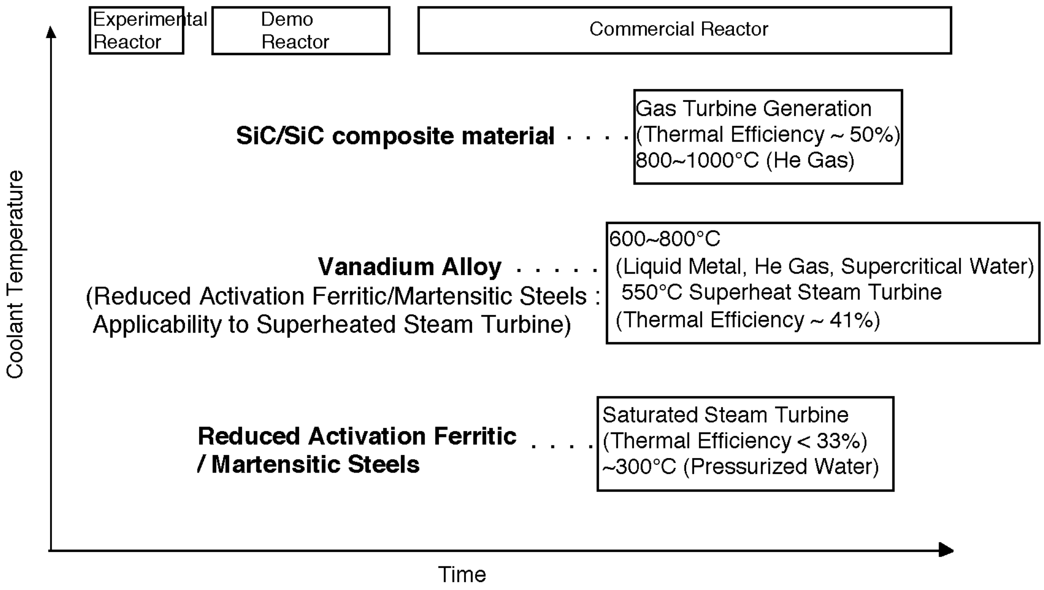


Fig. 3.3.2-3 Relations of Energy Systems and their Materials. Every system is expected to be economically competitive

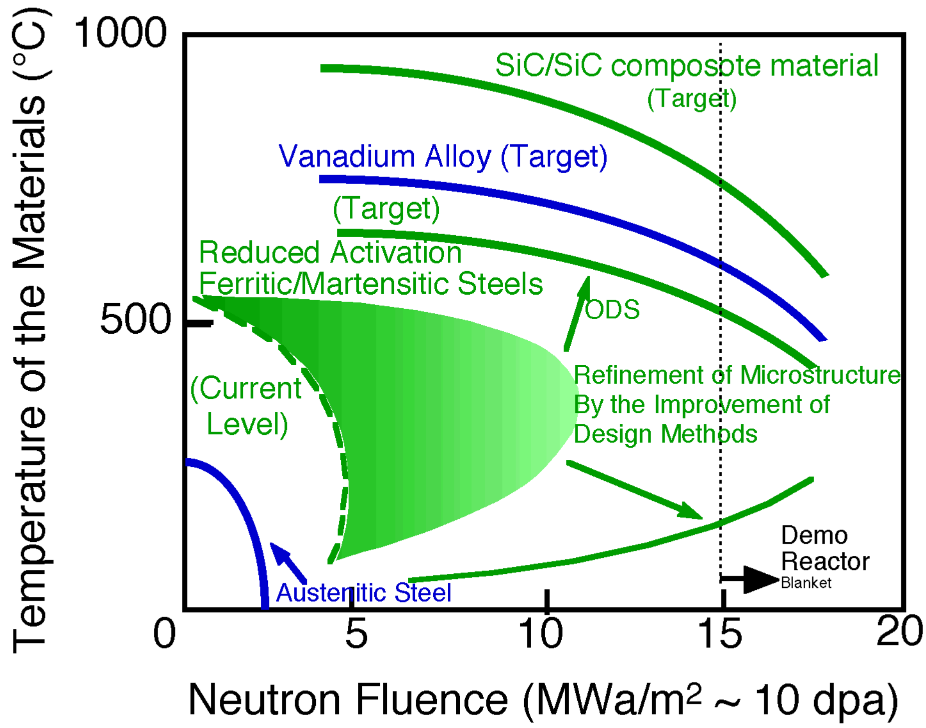


Fig. 3.3.2-4 Development of Structural Materials and their Target Performances in Feasible Temperatures and Neutron Fluences

Because the reduced-activation ferritic steel has a good base in materials engineering and is in an advanced stage of development compared with vanadium alloys and SiC/SiC composite, it is recognized as being the prime candidate structural material. Figure 3.3.2-4 indicates the performance goals for the reduced activation ferritic/martensitic steels together with the vanadium alloys and the SiC/SiC composites. The lower bound temperature is limited by the embrittlement during irradiation. The upper bound is limited by the transmutation-produced helium-induced embattlement (He embrittlement) and irradiation creep. During the last decade, reduced activation ferritic/martensitic steels with equal or superior resistance to irradiation compared with those without compositional modifications for reduced activation have been obtained. Ferritic/martensitic steels have been used successfully as duct materials of the fuel assemblies for fast breeder reactors to a displacement damage level of about 150 dpa (corresponding to a level produced by a wall loading to 15 MWa/m2). Evaluation with simulated irradiation condition experiments using HFIR in Oak Ridge (Japan-U.S. HFIR collaboration experiment) and with the basic analyses on the microstructure change and the fracture conditions of the component under the expected service condition revealed that the current reduced activation alloys have an irradiation resistance to a damage level corresponding to that produced by a neutron wall load to 4 MWa/m2 (equivalent to that for one full year of operation), as indicated by the dashed line in the figure. Exploring and demonstrating the irradiation resistance of the alloys to higher damage levels with simulated irradiation experiments (see the shaded area in the figure) are being planned. The results to date are suggesting the feasibility of the blanket system with reduced activation ferritic/martensitic steels, although the available service temperature range is limited. It is of course important for fusion reactor to have economic competitiveness with other energy sources. The Japan-U.S. HFIR collaboration experiments are being conducted to evaluate the response of materials to high damage levels and for the further improvement of the reduced activation alloys for this purpose. In addition, for accurate and quantitative evaluations for the development of the design database for the DEMO reactor and to further expand the service conditions, irradiation experiments with an accelerator-driven fusion neutron source producing a fusion relevant neutron environment are recognized to be necessary.

Critical issues for the development of the reduced-activation ferritic/martensitic steels are to manage the radiation induced embrittlement at low temperatures and the improvement of the high-temperature strength. Improvement of the corrosion resistance and processes to enable the large-scale production of the reduced-activation steels are also the important subjects.

For the improvement of high-temperature strength, strengthening of the reduced activation ferritic/martensitic steels by the dispersion within the alloy of nanometer-size oxide particles is expected to be one of the solutions. Composite materials including graded materials technology is thought to be one of the options to manage both corrosion resistance and fracture toughness after irradiation. After accomplishing these improvements, database and the knowledge base development using the accelerator driven neutron source need to be conducted.

For radiation induced embrittlement, modification by the minor alloying elements and the improvement of the design methodology are thought to be effective ways to determine the key to managing this problem. Radiation induced embrittlement is recognized to be rather strongly affected by the transmutation produced He and H elements. The addition of the minor alloying elements and the optimization of the mechanical heat treatment to make fine dispersion of radiation produced He and H cavities seems to be effective in retarding embrittlement. As for the improvement of the design methodology, the application of the recent progress of the fracture mechanics utilizing the margins of the small-size components to brittle fracture and utilizing the experience of the experimental reactor ITER to manage the radiation damage of materials are thought to be effective to expand the design window.

Evaluation of the effect of ferromagnetism on plasma control also needs to be accomplished. Theoretical calculations and small-scale experiments are suggesting the feasibility of such plasma control. Further verification of the compatibility with the advanced methods of plasma control need to be conducted. The chemical composition of the reduced activation ferritic/martensitic steels is planned to be determined by 2010.

The plant life of a fusion reactor with a thermal efficiency of 30% or higher is required to be equal to, or longer, than 30 years to be economically competitive. The cumulative neutron wall load for 30 years will be 90 MWa/m2 with 3 MW/m2 of the neutron wall load. This produces 900 dpa of displacement damage, 10,000 appm of He and 40,000 appm of H. This is beyond the present ability to estimate the lifetime of ferritic/martensitic steels based on fast breeder reactor experience and the present knowledge about irradiation induced property changes. From the experience of applications with fast reactors having damage levels of 150 dpa, it is expected that ferritic/martensitic steels can be used in combination with the planned exchange made at the damage levels ranging from 100 to 200 dpa. The application of vanadium alloys in fast reactors has been infrequent. However, the lifetimes of the vanadium alloys are also expected to be equivalent or even higher than ferritic/martensitic steel lifetimes based on the results of experimental studies. Cost is also a factor for the application of vanadium alloys, in addition to the performance under irradiation. Other industrial demands and the impact of recycling after service may affect it.

Service conditions found in several design studies for the first wall materials of the blanket structures of DEMO-class reactors are summarized in Table 3.3.2-1. The reference designs are SSRT, Proto-DREAM, and AREIS-RS for reduced activation ferritic/martensitic steel, SiC/SiC composite material, and vanadium alloys, respectively. The temperature of the blanket depends on the coolant used. The materials are required to have a high enough strength at elevated temperatures for the high thermal stress caused by the heat flux from the plasma under the effect of intense neutrons. Some design studies have been made projecting the DEMO reactor will be built in the 2030s.

The National Institute for Fusion Science is also conducting a design study for the FFHR using FLiBe as a coolant.

Table 3.3.2-1 Design parameters for the blankets of Tokamak DEMO-class reactors

Design Name	SSTR	ARIES-RS	Proto-DREAM
Structural Material	Reduced Activation Ferritic Steel	Vanadium Alloy	SiC/SiC Composit
Coolant	Pressurized Water	Liquid Lithium	Helium Gas
In/Out Temperature	285/325 ^oC	330/610 ^oC	500/800 ^oC
Max Heat Flux	1 MW/m^2	0.48 MW/m^2	0.3 MW/m^2
Max Neutron Wall Load	5 MW/m^2	5.6 MW/m^2	1.6 MW/m^2
Neutron Fluence	10 MWa/m^2	14 MWa/m^2	8 MWa/m^2
References	[3.3.2-1]	[3.3.2-2]	[3.3.2-3]

1) Technological Goal

The goals of materials development for the experimental, the demonstration, and the following commercial reactors are summarized in Table 3.3.2-2 together with the current status and the issues of the development. The requirements for the blanket materials are; (i) the nuclear and physical properties of the materials are appropriate for the function of the blanket (tritium breeding and extraction of heat for energy production), (ii) the strength and swelling properties are suitable as blanket structural materials, and (iii) the induced activity is low enough for maintenance and after service for shallow land burial.

ITER is going to have two types of blankets, namely the shielding blanket and the test blanket. The test blanket is a test module to evaluate the performance and examine the design technologies for the power generation blanket of the DEMO reactor. Although the expected end of life neutron fluence, 0.3 MWa/m2, is not high, the functions of tritium breeding and heat generation at an elevated temperature are required for the test blanket. On the other hand, breeding and heat generation are excluded as required functions for the ITER shielding blanket.

The requirements for the blanket of the DEMO reactor are similar to those for the commercial reactor, except economic production of electricity. Therefore, the materials for the first wall of the DEMO reactor are required to withstand the effects of radiation damage by intense neutron flux, high stress and strain induced by the high heat flux, corrosion by the coolant (water, helium gas, liquid metal, molten salt) at elevated temperatures, and corrosion caused by the breeding materials. This means that materials must withstand neutron irradiation damage, have adequate high strength, have acceptable compatibility with coolants/breeding materials at elevated temperatures, and have sufficiently low induced activities. Further, there must be no critical issues for materials processing and fabrication (forming, cutting, welding) of the blanket. Reduced activation ferritic/martensitic steels, vanadium alloys, and SiC/SiC composites are expected to have the potential to satisfy these needs. For the development of the materials available for DEMO-class reactors listed in Table 3.3.2-1, an irradiation experiments in a neutron environment (high neutron fluence level) equivalent to that of a D-T fusion reaction are necessary. Neutron fluence to the ITER test blanket will be limited to a level only available for the evaluation of the blanket functions. Therefore, the irradiation experiments with an accelerator driven fusion neutron source are essential for the materials development for the DEMO reactor. A neutron source having an appropriate capacity for the irradiation experiments must be built. Experiments performed in it will clarify the potential candidate materials and will obtain data for an engineering database of the irradiation effects on materials available for the DEMO structural design. The utilization of fission reactors and the experimental results from fission reactor irradiation experiments are also important to obtain the basics of such a database. For the development of the reduced activation ferritic/martensitic steels, fission reactor irradiation experiments will be used mainly for the improvement of the alloys and the development of welding alloys during this decade. This will be followed by irradiation experiments, using a simulated fusion neutron source driven by an accelerator, to clarify the limitations of the alloys.

2) Current status

Stainless steel 316IG, which is going to be used for ITER, is the type of 316 stainless steel specified for ITER applications. This alloy is similar to that used for fast breeder reactors. The irradiation response and other characteristics of this alloy are well characterized. In addition, the service conditions seem rather mild and no critical issues are expected except the effect on properties introduced by the component fabrication techniques, including thermal cycling accompanied by joining.

The effects of fabrication processes to join copper alloys needs to be evaluated. From the viewpoint of structural design, the development of copper alloys seems to be needed. Because the nuclides with slow activation decays cannot be used for the DEMO reactor to satisfy the reduced activation requirement, the freedom of the chemical composition for the materials tends to be limited. Improvement of methods to design a structure to accommodate the material ductility loss caused by irradiation is an important subject.

Reduced activation ferritic/martensitic steel and vanadium alloy, both of which are expected to be used for the blanket materials in the DEMO reactor, are going to be also used for the ITER test blanket. The use of these materials in ITER will supply the important knowledge about the behavior of these alloys during service conditions. However, this may supply only limited information about the service life of the alloy because of the relatively low neutron fluence in ITER.

3) Issues and prospective solutions

Issues for the reduced activation ferritic/martensitic steels are the radiation damage (embrittlement at low temperature, helium embrittlement, and swelling), the strength at elevated temperatures, the ability to process reduced activation alloys, the corrosion induced by the coolant, the permeation of tritium, and the compatibility with breeding materials. The requirements on the fracture toughness are to retain enough fracture toughness to avoid brittle fracture during maintenance (the tentative target: the ductile-to-brittle transition temperature < 80ºC) and to avoid fast fracture during service (the tentative value of fracture toughness: > 60 kJ/m2). It has not been reported that the post irradiation fracture toughness (measured at room temperature) of reduced activation ferritic/martensitic steels became smaller than 60 kJ/m2 after irradiation at the expected service temperatures. Also, without the effect of transmutation produced helium atoms, the ductile-to-brittle transition temperature following irradiation tends to saturate to a level below 100ºC. These results suggest that radiation induced embrittlement may not become a critical issue, except for the effect of the helium atoms. Several methods are being examined to reduce the helium effect. Refinement of the helium bubble microstructure by precipitation of fine carbides to reduce the migration of the helium atoms is one method to manage the effect.

The problems caused by the effect of swelling and compatibility may not become severe as compared with the problem of radiation induced embrittlement. The feasibility of reducing impurities to achieve the reduced activation requirement has been successfully evaluated by fabricating a test ingot with a commercial arc melting facility.

Issues for the vanadium alloys are radiation damage (embrittlement at low temperature, helium embrittlement, and swelling), radiation damage of self-healing insulator coating (to reduce the MHD effect), processing of reduced activation alloys, and tritium permeation. The issue of fracture toughness including the irradiation effects is thought to be less severe than that of the insulator coating. AlN and CaO are the leading candidates for the insulator coating. Also, manufacturing technologies for large ingots are in development.

Superior strength at high temperatures is one attractive feature of SiC/SiC composites. Another is the relatively high thermal conductivity, especially when compared with other ceramic composites. Both points are outstanding when considering the suitability for the blanket material. The first step in the development seems to be the improvement of non-irradiated properties to encourage the industrial application of this material. Next, the robustness against neutron irradiation needs to be improved for the application to the blanket structure. Specific issues of concern are the degradation of strength, the reduction of thermal conductivity, the embrittlement, and the tendency to swell which all result from irradiation. Moreover, technologies to fabricate the components to reduce the amount of impurities with slow radioactivity decay and those needed for high hermetic performance need to be developed. The methodology for the design of structures with relatively low ductile ceramic composites is also an important subject to promote to use these materials. Improvement of the thermal conductivity is a crucial subject when these composites are being considered as a fusion reactor material. Reaction bonding is one of the promising fabrication methods, and its development is progressing. It has been found that swelling of this material, which is a function of the degree of crystallization and irradiation, degrades its strength. The application of highly stoichometric fibers that have an increased degree of crystallization is expected to suppress the degradation of strength during irradiation. For reduced activation, nitrogen (N) free coating materials for the fibers need to be developed. A method to process and fabricate components is being developed in a non-nuclear application field.

An accelerator driven neutron source for the irradiation experiments of fusion reactor materials has not been developed yet. Therefore, the ability of irradiation facilities is limited. This limited ability is mainly the result of the difference between the neutron spectrums of fusion and fission reactors. The dissimilarity causes (i) different helium and hydrogen production rates and (ii) a different distribution of the point defect clusters, and results in a change in the irradiation response of materials. This is a major problem for materials development. The impact of the helium production rate on the microstructural evolution of materials is readily obtained by simulated irradiation using ion accelerators. To improve the accuracy in evaluating irradiation response of materials, application of the accelerator driven fusion neutron source is essential for an accurate evaluation of the irradiation effect on materials. On the other hand, fission reactor irradiation will be used in optimizing the combination of materials (integration of materials), before the irradiation experiments using the accelerator driven fusion neutron source.

(2) Divertor related materials

Divertor related material includes the structural material itself, material for a cooling tube, and an armor material. In divertor development, divertor components will be developed using blanket structural materials or existing materials, which will have resulted from blanket material R&D. (See Table 3.3.2-3)

1) Technological objectives

Structural materials for the ITER divertor are required to withstand a surface heat flux of 5 MW/m^2 at steady state and 20 MW/m^2 at a transient condition. Furthermore, these materials should endure a neutron load of more than 50 dpa for the DEMO reactor. For the commercial reactor, these materials will be required to have higher heat flux resistance, higher resistance against irradiation, and higher durability. The requirements for the cooling tube and the surface material are almost the same as that of the structural material.

2) Present status

Stainless steel, which is used in the blanket, is adopted as the structural material for the ITER divertor. The divertor structural material will continue to be the same as used for the blanket in and after the DEMO reactor. Because neutron flux at the divertor is several times less than that for the first wall, the divertor can be designed with a margin considering irradiation damage and activation.

For the cooling tube material of the ITER divertor, copper alloy (CuCrZr, Ds-Cu) has been adopted, which is used with low temperature water coolant at low pressure (~100ºC and ~4MPa) in order to remove a surface heat flux of 20 MW/m^2 at maximum. These materials have sufficient strength to withstand the thermal stress under the high heat flux conditions.

For the armor material, carbon fiber composites have been used for the areas where a heat flux of 20 MW/m^2 is expected. In other parts, the armor material is of tungsten, which has a low erosion yield against energetic plasma particles.

3) Issues and outlook for the future

The surface heat flux of the divertor in and after the DEMO reactor will be similar to that for the ITER. However, in the DEMO reactor, high temperature and high-pressure coolant should be used to produce electricity. Therefore, cooling tube material is required that has improved strength at high temperatures and adequate resistance against a high heat flux. At the same time, the method of design including the effect of local plasticity should be established. In a reference program, divertor components will be developed using the same structural material as used for the blanket. As a backup, development of Cu alloy and tungsten alloy, which will be used for ITER, should also be performed.

For the armor material in the DEMO reactor and the reactors that follow, if a suitable disruption control technique is developed, the structural material itself can be used as the armor material. If disruptions cannot be avoided, armor is necessary. In this case, it is necessary to accumulate irradiation data, such as the sputtering of irradiated W. In parallel with the data accumulation, tungsten materials, which have high resistance against neutron irradiation, should be developed as well as an in situ repair technique, such as a plasma spray method.

(3) Materials for tritium breeding (tritium breeders and neutron multipliers)

The composition of materials for tritium breeding depends on the blanket system. In the solid blanket system, ceramics are used as the tritium breeder and a neutron multiplier is used. In the liquid blanket system, liquid lithium and lithium-lead alloys are used as the tritium breeder and, if needed, molten salt (FLiBe, etc.), will be used as a neutron multiplier. Below, the status and problems of a solid blanket system are mainly described.

The components for tritium breeding are the tritium breeder and neutron multiplier. The service environments of components and materials used for these are shown in Table 3.3.2-4. Tritium breeding ratio (TBR) is an important parameter in selecting the materials for a tritium breeder and it is necessary to be greater than 1. The TBR is decided by arrangement and kind of tritium breeder and neutron multiplier, the enrichment of ^6Li in the tritium breeder, and so on. The temperature of the tritium breeder and neutron multiplier can be controlled by appropriate arrangement of the tritium breeder and the neutron multiplier. Therefore, the main problems concerning materials science are stability under neutron irradiation, chemical reactivity, and safety during a transient state. The pebble shape for the active materials is selected for relaxation of thermal stress and minimization of swelling. The pebbles are packed by binary pebble packing. Development of mass-production techniques and low-cost production of pebbles are the main target.

Table 3.3.2-2 Development of structural materials (Issue, Target and Prospect)

Items	Experimental reactor	Prototype reactor	Commercial reactor
Blanket structural materials	- Shielding blanket: 316 stainless steel (ITER grade) - Test blanket: Reduced activation ferritic/martensitic steel and Vanadium alloys	Reduced activation ferritic/martensitic steel (RAF/M), Vanadium alloys, SiC/SiC composite	Reduced activation ferritic/martensitic steel, Vanadium alloys, SiC/SiC composite
Technological goal	- Shielding blanket: Shielding of neutrons to a life corresponds to 0.3MWa/m2 - Test blanket: Trial of tritium production. Demonstration of life corresponds to 0.3MWa/m2 at high temperatures	Structural materials (and the structure) are required to be compatible with the tritium production and with the heat removal to extract thermal energy during a lifetime corresponds to 10MWa/m2 of neutron wall loading. Maximum temperature for RAF/M is 500C (for water cooling system), that for the vanadium alloy is 700C (for liquid metal cooling) and that for SiC/SiC composite is expected to be 1000C for He gas cooling.	Material lifetime is required to extend to about 20MWa/m^2.
Current status	- Most of the basic properties including the irradiation effect have been evaluated. Results are indicating the alloys are acceptable for application. Some of the subjects, such as the effect of thermal cycling during fabrication on IASCC will be evaluated. - Test blanket: Major composition of the reduced activation ferritic/martensitic steels and vanadium alloys have been established. Because the damage level is expected to be relatively low, materials available are expected to be acceptable.	Major chemical compositions of the three leading materials have been established. Irradiation properties of these materials are in progress. Technologies of processing and engineering basis for application except for the irradiation effect have been almost established from the industrial experiences of non-reduced activation ferritic/martensitic steels. For vanadium alloys and SiC/SiC composites, processing of the materials and the fabrication technologies of components are the important issues.	Requirements about life time and service temperature are rather sever comparing with those for the materials of prototype reactor. The materials are expected to be obtained by the improvement of the materials for prototype reactor.
Issues	- Shielding blanket: Effect of fabrication process on the properties need to be examined.- Test blanket: Evaluation of the irradiation performance need to be accomplished using accelerator driven neutron source. Development of fabrication methods taking the effect on the irradiation performance into account including the compatibilities with coolant and the breeding materials is need to be accomplished.	One of the most important issues of this stage is the evaluation and the improvement of irradiation performance. To develop of the materials database, the fabrication process from the materials to the component and accelerator driven fusion relevant neutron source for the evaluation of materials performance are also important issues. Reduction of the radiation induced degradation of fracture toughness and the improvement of corrosion resistance are the subjects with priority. Important subjects for vanadium alloys is the development of self-healing insulator coating to reduce MHD effect. Those for SiC/SiC composites are the improvement of the thermal conductivity and to develop the technology for the fabrication of the component.	One of the crucial subjects is to extend the lifetime (~ 20MWa/m2). Improvement of the high temperature strength is important for the reduced activation ferritic/martensitic steels.

Methodology for the development	- Shielding blanket: Establishing of available conditions by examining of the effect of fabrication process on the properties. - Test blanket: Optimization of joining (HIP) conditions in view of the effect of the process on the properties. Evaluation of the compatibility with the breeding materials is needed. Depending on the needs, improvement of design methodology taking the irradiation effect into account may be accomplished. Development of the accelerator driven neutron source is also important.	Improvement of properties by optimization of the additional elements, refinement of structure and the improvement of the processing are expected. Available service condition may be extended by the improvement of the design methodology.Refinement of microstructure before irradiation and that of the damage microstructure produced by irradiation by optimizing alloying element and the heat mechanical treatment are expected to be effective to suppress the degradation of fracture toughness and the He embrittlement at elevated temperatures for the reduced activation ferritic/martensitic steels. CaO and AlN are expected to be promising self-healing insulator coatings for the vanadium alloys. Reaction bonding method for SiC/SiC composite is expected to be effective to improve thermal resistance. Promotion of international collaboration is essential for the acceleration of the development including the task sharing.	For the application of the reduced activation ferritic/martensitic steels, it is essential to improve the high temperature strength by ODS and other technologies.
Prospects	- Shielding blanket: No major issue is expected except for the effect of the disruption, because of the rather mild service condition. - Test blanket: Also, no major issue is expected except for the effect of the disruption, because of the rather mild service condition. It is expected to establish the design methodology taking the irradiation effect into account. Conceptual design activities and other activities are being carried out.	Alloy development and the development of the design technology are expected to be effective to satisfy the requirements. This seems to me more feasible comparing with the application of SiC/SiC composite materials and other materials. For the application of the vanadium alloys, development of self-healing insulator coating is an essential way. The large scale ingot making technology for the vanadium does not seem to be difficult comparing with other issues. Improvement of irradiation resistance is expected for the SiC/SiC composite. However, it needs some time to examine the feasibility.	Extending of the lifetime will be required. From the experience of the development of the fuel cladding of FBR, some improvements are expected to be done. To satisfy the requirements, further improvement may be expected.

Table 3.3.2-3 Goal of material development and perspective (divertor related material)

Item	Experimental Reactor	Demonstration reactor	Commercial reactor
Divertor Structural Material	Structural material: SUS316 (ITER grade) Cooling tube : copper alloy (CuCrZr, Ds-Cu)	Structural material: reduced activation ferritic steel, vanadium alloy, and SiC/SiC composite etc. Cooling tube: Same as structural material or Cu alloy as in experimental reactor.	same as the demonstration reactor.
Technical objective	Structural material: keep neutron shielding performance up to 0.1MWa/m^2. Cooing tube: Heat flux of 5MW/m^2 (steady state), 20MW/m^2 (transient state) must be removed.	Highest temperature in use is the same as the blanket materials. It is important for the divertor that heat flux of 5MW/m^2 can be removed.	same as the demonstration reactor. Durability ~20MWa/m^2.
Present status	Structural Material: Evaluation of basic properties including the irradiation effect will be finished. Basically, same as blanket structural materials. Cooling tube: Evaluation of basic properties including the irradiation effect will be finished. Because neutron damage is estimated to be small, there is no problem to use the materials which is available now.	Structural material: Same as the blanket. Cooling tube: Dispersion strengthened and precipitation strengthened copper are candidates, but their chemical composition and manufacturing process is not optimized.	Same as demonstration reactor.
Issues	Structural material: Evaluation of the fabrication process is inevitable. Cooling tube: Development of manufacturing process and its effect on irradiation is required.	Structural material: Same as the blanket. Cooling tube: In the case of dispersion strengthen copper, optimization of particle shape and manufacturing process should be solved. For the precipitation strengthen copper, chemical composition and the heat treatment should be optimized.	Durability must be improved up to ~20MWa/m^2.
Concrete method and means to overcome problems	Structural material, cooling tube: Availability should be clarified by evaluation of effects during manufacturing process. Addition to this, high power neutron source will be constructed.	Structural material: Same as blanket structural materials. Cooling tube: The dispersion strengthened copper will be improved by micrifing the disperted particle. For the precipitation strengthen copper, decrease in toughness will be suppressed by heat treatment.	Same as demonstration reactor.
Perspectives	There is no problem about the lifetime of the material. Development of the structure design method including the effect of irradiation is now under consideration.	Structural material: Same as blanket structural materials. Cooling tube: The probability to meet the requirement is considered to be high for both dispersion strengthened copper and precipitation strengthened copper. Method for structure design including the irradiation effect and local plasticity is under consideration.	Same as demonstration reactor.
Divertor Surface Material	CFC and tungsten	Same as structural materials or tungsten material.	Same as structural materials
Technical objective	Ability of neutron shield must be maintained up to 0.1 MWa/m^2. Excessive erosion or delamination should not occur. Heat flux of 5MW/m^2 (steady state), 20MW/m^2 (transient state) must be removed.	The ability to remove the heat must be retained up to 10MWa/m^2. Steady state heat flux of 5MW/m^2 must be removed. In case of W, highest temperature in use is about 1500 °C.	Basically the same as demonstration reactor. Durability ~20Mwa/m^2.

Present status	Evaluation of basic properties including the irradiation effect will be finished. Evaluation of adsorption and desorption characteristics of redeposited layer, dust, mixed materials are now on going. There is no problem to use existing materials.	Basic properties have already been evaluated. However, the data of redeposited layer and neutron irradiation effect is still lacking.	Same as demonstration reactor.
Issues	Adsorption and desorption characteristics of redeposited layer, dust, mixed materials will be evaluated.	Adsorption and desorption characteristics of redeposited layer and neutron irradiated materials will be evaluated and chemical composition, surface treatment and manufacturing process will be optimized.	Same as demonstration reactor.
Perspectives	There is no big problem in the performance of the material.	Taking the progress on plasma control technique into consideration, it will be able to use structural material as an armor material. In case of the ferritic steel, adsorption of water is the problem, but it will be solved by the surface treatment. As for the W materials, it is important to develop the manufacturing technology to repair the material in a short time. This can be archived by extending the existing technology.	Same as demonstration reactor.

Table 3.3.2-4 Environment of the materials for tritium breeding in experimental reactor (ITER), demonstration reactor (DEMO) and commercial reactor

(Tritium breeder)

Items	Experimental reactor (ITER)	Demonstration reactor (DEMO)	Commercial reactor
Li burn-up	~5%	5~20% (replace per 2 years)	5~10% (replace per 2 years)
Nuclear heating (MW/m^3)	~50	~150	~150
Temperature (˚C)	200~400*	400~1000*	600~1000*
Environment	He (~0.1MPa)	He (~0.1MPa)	He (0.1~10MPa)

* : These values are example for Li_2O and are difference with tritium breeder materials.

(Neutron multiplier)

Items	Experimental reactor (ITER)	Demonstration reactor (DEMO)	Commercial reactor
He generation ratio (appmHe)	~3000	~20000 (replace per 2 years)	~20000 (replace per 2 years)
Nuclear heating (MW/m^3)	~10	~30	~30
Temperature (˚C)	150~350	400~900	600~900
Environment	He (~0.1MPa)	He (~0.1MPa)	He (0.1~10MPa)

Concerning the use of the tritium breeder and neutron multiplier, similar materials now existing can be used because the service conditions of these materials are not severe in the experimental reactor (ITER) conditions. However, stability under high temperature and high neutron fluence is necessary in the DEMO reactor. Therefore, development of modified and advanced materials is necessary. The neutron irradiation experiments under the international cooperation and in IFMIF will be able to resolve the issue of stability under high neutron fluence. The technical target, status, problems, and prospects of tritium breeder and neutron multiplier are described in the following (see Table 3.3.2-5).

1) Technical aim

The main technical items for the development of materials for tritium breeding are the establishment of mass production technology, the improvement of the properties for tritium breeding and release, and the extension of lifetime.

For the mass production technology of the solid tritium breeders, a large quantity of pebbles (diameter of small-size pebble: 0.1-0.2 mm, diameter of large-size pebble: 1-2 mm) will be necessary to be produced for the fusion reactor. About 50 tons and 100 tons of tritium breeder pebbles will be used for ITER and the DEMO reactor, respectively. For the properties of the tritium breeding and release, the selection of the suitable concentration of Lithium-6 (^{6}Li) and a decrease of tritium inventory is necessary. A lifetime up to 20% Li burn up and 80 dpa at a temperature of 400-1,000ºC is also an important target for the DEMO reactor.

Liquid lithium, lithium-lead alloy, and molten salt (FLiBe) are candidates for the liquid tritium breeders. Tritium recovery and inventory control, refining and purity control, and decreasing the MHD pressure drop are all key issues for the use of liquid lithium. For the lithium-lead alloy, the key issues are the containment of tritium and corrosion control. For the FLiBe, the key issue is the containment of tritium, the corrosion protect of the pipe, the change of chemical properties by radiation effects, etc. The development of coating technology characterized by the electrical insulation, the tritium permeation barrier, and the protection against corrosion are all important problems to solve.

As the neutron multiplier, beryllium is the first candidate material, and beryllium intermetallic compounds for the high-temperature application are also proposed as candidate materials. The main technical items for the development of the neutron multiplier are the establishment of mass production technology of a large quantity of pebbles (diameter of small-size pebble: 0.1-0.2 mm, diameter of large-size pebble: 1-2 mm) and the improvement of the properties for cracking and swelling to extend of the lifetime under the neutron irradiation. For mass production technology, a large quantity of pebbles about 150 tons and 300 tons will be used for ITER and the DEMO reactor, respectively. For the irradiation conditions, 150~350ºC, ~3,000 appm He, ~3 dpa for the ITER reactor, and 400~900ºC, ~20,000 appm He, ~20 dpa for the DEMO reactor.

2) Status of development

For the solid tritium breeding materials, many studies were performed, and Li_2O, Li_2TiO_3, $LiZrO_3$, Li_4SiO_4, and $LiAlO_2$ have been selected as candidate materials. For the mass production technology of pebbles, the trial fabrication by the melting granulation method, the rotating granulation method, and the wet process were carried out for the above materials. The wet process was selected because of its mass production and cost advantages. In the selected wet process, the dehydration type gel method is proposed for large-size pebble production, and the permutation type gel method for small-size pebble production. For the dehydration type gel method, a mass production of about 150 kg/y is possible. For the characterization of the above materials, except for Li_2TiO_3, almost all data of the non-irradiated materials were obtained. However, the material data of neutron irradiation behavior, i.e., thermal properties (thermal conductivity, thermal expansion coefficient, etc.), mechanical properties (compression strength, creep, fracture strength, etc.), tritium release properties and swelling, were still not sufficient to design even the ITER test blanket pebbles. The limited basic material data up to 5% Li burn up for Li_2O discs and $LiZrO_3$ pebbles were obtained by the US-Japan international collaboration (BEATRIX-II). However, the engineering data for lifetime has not been obtained, and the detailed design of the tritium-breeding blanket is impossible without this data. The development of mass production technology and basic characterization for the improved material, which has other chemical compounds added as candidate materials to increase the tritium content to above 90% and to reduce the grain size to below 10 microns, has been started to increase the tritium breeding ratio. The high Li burn up test by a fission reactor for the first-stage selection of tritium breeding materials is being discussed under the international collaboration at the ISTC (International Science and Technology Committee), since it needs a large amount of resources and a long time to prepare such a specific test facility in Japan.

For the liquid lithium, the compatibility tests between liquid lithium and each structural material in the thermal convection loop are in process. Also, the manufacturing and the evaluation for the several insulation films to reduce the MHD pressure drop are in process. As for lithium-lead alloy, the corrosion test, the development of coating materials and its performance test are being carried out. The compatibility test of each material with molten salt (FLiBe, etc.) is in process, and the fabrication of the thermal convection loop is underway.

As for neutron multiplier, beryllium has been chosen as the primary candidate material, and study is focusing on manufacturing engineering and the neutron irradiation behavior.

On the development of mass production technology, the rotating electrode process has enabled the production of beryllium pebbles of up to 120 kg per year. The study to clarify several items of behavior is being carried out from the following viewpoints, but the data for the detailed design of the blanket are not yet complete.

· Thermal design of pebble bed

  Thermal conductivity, thermal diffusion coefficient, thermal expansion coefficient, etc.

· Lifetime estimate

  Compression strength, pebble bed strength, swelling, etc.

· Tritium inventory estimate

For the neutron irradiation behavior evaluation for the experimental reactor (ITER) including the solubility of tritium and the thermal diffusion coefficient, only the tritium diffusion coefficient and the swelling have been measured under the condition of 400ºC, ~3,000 appm He, ~30 dpa in an irradiation test using EBR-II. As for the evaluation of the neutron irradiation behavior for the DEMO reactor, no data have been obtained. In addition, the existing beryllium data was unsuitable because the DEMO reactor will have severe conditions, with temperatures over 600ºC and neutron fluence ten times higher than that for ITER. Therefore, the advanced materials for the refractory multiplier have to be developed. For the refractory material, beryllium intermetallics ($Be_{12}Ti$, etc.) have been studied recently for aerospace applications, and it is becoming clear that this material has little reactivity with water. However, the study of the basic physical properties is still in its early stages. It is necessary to estimate the lifetime, but there is no irradiation facility in Japan that provides the irradiation conditions of the DEMO reactor (400-900ºC, ~20,000 appm He, ~20 dpa).

Table3.3.2-5 Target subjects and prospects for material development

(Tritium breeder)

Items	Experimental reactor (ITER)	Demonstration reactor	Commercial reactor
Tritium breeder	Material : Li_2O, Li_2ZrO_3, Li_2TiO_3, Li_4SiO_4 Shape : pebble	Material : Li_2O, Li_2TiO_3 or improved material Shape : pebble For the other material, i.e. liquid lithium, molten salt , etc	The same as demonstration reactor.
Technical target	Solid Blanket System : · Mass production development of tritium breeder pebbles (large size pebble : φ1 2mm, small size pebble : φ0.1 0.2mm) (Amount of tritium breeders : about 50 tons) · Good tritium breeding and release behaviors of tritium breeders (Enrichment of ^6Li : 90% (in the case of Li_2ZrO_3 and Li_2TiO_3)) (Tritium inventory : <100g) · Stability of tritium breeder at 5% Li burn-up and 400 ~1000˚C (No crack, swelling, etc.).	Solid Blanket System : · Development of the mass-product method for pebble of tritium breeders (large size pebble : φ1 2mm, small size pebble : φ0.1 0.2mm) (Amount of tritium breeders : about 100 tons) · Good tritium breeding and release behaviors of tritium breeders (Enrichment of ^6Li : 90% (in the case of Li_2ZrO_3 and Li_2TiO_3)) (Tritium Inventory : <100g) · Stability of tritium breeder up to 20% Li burn-up and 400 1000˚C (No crack, swelling, etc.) Liquid blanket : · Liquid lithium Reduction of tritium inventory, purity controls in liquid lithium and decrease of MHD pressure loss. · Lithium-lead alloy Development of coating materials for insulation, prevention at tritium penetration and anticorrosion. · Molten salt (FLiBe, etc.) Development of coating materials for insulation, prevention at tritium penetration and anticorrosion.	The same as demonstration reactor.
Status	Solid Blanket System : · Pebble fabrication by wet process with substitution reaction (150kg/year). · Evaluation of tritium release behavior from tritium breeding materials in low-neutron irradiation. · Stability of Li_2O and Li_2ZrO_3 irradiated at the condition of 5% Li burn-up by BEATRIX-II.	Solid Blanket System : · Development of pebble fabrication by wet process. · Development of improved materials with good tritium release behavior (start of the grain size control tests by the addition of other material). · High neutron irradiation test by fission reactor at international cooperation. Liquid Blanket System : · Liquid lithium 1) Compatibility tests between liquid lithium and another materials by heat flow loop test at 200 Liters. 2) Fabrication and evaluation of insulation coating for the reduction of MHD pressure. · Lithium-lead alloy 1) Compatibility tests between liquid lithium and another materials. · Molten salt (FLiBe, etc.) 1) Compatibility tests between liquid lithium and another materials by heat flow loop test.	The same as demonstration reactor.
Subjects	Solid Blanket System : · Development of the fabrication method for tritium breeder pebbles	Solid Blanket System : · Development of pebble fabrication by wet process. · Development of improved materials. · Stability evaluation of neutron multiplier at high-neutron irradiation and at neutron irradiation by neutron spectrum of fusion reactor (including the determination of the material specification).	The same as demonstration reactor.

		· Development of ^{6}Li enrichment technique and recycle technique. Liquid Blanket System: · Common research and development of heat flow behavior, chemical compatibility of liquid lithium, tritium recovery behavior, etc. by IFMIF · Start of common research and development on liquid lithium and molten salt (consideration of the application for advanced divertor and first wall). · Development of irradiation capsule for irradiation tests with liquid metals and molten salt.	
The concrete solution	Solid Blanket System : · Fabrication development of large-size pebbles by wet process with dehydration reaction.	· Same as the experiment reactor. · Characterization of improved materials. · High-neutron irradiation tests with IFMIF by international cooperation. · Development of lithium isotope separation with lithium ionic conductor.	The same as demonstration reactor.
The prospect	Achievement possible	Prospect of achievement at about 2010.	Prospect of achievement

*This table is the example of Li_2O, it will be changed as the kind of tritium breeder.

(Neutron multiplier)

Items	Experimental reactor (ITER)	Demonstration reactor	Commercial reactor
Neutron Multiplier	Material : Be Shape : Pebble	Material : Be or advanced material (beryllium intermetallic compounds) Shape : Pebble	The same as demonstration reactor.
Technical target	· Mass production development of neutron multiplier pebbles (large size pebble : φ1 2mm, small size pebble : φ0.1 0.2mm) (Amount of tritium breeders : about 150 tons) · Stability of neutron multiplier at 3000appmHe, 3dpa and 150 350˚C (No crack, swelling, etc.).	· Mass production development of neutron multiplier pebbles (large size pebble : φ1 2mm, small size pebble : φ0.1 0.2mm) (Amount of tritium breeders : about 300 tons) · Stability of neutron multiplier at 20000appmHe, 20dpa and 400 900˚C (No crack, swelling, etc.).	The same as demonstration reactor.
Status	· Pebble fabrication by the rotating electrode method (120kg/year). · Stability of neutron multiplier irradiated at the condition of 3000appmHe and 30dpa.	· Development of pebble fabrication by the rotating electrode method. · Development of beryllium intermetallic compounds. · High neutron irradiation tests by fission reactor at international cooperation.	The same as demonstration reactor.
Subjects	none	· Development of beryllium intermetallic compounds. · Stability evaluation of neutron multiplier at high-neutron irradiation and at neutron irradiation by neutron spectrum of fusion reactor (including the determination of the material specification). · Development of beryllium reprocessing technology.	The same as demonstration reactor.
The concrete solution		· Fabrication development of beryllium intermetallic compounds by the rotating electrode method. · High-neutron irradiation tests with fission reactor and IFMIF by international cooperation. · Developments of beryllium reprocessing with dry process.	The same as demonstration reactor.
Prospect	Achievement possible	Prospect of achievement at about 2010.	Prospect of achievement

3) Future Subjects and Forecast

For the demonstration and commercial reactors, the conditions surrounding the blanket will be much more severe than conditions in ITER, where more huge amount of materials for tritium breeding will be used. For example, severe irradiation damage, high heat flux on first wall surface, and high temperature operation to produce the energy will all have to be endured.

Addressing the materials for tritium breeding, the production of tritium by the same process will be desired from the viewpoint of the reducing the production cost. For this purpose, R&D for large pebble manufacturing by the wet process with the dehydration reaction will be carried out. Further, R&D for small pebble manufacturing and an evaluation of grain size refinement will be started. In addition, it is urgent that the properties of the tritium breeders under the DEMO reactor conditions be clarified. Therefore, neutron irradiation tests using a fission reactor will be effectively accomplished by utilizing international collaboration like the ISTC. Since a large volume of tritium breeders will be required in the future, development of ^{6}Li-enrichment technology and lithium recycling technology for the effective utilization of resources should be continued. In addition, after the future materials selection, the irradiation tests with the 14-MeV-neutron source will have to be carried out to verify the major properties under the actual fusion reactor conditions.

R&D to determine the thermal convection properties, the chemical properties, and the recovered tritium properties have many common subjects with the development of the strong neutron target, and thus they should proceed based on a consistent plan. Since it is important to develop a material coating for the liquid breeder, systematic R&D should be performed based on the assumption that it will be used in fusion reactor conditions. Research on both liquid lithium and molten salt (FLiBe, etc.) should proceed intensively from the common subject viewpoint because the application of these materials to the advanced liquid divertor and the first wall is considered. Liquid blanket material R&D is necessary to investigate the neutron irradiation effects of liquid/coating material/piping materials systems. The development of the capsule for fission reactor irradiation has to proceed while keeping abreast with the other R&D activities. For the molten salt, it is necessary to have basic studies of tritium confinement and the control of corrosion, and this should be accomplished by effective utilization of international collaboration.

Concerning the neutron multiplier, evaluation of the behavior under high neutron irradiation while adjusting the ratio of helium production and dpa is necessary.

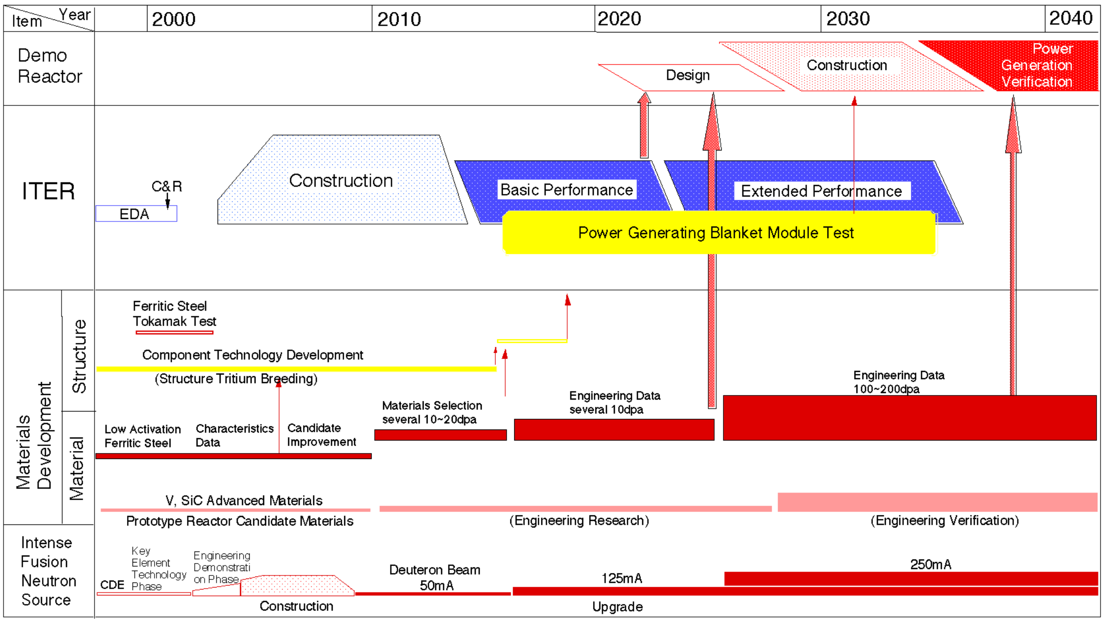
Since it is impossible to conduct these irradiation tests in Japan and also from the viewpoint of effective utilization of resources, the irradiation test utilizing international collaboration, like the ISTC, is necessary. For the development of beryllium intermetallics, it is necessary to develop pebble production technology. Also, it is necessary to develop the recycling technology using a dry process, while keeping the effective utilization of resources in mind.

(4) Schedule of Materials Development and the Role of the Neutron Source

An overview of the strategy of fusion reactor materials development is shown in Fig. 3.3.2-5, where the relationship of the reactor development and the neutron source program is shown. The schedule of materials development is expressed in two regimes, i.e., the development of raw materials and the development of functional structure technologies. The task of raw materials development consists of determining the detailed composition of alloys, establishing fabrication processes, evaluating comprehensively the properties of the candidate materials, establishing welding and other joining methods, determining coolant and breeding materials compatibility with the candidate materials, and estimating the neutron irradiation effects on the various properties of the candidate materials. The development of functional structure relates to the fabrication of components and it consists of the fabrication of the blanket component structure, the characteristics testing of blanket functions, the evaluation of the ferromagnetic effects of the ferritic steel structure, etc.

The fusion materials intense neutron source is an accelerator-based deuterium-lithium neutron source studied under the international collaboration (IFMIF: International Fusion Material Irradiation Facility). The construction schedule for this facility is based on the staged construction approach as agreed in the IEA international cooperation activity. The objectives are: 1) development of materials that can be used under intense neutron irradiation by controlling the growth of microscopic structures specific to the high-energy being applied, 2) acquisition of data on property changes due to fusion neutrons for various device and component materials, and 3) establishment of engineering design data for the DEMO reactor. Such an intense neutron source requires the establishment of the accelerator, the target, irradiation technology, and post-irradiation test technology. The operation of a 50-mA deuteron beam current should be attained as early as possible through component development because the early commencement of irradiation tests using high-energy neutrons is necessary.

On the other hand, considering the operating conditions of a fusion reactor, it is important to account for the behavior of materials under the variations of temperature and other parameters and the dynamic phenomena that only appear during irradiation. The irradiation tests under varying conditions and the in situ tests of the characteristics of irradiated materials have so far been studied by using fission reactors. The information derived from these investigations along with the development of small specimen test techniques will be indispensable in performing the described fusion neutron irradiation tests. The key issue to efficiently progress to the final blanket structure from the raw materials development stage is to integrate the various structural, coolant, breeding, neutron multipliers, etc., components. These components need to function as a system in the fusion reactor environment. This issue includes the joining methods of the different kinds of materials, the compatibility among the gas-, liquid-, and solid-phase materials, and the soundness of these characteristics under the neutron irradiation. Making an outlook on the integration of these materials is necessary prior to the fusion neutron source development. Therefore, the early start of the irradiation work under the international collaboration scheme is required at a fission reactor where the irradiation test of the relatively large sample is possible.



*dpa: integrated neutron damage parameter to study the effect of neutron irradiation on the materials characteristics

Fig. 3.3.2-5 Schedule of fusion materials development

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3.4 Present status of safety related technologies and future issues

3.4.1 Safety characteristics of fusion reactors

The safety attractiveness of fusion reactors is characterized by its unique and inherent safety features that are derived from the fundamental principles such as there can be no nuclear chain reaction excursion, there will be a passive termination of the fusion reaction under any disturbance due to operation limits constraints, and the decay heat density is low. Accordingly, the safety functions for terminating nuclear reactions and cooling the reactor to avoid melting, which are essential in fission reactors, are low priority technical requirements in fusion reactors. For this reason, the structural integrity of the enclosure or confinement of the radioactive materials is the most important requirement for ensuring safety. The following characteristics are generally common to fusion reactors and are independent of the type of plasma confinement used.

1) No nuclear excursion and no nuclear criticality accident

In fission reactors and facilities handling fissile materials, prevention of nuclear excursion is imperative by the quantitative management of fissile materials. Particularly in facilities handling fissile materials, the amount of fissile material must be maintained below its critical mass. This is because in a "criticality" condition a chain reaction is sustained, where neutrons generated by fission reactions produce progressively next fission reactions. It is, therefore, necessary to regulate with extreme attention the quantitative management of fissile materials from fuel processing through waste treatment. In case of fusion, on the other hand, neutrons and alpha particles generated by the fusion reaction do not produce the next reaction. The result is that there is no "critical mass" for the deuterium and tritium fuel, and thus no nuclear criticality accident can occur.

Magnetic confined fusion is a process dominated by a thermal energy balance, which is similar to ordinary chemical combustion. Although an uncontrolled reaction, such as an explosion and fire, would be of concern even for a chemical reaction process, it can be shown that no possibility of such an excursion exists in the case of fusion for the following reasons.

- Correlation between plasma pressure and magnetic pressure:

In magnetic confined fusion, an inherent property limits the plasma pressure. The ratio, the "beta value," of the plasma pressure (the product of plasma temperature and density) to the magnetic pressure (the square of the magnetic field strength) is limited to a certain value, usually a few percent, by the instability of the Magneto Hydro Dynamics (MHD).

For example, if the fusion reaction were abnormally accelerated for some reason, the plasma pressure would be increased, which would increase the fusion power. When the plasma pressure reached the critical beta value, the plasma particles would dissipate from the "confining cage" made by the magnetic fields, which would result in a lowering of the fusion power or the termination of the fusion reaction.

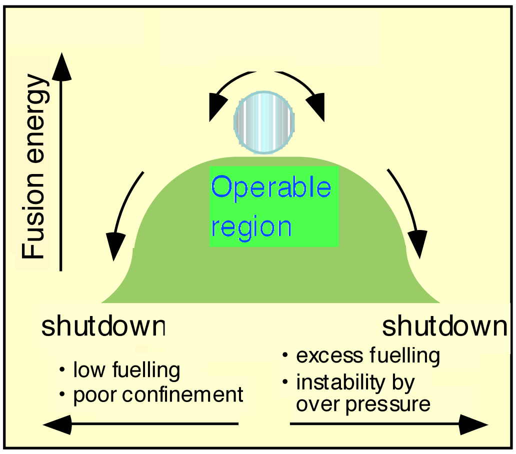

Fig. 3.4-1 Inherent safety responses of the fusion reaction

- Correlation between plasma density and fusion power:

The upper bound of plasma density is also another constraint. For example, when the fueling rate is excessive, the plasma density increases quickly and reaches the density limit. After that, degradation of confinement reduces the fusion power or terminates the fusion reaction.

- Termination of fusion reaction by ingress of impurities, etc.:

The density of plasma is 1/100,000, or less, than the pressure of the atmosphere. Thus, the plasma temperature cannot be maintained with ingress of even a very small amount of impurity, since such ingress would result in a reduction or termination of the fusion reaction. For example, when an over-heating event occurs, a part of the first-wall material is heated up and evaporated owing to the high heat flux onto the first-wall. The evaporated material is mixed into the much hotter plasma, which decreases the plasma temperature, and the fusion reaction is passively terminated. A similar phenomenon will occur when air and/or steam are leaked into the plasma.

2) Low decay heat density

"Decay heat" generated in a fusion reactor from radioactive materials after the operation is terminated is generally smaller than that of the fission products in a fission reactor since the decay heat in a fusion reactor originates only from nuclides activated by neutrons. On the other hand, a major part of the decay heat is generated in the structures surrounding the plasma, such as the blanket and first wall, in the case of magnetic confined fusion. The mass and volume of such structure is comparatively large with respect to the total decay heat, so the decay heat density becomes low. As a result, heat removal is easily attained and the safety function can be passively maintained under abnormal events. In a light water reactor, active safety measures such as the emergency cooling system are inevitably required to avoid core melting due to the large decay heat. In contrast, in ITER for example, natural convection cooling can achieve decay heat removal even if forced convection cooling is stopped.

3) Dispersed existence of mobile radioactive materials

- In a fusion plant, tritium is dispersed and contained in the vacuum vessel and fuel processing systems for breeding, supplement, collection, and purification. The activated products are also contained in the vacuum vessel and cooling systems. Considering those conditions, the safety measures to cope with postulated accidents where radioactive materials are released from the enclosure are essential. This is a key engineering issue for ensuring the safety in a fusion reactor.
- In general, the radioactive materials are confined by a series of measures, which is usually based on the integrity assurance of the enclosure that contains radioactive materials, passive safety or fail-safe features, and the engineering safety measures for clean-up of radioactive materials and for prevention of contamination dispersion. In a fusion reactor, the amount of the radioactive materials to be released in an accident can be limited because of the dispersed existence of the radioactive materials, though the enclosure boundary is located in relatively wide regions and the boundary structure tends to become complicated. In addition, as previously mentioned, the hazard potential of radioactive material inventory is significantly small compared with that of a fission reactor. This fact implies the possibility that the requirements for the engineering safety measures for radioactive material confinement/clean-up functions become moderate.

4) Human error safety concerns

The most important issue for controlling the fission reaction is to prevent the escalation of abnormal events and accidents so as to avoid an excursion by an uncontrolled reaction. An accident may occur as an escalation of an abnormal event when safety functions are overridden or canceled. The concepts of fail-safe and foolproof are adopted to prevent such a situation. However, in spite of the fact that these concepts were adopted in nuclear fission plants, accidents actually occurred.

In the case of fusion reactors, workers and operators have to have knowledge and judgment through appropriate education and training for handling radioactive materials. However, the deuterium and tritium fusion fuels do not need management of the critical mass during the entire operating cycle, since no nuclear excursion can occur, as described before. In addition, an abnormal event cannot escalate to an accident or an excursion, even though a human error may cause a fluctuation in fusion power or stop the reaction. Even if the radioactive materials contained within the reactor are released inside the building by human error, this event may cease as a single event, and the effects of the released material can be maintained sufficiently below the regulatory limit by multiple safety functions.

Moreover, since the hazard potential in a fusion plant is small, it is relatively easy to design a fusion plant such that the exposure for workers and the environment is small even in the event of a radioactive material release caused by multiple human errors. For example, it is technically possible to assure that evacuation of the public is not needed even in an accident where a radioactive material release is much more than that resulting from any design-basis accident. Of course, this does not mean that the effort to promote the safety consciousness of workers and operators can be belittled. A positive attitude toward greater safety is indispensable in an organization operating a large energy system.

3.4.2 Issues for ensuring safety

Issues and technical measures for ensuring safety in a fusion reactor are discussed in this section taking ITER as an example. The fundamental approach is to maintain the confinement of radioactive materials, as mentioned above.

3.4.2.1 Distribution of radioactive materials such as tritium in a tokamak

According to the final design report (FDR) of the ITER engineering design activity until July 1998, the distribution of major radioactive materials in the ITER facility is as shown in Fig. 3.4-2 and as described below. It should be noted that these values will be reduced for the current design since the machine size and fusion power are reduced in comparison with the FDR.

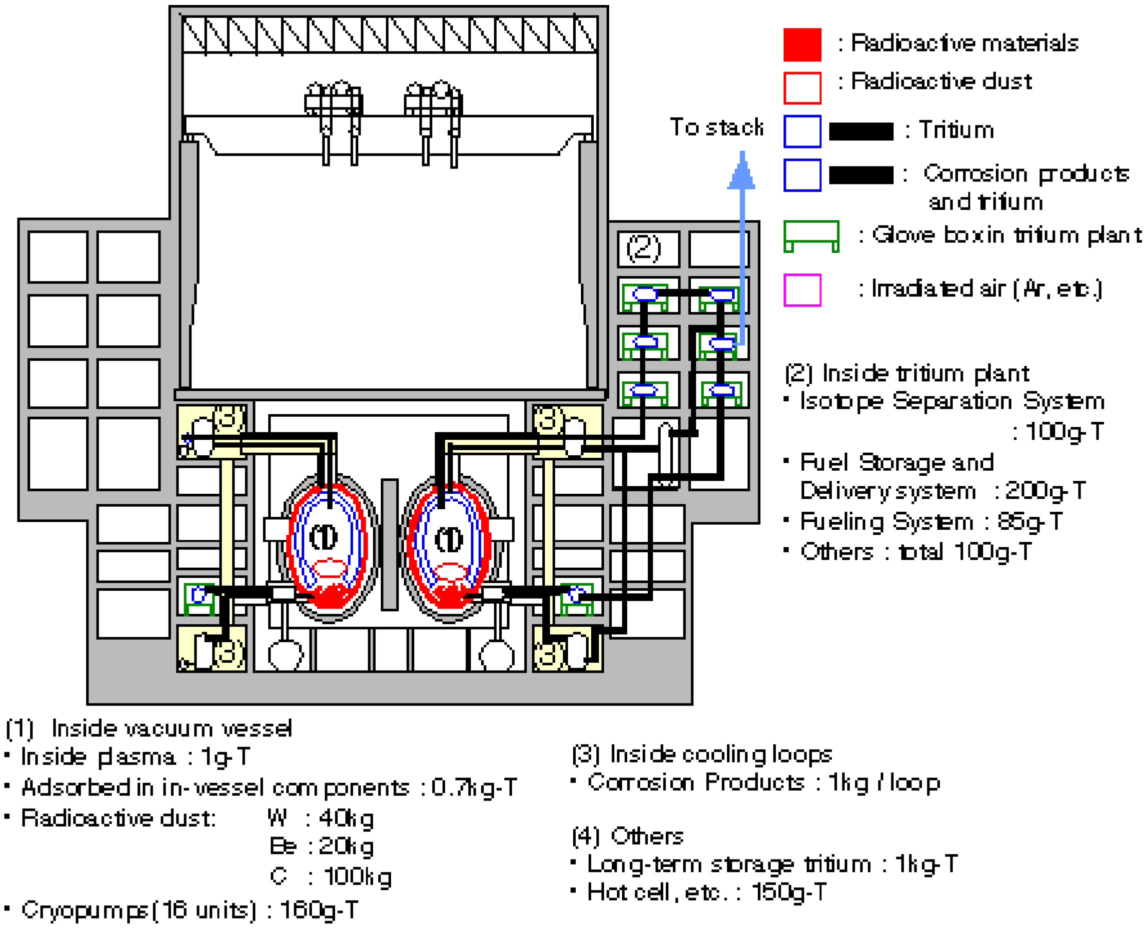

Fig. 3.4-2
Distribution of major radioactive materials in tokamak building (based on FDR)
(Tritium inventory inside the vacuum vessel will be reduced about half in the current compact-type-ITER)

(1) Inside the vacuum vessel:

- Tritium inventory in the plasma: < 1 g
- Tritium inventory occluded in the in-vessel components: < 1.2 kg
- Radioactive dust: Tungsten < 100 kg,
  Beryllium< 100 kg,
  Carbon < 200 kg.
  (fine-grain particles caused by erosion of the plasma facing materials, and accumulated on the bottom of the vacuum vessel)
- Tritium inventory adsorbed in the cryopanel in the cryopump: < 200 g

(2) Inside the tritium plant:

- Tritium gas of the hydrogen isotope separation system: < 230 g
- Tritium gas occluded in metal in the fuel storage and delivery system: < 320 g
- Tritium gas and tritium in the solid pellets in the fueling system: < 85 g
- Tritium in the other tritium subsystems: < 100 g

3.4.2.2 Thermal and magnetic energy specific to the tokamak type fusion device

ITER is associated with distinctive energy categories and the energy sources listed in Table 3.4-1, which are different from the energy categories and sources experienced in ordinary facilities. Safety measures for these energy sources should be considered to ensure the integrity of the enclosures that contain radioactive materials. Although the sources in Table 3.4-1 include the energy that does not result from the fusion reaction, the level of energy is quite large. Thus, it would be a technological issue in terms of safety if the energy has the potential to damage the enclosure (e.g., vacuum vessel). Typically, electromagnetic forces, and the pressure rise due to evaporation, of leaked coolant in the vacuum vessel and of leaked helium for cooling the superconducting magnets, are to be assessed.

Table 3.4-1 Categories and distribution of energy sources in ITER (provisional values)

Category		Energy		
Vacuum Vessel	Plasma	Fusion power		0.5 (GW)
		Thermal energy		0.4 (GJ)
		Magnetic energy		0.4 (GJ)
Superconducting magnet	Coil	Magnetic energy	Toroidal field coil	44 (GJ)
			Poloidal field coil	8 (GJ)
Tritium system	In vacuum vessel	Chemical energy		< 0.1 (GJ)
	Fuel cycle system	Chemical energy		

Note: These values are provisional and in accordance with the progress of the compact type ITER design. Energy in tritium system corresponds to the burning (chemical) energy of hydrogen isotope gas (deuterium and tritium).

(i) Energy sources involved in plasma:

The plasma, which is a rarefied gas having a very high temperature, is maintained inside the vacuum vessel, and has a thermal energy and a magnetic energy originated by plasma current (~ 4×10^{8} J each type of energy). The energy is released, within a short time period, on the vacuum vessel and in-vessel components such as the divertor during a rapid disappearance of the plasma (disruption). Consequently, the vacuum vessel and in-vessel components are designed to endure an excessive number of disruptions during operation. No safety function is required for the components located inside the vacuum vessel, since the vacuum vessel is the enclosure to ensure the containment of tritium.

(ii) Magnetic energy in the superconducting coils:

The total magnetic energy stored in the superconducting coils is approximately 5×10^{10} J. When the superconducting state is failed (quenched), the stored energy is normally released to the external resistance and absorbed as joule heat. Even in the worst scenario, the superconducting coils would not affect the containment function of the vacuum vessel, since the coils are designed to have sufficient strength for the severe operating loads, and the clearance provided between the vacuum vessel and the coils can avoid mechanical contact of them.

(iii) Chemical energy of hydrogen isotopes (deuterium and tritium):

The hydrogen isotopes can enter into a chemical combustion reaction with oxygen. The hydrogen isotopes handled as fuel in the vacuum vessel and in the fueling systems are estimated to total about an order of one kilogram. If all hydrogen isotopes were burned, the generated energy would be less than 1×10^8 J, which is relatively small. In addition, the combustion reaction can be prevented by multiple safety measures such as a physical barrier (enclosure), inert gas or vacuum surrounding the enclosure, and the limiting of oxygen ingress by isolation valves. The facility is designed to maintain necessary safety functions even if combustion should occur.

3.4.2.3 Containment/confinement of the radioactive materials in ITER

The safety issue for a fusion reactor is to reduce the release of mobile radioactive materials to amounts as low as practicable in an abnormal event or an accident, which may result from an equipment malfunction or stored energies, by utilizing inherent safety characteristics of fusion as previously mentioned, and to limit radiation exposure of the public and workers below prescribed levels.

Safety analyses assuming the following conditions have been conducted in the ITER-FDR and countermeasures preventing the radiological risks have been developed.

- Four kilograms tritium in the ITER facility
- Maximum 1.2 kg of tritium occluded in the vacuum vessel
- Energy sources as listed in Table 3.4-1
- Radioactive waste resulting from neutron irradiation

The safety goal of ITER is to design, construct, and operate the ITER facility, aiming at protecting the public and workers from the radiological risks and demonstrating a high level of safety attractiveness for future fusion power plants.

To fulfill this safety goal, the following safety principles are adopted in ITER.

(i) Under normal operating conditions, the release of radioactive materials to the environment shall be controlled and maintained as low as reasonably achievable, and the worker exposure shall be controlled with appropriate management.

(ii) In case of an accident, excessive release of radioactive materials to the environment shall be prevented by means of the confinement facility with the emergency clean-up system, as shown in Fig. 3.4-3

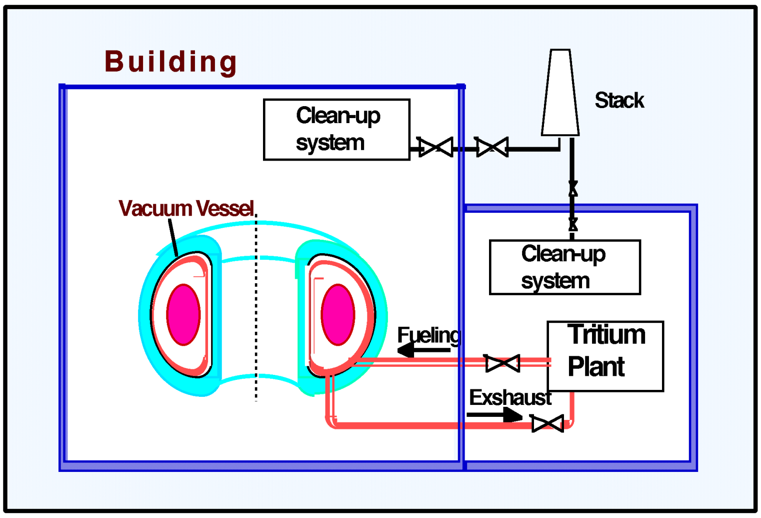

Fig. 3.4-3 Containment/confinement concept of radioactive materials in ITER (Major containment boundaries are the vacuum vessel, the tritium plant, piping, etc.)

3.4.2.4 Safety analyses of ITER

For ensuring radiological safety, a quantitative assessment by analyses is necessary as well as the adoption of engineering countermeasures. In ITER, considering the inherent safety features mentioned previously, the attainment of safety has been pursued with the containment and confinement concept along with the defense-in-depth philosophy. In the design activity of ITER, the abnormal events and accidents, 25 events in total (11 types), were categorized into four groups with the probability of every event being greater than 10^{-6} /year. The consequences of these events and accidents were quantitatively assessed taking into account the event sequence. As a result, it has been confirmed that in all postulated abnormal events and accidents addressed in the safety design of ITER, the released tritium was sufficiently below 100 g, which is defined as the project guideline for the tritium release limit for protecting the public from radiological risks. As a whole, the adequacy of the ITER safety design has been well confirmed.

According to the estimate of the radiation dose (effective dose equivalent) for a unit mass of tritium released to the environment, the dose due to the tritium in the form of HTO was about 10 μSv/g-T assuming the release was from a stack (100-m high) and with typical Japanese weather conditions. The public exposure dose even in an accident with a maximum release beyond the design basis accident was confirmed to be within the limit (< 50 mSv) recommended by IAEA for not requiring public evacuation.

A similar analysis was also performed for radioactive dust. The amount of released dust was confirmed to be sufficiently below the limit of 500 g established as a project guideline. These results indicate that the hazard potential of the fusion reactor (ITER) is very small.

For the normal operation of ITER, the project guideline release limits were defined to be less than 1 g/year for tritium and less than 0.5 g/year for radioactive dust. The analysis results have shown that the release of tritium and radioactive dust is about 0.3 g/year and 0.4 g/year, respectively, which are below the guideline release limits.

The safety analyses described above have been carried out to evaluate the safety characteristics of ITER as a part of the design activity. Therefore, these safety analyses were not directly corresponding to the regulatory-based analyses and criteria required for construction in Japan. However, it can be stated that the ITER design demonstrates a high level of assurance for the radiological safety against the hazard potential of this fusion reactor.

3.4.3 Present status of the safety research for ITER

Accumulation of qualified data, which is required for the regulatory application, is an important and urgent issue for the construction of ITER. It is also important to establish the codes and standards that will be used as the technical basis for the design and fabrication.

For this purpose, R&D for several technologies has been performed in the ITER program for the qualification and assurance of component integrity. In addition, safety related R&D has been conducted for the accumulation of engineering data, and for the development and validation of analysis codes for the safety analyses. Furthermore, the following R&D activities have been promoted as high priority domestic activities in Japan, since they are essential to gain regulatory permission in addition to being achievements of ITER R&D.

(i) Simulation test of tritium release
(ii) Integrated thermal hydraulics test in the vacuum vessel
(iii) Qualification of the double-walled vacuum vessel
(iv) Verification of the safety of the superconducting coil
(v) Demonstration of seismic isolation

The safety research of the fusion reactor has progressed well, so the safety characteristics can be quantitatively assessed in detail. However, most analysis codes utilized have been modifications of codes that were originally developed for fission reactors or general purpose analyses. Thus, to pursue the regulatory application for construction, further qualification is necessary to validate the appropriateness of the modification of the codes and the reliability of the utilized data for the application to the fusion reactor.

3.4.4 Safety-related issues for future fusion reactors including DEMO reactor

The demonstration (DEMO) and commercial reactors will install high-temperature tritium breeding blankets for energy production, which is different from ITER in terms of safety. However, the fundamental aspect of fusion safety would not change, although new technological research and development to ensure the safety related with the blanket would be required. Therefore, it can be said that demonstrating the safety of ITER through each stage of construction, operation, and decommissioning establishes the technological basis and methodology of safety assurance for fusion power reactors including the DEMO reactor.

The fusion power reactors (DEMO and commercial) are different from ITER in the following respects. The breeder blanket is installed for tritium breeding and power generation, so the tritium inventory would be increased. As a prediction from the present design, for example, the maximum tritium inventory in the blanket would be several hundred grams, although it cannot be estimated accurately because it strongly depends on the type of blanket. The containment boundary for radioactive materials penetrates the vacuum vessel and extends to the tritium recovery system and the cooling system, which is different from the ITER system. On the other hand, the tritium inventory in the plasma facing components would be significantly decreased since tritium-retaining carbon-based materials would not be used, resulting in almost no tritium in the co-deposited layer.

Increasing the operating temperature and improving the economic efficiency is essential for the fusion power reactors, and this introduces a new safety-related issue. In addition, the development of low-activation materials and reduction of impurities that form long-lived nuclides can drastically decrease radioactive wastes.

In the fusion power reactor, the environmental effects and the safety would be evaluated as an integrated plant system, including the demonstration of the tritium breeding cycle, tritium recovery from the tritium containing radioactive waste, and the waste treatment processes. The advantages of the fueling cycle of the fusion reactor can be demonstrated in this way. However, this requires further R&D since these evaluations include presently undeveloped technology and require integration into the total system.

As discussed above, some safety-related issues of fusion power reactors including the DEMO reactor can be less serious and some can be more serious in comparison with those of ITER. In general, however, it can be said that major fundamental safety issues specific to fusion reactor technology would be resolved through ITER construction and operation. The blanket and the components of the fueling cycle are the processing systems of the radioactive materials and thus the safety assurance of those systems is the fundamental requirement for the system development stage. Those issues are not technically specific to fusion but can be solved in every process as thermal and chemical problems.

In summary, the safety-related issues for fusion power reactors including the DEMO reactor are summarized below.

(i) Improvement of the economic efficiency

- High power density: active removal of decay heat
- High availability: efficient maintenance
- High efficiency: materials and coolants for high temperature use   individual issue

(ii) Improvement of environmental and safety characteristics

- Prevention of tritium permeation
- Reduction of tritium inventory
- Material development   low activation, reduction of radioactive waste

3.5Present status of operation and maintenance technologies and future issues

The fusion reactor will be put to practical use after the phased development of an experimental reactor, a demonstration reactor, and a prototype reactor. Naturally, the requirements for efficient operation and maintenance of the commercial fusion reactor will be equivalent to those of existing power plants, namely, the construction and commissioning periods should be as short as possible, and high efficiency and high reliability (= high availability) should be guaranteed during operations. What are the issues that should be addressed at each development stage to meet these requirements? To answer this question, the present status of operation and maintenance technologies and related issues will be discussed in this section.

This examination is carried out as follows. Operation and maintenance conditions of the fusion commercial reactor are discussed using commercial light water fission reactors (LWRs) [3.5-1,2] as a reference, and the feasibility is examined for a fusion commercial reactor on the basis of available data. This examination makes reference to the data from JT-60 [3.5-3] as an existing large tokamak device, and to the data from the Final Design Report (FDR) [3.5-4] of ITER or from the compact ITER as a planned fusion experimental reactor. As a reference for a demonstration reactor, data from the steady-state tokamak reactor SSTR [3.5-5] is used.

3.5.1 Operation and maintenance conditions of commercial LWRs as a reference [3.5-1,2]

A quantitative assessment is made on operation and maintenance conditions necessary for commercial fusion reactors using the data of commercial LWRs as a reference.

a) Construction and start-up test periods:

The construction period of an LWR spans about 4 years, encompassing the start of bedrock inspection and base mat construction for buildings to the loading of fuel. This period is followed by a start-up test period of about one year during which the soundness of the reactor and ancillary systems is confirmed. These start-up tests are performed after the plant has already been completed. The start-up tests are carried out to confirm the function of various protection systems while the output of the reactor is progressively increased from 0 to 100%. These tests include: (i) the fuel loading (+ atmospheric pressure test (BWR)), (ii) the zero output test (PWR)/nuclear heating test (BWR), (iii) the power ascension test, and (iv) the output demonstration test (comprehensive test with an actual load). Following these tests, the operation is continued with 100% output. After the start-up period, the reactor is allowed to begin commercial operation.

b) Capacity factor

In 1997, the capacity factor of LWRs reached about 83% in Japan and in Germany, about 70% in France, and about 61% even in Canada, where the capacity factor is lowest among the developed countries [3.5-6]. The capacity factor in Japan is in the range of 80% on the annual average, which is a remarkable record. This means most plants operate at an almost 100% capacity factor except for the periodic inspections.

c) Operators:

According to Reference [3.5-1], an LWR is operated in three shifts, and all operators work in the central control room. One operating group consists of one supervisor, one sub-supervisor, one chief operator, one sub-chief operator, one main-system operator, and one or two auxiliary- system operators, for a total of six or seven operators. Where two reactor units are controlled from one control room, the total number of operators is increased to ten or eleven with two chief operators, two main-system operators, and three or four auxiliary-system operators. A three-shift system with six operating groups is adopted. Generally, the number of operators in an operating group during the inspection period remains unchanged from the operation period though the roles of operators change.

d) Periodic inspection:

The periodic inspection is imposed on the reactor and ancillary systems every 13 months and on the steam turbine every 25 months. The period from power disconnection to power reconnection (off-line period) is from 50 days to 70 days for a standard periodic check. Therefore, continuous efforts are being directed toward reducing the duration of the periodic inspection to improve the capacity factor. During the periodic inspection, the average radiation exposure level of workers is about 2-3 person Sv/reactor, and is 1 mSv for one worker.

e) Unplanned shutdown:

Machine troubles are the main causes of lowered capacity factors, except for the periodic inspections. The frequency of unplanned shutdowns in Japan is very low, about 0.2 times/reactor/year (in 1995). In France, where the number of reactors is the highest among the developed countries, the frequency of unplanned shutdowns was 4 times/reactor/year (in 1995), and the capacity factor was 75%. The capacity factor was the lowest in Canada (61%). The number of troubles (reported on the basis of laws and regulations) is also very small in Japan, and was 0.5 events/reactor/year (in 1997). The performance record of LWRs is summarized in Table 3.5-1.

Table 3.5-1 Performance record of LWRs as a reference for the fusion reactor

Items	Working results
Start-up test	The output of 100% was confirmed within a year after the completion of individual equipment tests
Capacity factor	> 60-83%
Operator	6-7 persons/unit, (10-11 persons/2 units)
Periodic inspection	< 50-70 days/unit/year, (worker's exposure ~ 2-3 persons Sv/unit)
Unplanned stop/trouble	< 0.2-4.0 times/unit/year, (<0.5 times/unit/year in Japan)

3.5.2 Projected operation and maintenance conditions in fusion reactors

The high capacity factor of LWRs shown in Table 3.5-1 has been achieved after many laborious years, solving various technical problems and optimizing the systems from their early rudimentary stages following the introduction of nuclear power generation. Therefore, it would be difficult to accomplish these records immediately after a fusion reactor is put to commercial use. However, for the discussion of the technical feasibility of meeting the performance results of LWRs, adoption of these values as a guide for comparison would probably not incur a serious qualitative problem.

A commercial fusion reactor will be required to achieve a capacity factor equivalent to that of LWRs. Is this capacity factor feasible based on the present technology level? To investigate this point, plant characteristics are presented at each stage of reactor development: large tokamak device, experimental reactor, demonstration reactor, and prototype/commercial reactor, and the conditions for fusion reactors are defined in relation to those for LWRs. Next, the possibilities and issues for each operation and maintenance condition are discussed mainly on the basis of the JT-60 operation record [3.5-3] and the ITER final design report (FDR) [3.5-4]. Operational parameters and conditions of fusion devices at their development stages are shown in Table 3.5-2. The operational parameters and conditions that exert a strong influence on the operation depend on each stage. Therefore, progress in development may uncover new critical elements. Especially, the shift from the present large tokamak device to the experimental reactor ITER will witness changes in attributes in the last five rows of Table 3.5-2, in addition to the device size. These changes will have a significant influence on operation.

Table 3.5-2 Operational parameters and elements of fusion devices

Items	Large tokamak JT-60	ITER	Demonstration reactor (e.g. SSTR)	Prototype/ Commercial reactor
Plasma current	< 5 MA	13-17 MA	12 MA	12 MA
Major radius	~ 3 m	6.0-6.5 m	~ 7 m	~ 7 m
Toroidal magnetic field	4 T (10 T max.)	< 6 T (12 T max.)	9 T (16 T max.)	9 T (16 T max.)
Fusion power output	~ 10 MW equivalent	>0.5 GW	~ 3 GW	3-5 GW
Neutron load to wall	---	>0.5 MW/m^2	~ 3 MW/m^2	~ 5 MW/m^2
Pulse length / Steady state	15 s	300 – 500 s	Steady	Steady
Normal Conduction / Superconducting	Normal conduction	Super-Conducting	Super-Conducting	Super-Conducting
Blanket	None	Use (test)	Use	Use
Tritium	No use	Use	Use	Use

Here, the operation and maintenance conditions in fusion reactors corresponding to those in LWRs are defined.

a) Start-up tests:

The start-up tests in LWRs are considered to correspond to the commissioning of fusion reactors. In the commissioning, all systems are operated in the same condition as in the actual operation. Control system tests, measurement system tests, protection system tests, energization tests of each coil, and tests of the fuel supply system, etc., are carried out step by step. The commissioning is carried out first without plasma, next with plasma, and ends with a final test confirming its function at full power.

b) Capacity factor:

Similar to LWRs, the capacity factor of fusion reactors is defined as the percentage of time when the reactor is producing power. It does not include the period in which power generation is stopped due to periodic inspections or troubles, etc. The number of operators is defined as the number of operators necessary for monitoring the operation of reactor during steady state operation.

c) Periodic inspections and maintenance:

The items for periodic inspection and maintenance of fusion reactors include exchange of blankets, divertors, and heat resisting materials, and inspection of coils, refrigerators, control and monitoring systems, auxiliary machinery, and turbines. These have a similarity to LWR tasks. However, unlike LWRs, fusion reactors are associated with tritium and components activated by 14-MeV neutrons. Therefore, remote maintenance is indispensable to reduce the inspection work radiation exposure to the level of LWRs.

d) Troubles:

A trouble is defined as an event that will or might cause an unplanned shutdown of power generation. Many components of fusion reactors may cause troubles by themselves or under the influence of other components or plasma. Minimizing the number of unplanned shutdowns requires prudent designs, including redundant instruments like are found in LWRs. As in LWRs, redundancy and diversity are essential to ensure the reliability of a fusion reactor as a whole.

(1) Start-up tests (commissioning)

A fusion reactor will be commissioned as follows. After the completion of equipment assembly and installation, the soundness of all equipment is confirmed by tests without plasma. Next, power-ascension tests, etc., are performed in tests with plasma. Here, we will examine these tests for fusion reactors with reference to the JT-60 records and the commissioning plan of ITER (FDR).

(i) Large tokamak JT-60 case

(a) Coil energization tests:

There were 26 items tested and a total of 680 energization shots were made. The individual and combined energization tests of the toroidal/poloidal field coils were carried out with gradual increases in the coil currents.

(b) Comprehensive functional tests:

There were 40 items tested and a total 150 energization shots were made. All systems were operated as they would in real operation, and the soundness of all the systems including the protection system was confirmed. Next, the tests proceeded to the production of the first plasma. The total period for tests (a) and (b) spanned 4 months (Dec.1984 – Mar. 1985).

Since there was no experience in making such a large tokamak operational at that time, a 4-month test period was scheduled. This had a sufficient margin to progressively conduct the tests and to correct and repair unexpected troubles. Taking this situation into account, optimization may reduce the test period and number of test items. For example, the period of the coil energization tests can be shortened by 50%, i.e., to about two months if the tests are performed twice as fast. JT-60 was later modified, the JT-60 upgrade, and all the poloidal field coils were replaced with new coils. The energization tests were performed again on the new coils. Although a total of 70 shots were made, the tests were completed in only 10 days.

(ii) Experimental reactor ITER (FDR) case [3.5-4]

(a) Commissioning without plasma:

The number of the test items is about 30. The period is about one year.

(b) Commissioning with plasma:

After completion of comprehensive tests of all systems as a fusion reactor, the first plasma is produced and commissioning proceeds with plasma.

In the case of ITER, two and half years of commissioning are planned for trial-and-error adjustment of various plasma control parameters using hydrogen plasmas. Such a test period is considered necessary for this experimental reactor stage because the scenario for achieving target plasma performance is still not definite. However, once the plasma production scenario is established, it will be possible to reach the target performance in a short time by using computer control techniques. For example, JT-60 took only 1.5 months to progress form first plasma production to a 1.6-MA plasma current. This suggests that the commissioning can be completed very rapidly once a knowledge-based control system is established since such a system will leave little uncertainty. Therefore, as a matter of course, the scenario for plasma production and the adjustment procedures have to be standardized before a commercial reactor is built.

The major goals of kinetics control in a fusion reactor are control of the fusion reaction rate and safe shutdown control. From the present control technology, problems in achieving these goals are surveyed below.

Three fundamental plasma performance parameters related to the control of the fusion reaction rate (basically equivalent to the control of energy multiplication factor) are ion temperature(T_i), electron density (n_e), and energy confinement time (τ_E). Ion temperature and electron density are physical quantities directly measurable. It is possible to treat these quantities as control parameters of feedback control, because they are directly controllable by actuators, e.g., heating systems and gas injection systems, respectively (Fig. 3.5-1).

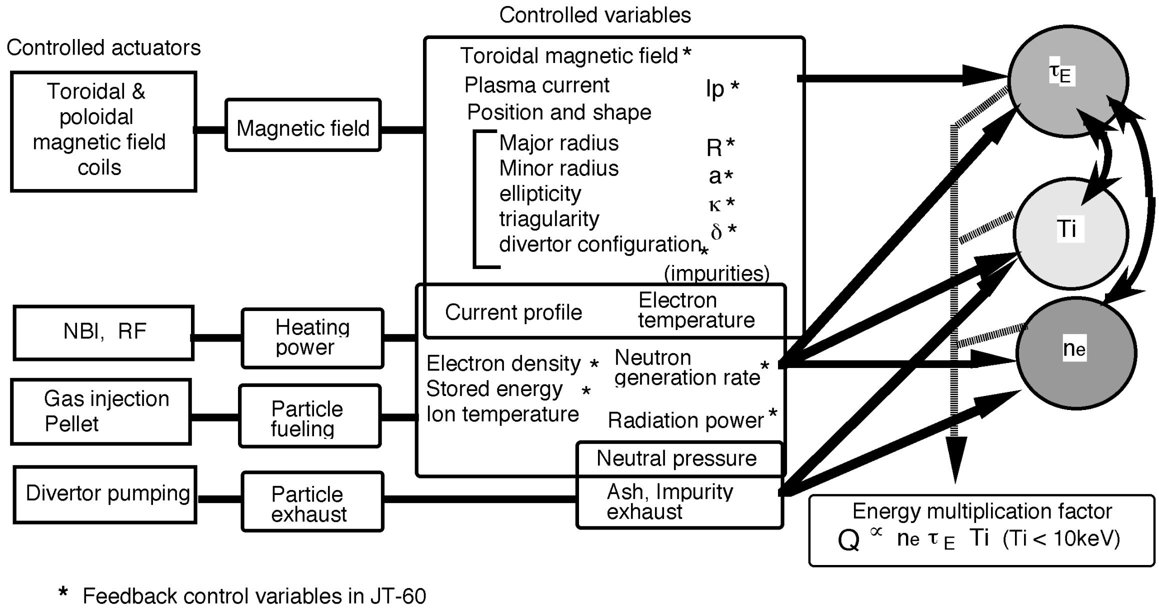

Fig. 3.5-1
Mutual relationship between plasma control and basic quantities relating to plasma performance

In addition, there are aspects that cannot sufficiently be investigated in large tokamak devices, such as the behavior of alpha particles generated by the DT reaction. Furthermore, first wall conditions and eddy current strengths cannot be measured. Their effects on plasma discharges remain important subjects for future investigation.

During the commissioning, as the fusion power-increase (power-ascension) tests progress, the expectations made during the design stage on the evolution of these quantities and the relationships among them will be validated. The ultimate fusion reactor control method will probably be a very sophisticated one, although the operator will only have to choose one from the available operation scenarios. However, to establish such optimized control scenarios, various trial-and-error operations must be conducted for both the experimental reactor and the demonstration reactor.

The safe shutdown control aims to forecast and avoid abnormal phenomena including disruptions, and to safely shutdown the reactor when abnormal events occur in the plasma or the system. This control is one of the functions required for a fusion power plant. Therefore, the shutdown control will actually be tested within the safe bounds of the equipment. Understanding the detailed actions of all systems including plasma remains a future issue. However, once a full understanding is developed, this test will be carried out without any difficulty.

The start-up power required for ITER will be several hundreds of MW. Therefore, the reduction of start-up power is an important issue. The following are important for start-up power reduction: reduction in motor-generator power loss, reduction of the coolant pump power by optimizing the motor/pump design, reduction of the helium refrigerator power by using high-temperature superconductors, development of the control method of high spontaneous current to reduce electric power for heating and current drive, and so on. The electric power may be reduced to a level of one hundred MW. However, it is not easy to reduce the start-up power to the level required by an LWR. For this reason, the electric power system available to the fusion reactor must be stable.

(2) Capacity factor

Availability is an important reliability index of the whole power generating system. Even the present large tokamak is larger and more complicated than an LWR. Therefore, it is necessary to identify the issues that will improve the capacity factor of a fusion reactor.

Here, the performance record or “net working rate” of JT-60 [3.5-3] is described. The operation of JT-60 is planned for 9 cycles per year, each cycle having a period of about two weeks. The total operation duration is about 100 days per year. The period called “periodic inspection” requires 2 - 3 months per year, and the remaining period is used for improvement of equipment, calibration tests, etc. Unlike LWRs, the operation period of JT-60 is not determined as the period excluding the period of periodic inspection and other necessary maintenance work, etc. Thus, it is not possible to calculate the net working rate for JT-60. Therefore, a new term, “effective net working rate,” defined as the ratio of “effective operating time” to the “sum of the effective operating time and troubleshooting time” is used. Figure 3.5-2 shows the history of the effective net working rate and the number of shots per day in JT-60.

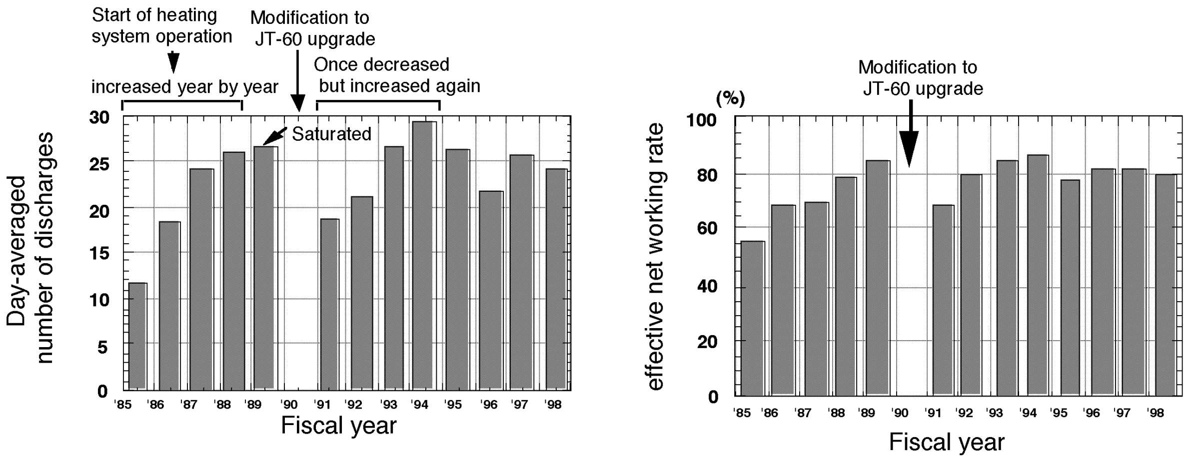

Fig. 3.5-2 History of averaged number of discharges per day and the effective net working rate of JT-60

The number of discharges tended to saturate at around 25 shots/day in Fiscal Years 1987-1989 after gaining operational experience during the starting year of the tokamak device in Fiscal Year1985 and in the use of the heating system in Fiscal Year 1986. The number of discharges dropped to about 20 for the two years following the JT-60 upgrade modification due to the new operation with noncircular plasmas. However, it increased to 25-30 shots/day after Fiscal Year 1993. The effective net working rate varied similarly and saturated at about 80%. Both the number of discharges and the effective net working rate dropped in Fiscal Year 1995. This was due to degradation caused by the aging of equipment, parts, electronic circuits, control elements, etc. Following this, suitable measures to correct the aging degradation helped recover the number of discharges to 25-30/day and the effective net working rate to 80%.

Conversion to capacity factor:

This effective net working rate can be interpreted as the net working rate in the period excluding the periodic inspection period. In the case of an LWR, the net working rate in the period that excludes the periodic inspection is about 80%. This value is applied to the fusion device. Therefore, the net working rate becomes a value obtained by multiplying the effective net working rate by this 80%. Consequently, it becomes about 64% of the value corresponding to the capacity factor for LWRs.

Since JT-60 is an experimental device, many changes take place including modifications and the addition of various systems, changes in the operation of the electric power supply, and changes in the control algorithm. These changes affect the effective net working rate. Unstable discharges leading to disruptions also reduce the effective net working rate. These caused a effective net working rate of 56-64%. To improve this value, an operation scenario for commercial fusion reactors must be established and the specified operation must be sufficiently analyzed. These efforts are urgently needed. Besides, it is necessary to increase the reliability of reactor hardware and software by improving the reliability of equipment and by adopting redundant instruments. It is also important to develop methods for minimizing repair time.

(3) Operators

From a viewpoint of manpower efficiency, the number of operators should be the minimum number that can deal with troubles. In the case of JT-60, one operating group consists of a JAERI staff of 14 and consigned industry members. Tasks of the consigned members include operation, start-up and shutdown of equipment at the local control room, and inspection.

Fusion reactors will operate in the steady-state mode. For a reactor in steady-state operation, start-up/shutdown operations as performed everyday in large tokamaks like JT-60 will not be required. Therefore, the tasks of consigned members can be remarkably reduced. Besides, since the operational conditions do not change during steady-state operation, the required number of operating staff members is also expected to decrease.

(4) Periodic inspection

The most influential item in determining the capacity factor is the duration of the periodic inspection. Therefore, to maintain a high capacity factor, it is necessary to make this period as short as possible, provided that the safety of the reactor and the reliability of the plant are assured. Periodic inspection will include the exchange of blanket, divertor cassettes, and heat resisting materials and will include the inspection of the tritium fuel circulation system, coils, refrigerators, control and monitor systems, auxiliary machinery, turbines, etc.

In JT-60, the periodic inspection of the tokamak device, the power supply system, the heating systems, the measurement system, the control system, and the auxiliary machinery is mainly performed in two months [3.5-3]. However, this list does not contain all the systems necessary for a power reactor. On the other hand, tasks to improve equipment are carried out in this period in JT-60. Therefore, the inspection period for JT-60 does not correspond to a periodic inspection period of a commercial fusion reactor.

Table 3.5-3 lists the large equipment in ITER along with the periodic maintenance this equipment will require.

Table 3.5-3 Large equipment that needs periodic maintenance in ITER (FDR)

Items	Frequency	Units	Total period
Divertor cassette	1 time / 1.5 year	60 cassettes	112 days
Shield blanket	-----	730 modules	349 days
Test Blanket	-----	several assemblies	-----

As a reference for examining the exchange of the breeder blanket of the fusion reactor, the study made for SSTR was referred to [3.5-5]. In SSTR, all breeder blankets are exchanged about every three years under an irradiation condition of 3 MW/m^2, and with the assumption that the allowable neutron fluence of ferritic steel is 100 dpa. At present, ferritic steel with a target allowable neutron fluence of 200 dpa is being developed. In this case, a half as many modules (200 modules in SSTR) will be exchanged on average every three years. According to a periodic exchange scenario using the vehicle type manipulator shown below, the period necessary for exchanging these blankets is about 28 days. This corresponds to 10 days/year if averaged over three years. The exchange period in the range of 60-70 days seems to be feasible if the period of preparation for, and the adjustment after, the blanket exchange is 30-40 days.

Estimate of days necessary for the periodic exchange of modular blankets in fusion reactors:

Shortening the period for periodic exchange of blankets is an effective approach to increase the capacity factor of the fusion reactor. At present, there are three systems for exchanging blankets, the “modular-unit system” adopted in ITER, the “banana-shape blanket-drawing system” adopted in SSTR, and the “integrated drawing system” studied in DREAM. The design of the “modular-unit system” and its R&D are the most advanced. Here, we considered the use of this system on SSTR and estimated the exchange time of the blanket on the basis of the evaluation method for ITER, taking into account expected technological progress. The vehicle type manipulator (Fig. 3.5-3) of ITER was used to perform remote maintenance. The pipe cutting and welding tool installed at the top of the vehicle type manipulator accesses the inside of the module from an access hole at the first wall side of the module and cuts and welds the cooling pipe using a YAG laser. Conditions for the blanket exchange work of ITER in the Final Design Report are as follows.

- The preparation time such as for baking, etc., before and after the exchange work is not considered as working time.
- Four vehicle type manipulators are installed (1 unit per quadrant).
- Sixteen units of pipe cutting/welding/inspection tools are installed (4 units per quadrant).
- A blanket weighs 4 tons.
- The number of bolts for fixing a blanket is 14 per four blankets, and 4 bolts are simultaneously mounted/dismounted.
- The working time for 4 days is 16 hours.
- The operating speed of the remote manipulation equipment is assumed to be the actual value obtained by a full-scale vehicle-type maintenance equipment test performed in the engineering R&D of ITER.
- Considering the uncertainty in the working time, 30% is added to the total exchange time.

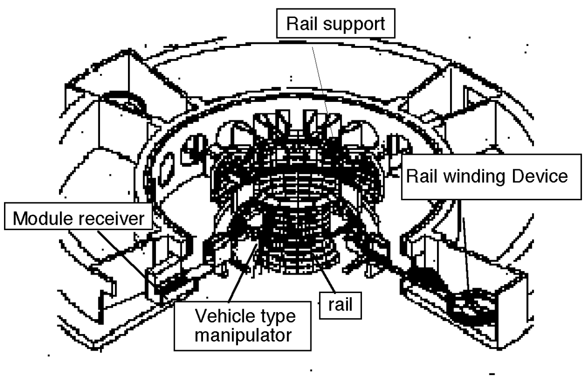

Fig. 3.5-3
Vehicle type manipulator for ITER

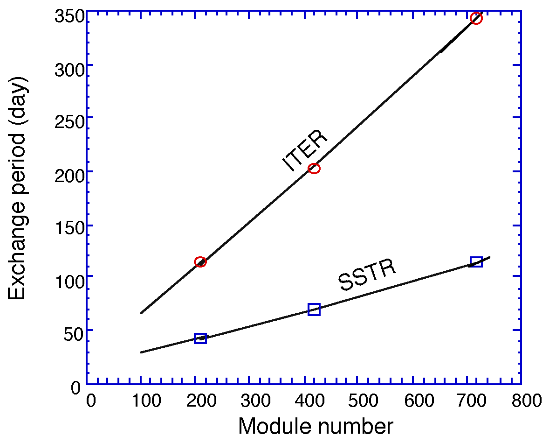

Fig. 3.5-4
Estimate of exchange period of blanket in ITER and DEMO

The blanket exchange time estimate is shown in Fig. 3.5-4. ITER has 730 blanket-modules. The exchange time for these modules is 349 days, including the 30% uncertainty contingency. This evaluation was based on the design of the vehicle-type manipulator for ITER. In SSTR, which is the next step after ITER, it is possible to expect technical advances in the design of remote maintenance equipment and the blankets. Therefore, the three following assumptions were added to the above conditions for the evaluation of blanket exchange time in SSTR.

- The number of fixing bolts is four per blanket. Mounting/dismounting of 4 bolts is performed in parallel using the tool installed at the top of the manipulator.
- The available work time per day is 24 hours.
- The operating speed of the remote manipulation equipment is improved by 30%.

The blanket exchange time estimate for SSTR is also shown in Fig. 3.5-4. From the ratio of the surface area of blankets for ITER and for SSTR, the number of blanket modules in SSTR is estimated to be about 400. The period for the exchange of all blanket modules is about 46 days, while the period for exchanging half the blanket modules, 200, is about 28 days. Further, this period may be shortened by future technological innovations. The "banana type" and the "integrated drawing type" are prospective candidates as future exchange systems, because the number of blanket structures to be exchanged is smaller. However, a single integrated unit becomes a large and heavy structure weighing in the order of 100-1,000 tons. Therefore, the structural strength as well as the carrier route to the hot cell must be thoroughly considered in the design of the building. Also, it is necessary to develop a support structure that can position and remove such a large and heavy structure to within an accuracy of several mm. These aspects are the challenges for future design and development.

(5) Unplanned shutdown (trouble occurrence rate)

Causes of unplanned shutdowns that reduce the capacity factor are troubles (machine troubles, operator errors, etc.). In an experimental device such as JT-60, troubles reduce the experimental efficiency. Reducing troubles is therefore an important subject also for JT-60. JT-60 data will now be referenced in discussing fusion reactor troubles. It is true that troubles are more complicated in fusion power plants. On the other hand, the fusion power plant will be operated under a fixed steady-state operating condition, which is expected to reduce the occurrence of troubles.

Figure 3.5-5 shows the variation in the number of troubles for every year in JT-60 from its completion [3.5-3]. In JT-60, a "trouble" is defined as failure or malfunction of equipment that interrupts the experiment discharge for one shot, namely, more than 15–20 minutes. This figure shows that troubles increased just after the start of operation of the heating system and just after the modification to the JT-60 upgrade. This observation suggests that initial malfunctions just after completion of construction or modification are a main cause of troubles. The recent increase in troubles can be explained that equipment and parts have been degraded through 10 years of use in JT-60. The number of troubles average about 2/day after the initial troubles were eliminated.

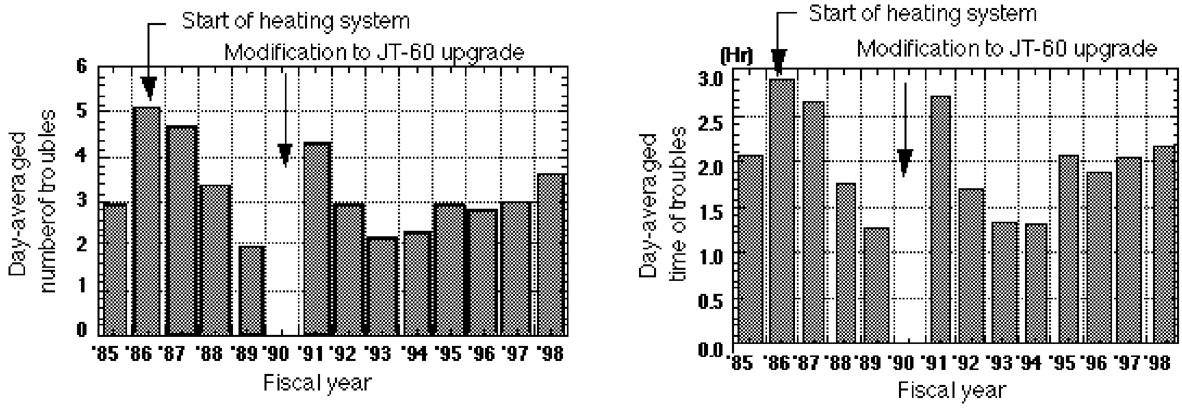

Fig. 3.5-5
Variation of frequency and time of daily troubles by year. (Both the trouble frequency and trouble time decreased year by year after the completion of JT-60, then became constant, and now are increasing.)

The most frequent time troubles occur is just before and just after a discharge according to an analysis of operational records. The equipment conditions change greatly at these times. Troubles tend to occur in the circuit breakers of power supplies (for example, an incomplete switching operation within a given time), as equipment failures during a preparatory action, in communication failures of the control system (for example, processing delays due to the change in network loads), etc.

The second most frequent time troubles occur is at plasma disruptions. A disruption induces electromagnetic disturbance to the power supplies through the coils, which may damage the circuit elements. A disruption may also saturate signal-processing systems for the electromagnetic field and may cause abnormal feedback control of the plasma. These could lead to other troubles, such as excessive torsional stress to coils, vacuum leaks, and loosening and falling of wall materials, which may need a relatively long time for repair.

The troubles before and after the discharge are expected to be uncommon in steady-state operation. Especially, redundant instruments can reduce troubles in the control system. However, avoidance techniques for disruptions must be established as a premise of practical use of the fusion reactor. Also, fusion reactors must be designed so that vacuum leaks and loosening of the wall materials, etc., will not occur even if a disruption does occur. Without the aforementioned problems, the reactor can be restarted rapidly after the disruption. In addition, it is necessary to reduce the probability of troubles due to the plasma itself by understanding the plasma behavior.

In a "sophisticated fusion plant" with established techniques of steady-state operation and disruption avoidance, a low trouble frequency close to the value of the present LWRs is expected to be a possibility. This is supported by the fact that a trouble seldom occurs in discharges for machine conditioning where the operation scenario is established.

The overall reactor reliability should be ensured so that power generation interruption will not occur either due to plasma problems or other causes. This requires improvement of the reliability of individual pieces of equipment and the use of redundant (and diversified) instruments, etc., as has been done in LWRs. If redundant instruments are widely used, plant reliability will be improved, and the probability of unplanned shutdowns will decrease. However, these measures increase the cost. This is similar to the discussion on the remote maintenance. This subject cannot be discussed from the technical aspects only, and further examination is necessary.

3.5.3 Assessment of the feasibility of operation and maintenance of fusion reactor

The substance of the above discussion is compared, in Table 3.5-4, with the performance record of LWRs described at the beginning of this section. The assessment of each item is summarized below.

(1) Since the duration of start-up tests for an LWR is one year, the start-up test of a fusion reactor will be shorter than that of ITER, 3.5 years. Commissioning without plasma in the fusion reactor is expected to be considerably shorter than one year in ITER. Commissioning with plasma is estimated to be 2.5 years in ITER, considering the fact that uncertainty remains in the plasma operation scenario and that plasma experiment is an important mission of ITER. Establishing an operation scenario may further shorten the period until confirmation of 100% output. This depends on the future progress in operation studies.

(2) JT-60 has 14 operators per shift; considerably more than that of a present LWR, 6 persons per shift. Establishment of standard operation will minimize the number of operators in a fusion reactor.

(3) As for the periodic inspection, the exchange of modular-unit blankets is estimated to take 30 days every 3 years. It is important to conduct research and development to achieve this goal.

(4) In the present large tokamaks such as JT-60, the trouble occurrence rate is fairly high. However, most of these troubles occur before and after a discharge and at disruptions. Therefore, there is a possibility of remarkably decreasing the frequency of troubles if the following conditions are satisfied: an operation scenario is established for the steady-state operation, a disruption avoidance technique is developed, redundant instruments are adopted, etc.

(5) The capacity factor will be able to reach the LWR level if the following are achieved; the duration of periodic inspection can be shortened to the LWR level, plasma instabilities such as disruptions can be completely avoided, and steady-state operation is accomplished. Adoption of redundant instruments will sufficiently reduce machine troubles.

Especially, it is important to promote studies to decrease uncertainty in plasma operation, development of efficient remote maintenance systems, and improvement of the reliability of various types of equipment. The progress in these R&D tasks will significantly improve the operation and maintenance feasibility of commercial fusion reactors.

Table 3.5-4 A feasibility outlook on fusion reactor operation and maintenance from the viewpoint of present fusion devices

Items	Present status of fusion devices	prospect	Performance of LWR
Starting test	< 3.5 years (until initial experiment) (ITER)	2	< one year (100% output)
Operators	< 14 JAERI's staff + consigned staff (JT-60)	1	6 or 7 persons/unit
Periodic inspection	~70 days (ITER/SSTR)	2	50-70 days /unit year
Trouble occurrence rate	~2.5 events/day (JT-60)	2	< 0.5-4.0 events/unit year
Net working rate	> about 64% (JT-60)	2	> 60-80%

1: Possibility of realization is large.

2: Possibility is increased if operation scenario of the steady-state operation is established, if disruption avoidance techniques are developed, and if redundant systems are adopted.

References

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[3.5-4] ITER Final Design Report (Draft), IAEA, 1998.

[3.5-5] Fusion Reactor System Lab. "Concept Study of the SSTR," JAERI-M 91-081 1991.

[3.5-6] Atomic Energy Drawing Collection, (Found.) Atomic Energy Culture Promotion Foundation (1998).

[3.5-7] ITER Design Description Document (WBS2.3 Remote Handling), IAEA, 1998

3.6Technical issues from the viewpoint of industry

In this section, two technical issues from the viewpoint of industry are discussed with reference to nuclear power plant construction experience. “Industry” means companies that construct and supply facilities or are plant suppliers. Issues considered are:

- Issues related to “commercializing” fusion reactor technology, and
- Issues related to “commercializing” a commercial fusion reactor after participating in the construction of the demonstration fusion reactor developed as a national project.

(1) Issues of fusion reactor with reference to the construction plan and work schedule of light water fission reactors [3.6-1 - 3.6-5]

For fusion reactors, presently there exist neither laws nor regulations such as the Law for the Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors (hereinafter referred to as “Reactor Regulation Law”). There are neither procedures nor application forms to obtain licenses for construction and operation. For example, for construction of a present nuclear reactor facility, it takes 1.5 years for the safety review by the regulatory body, after the submission of the application document for the reactor installation license, which take more than one year to prepare [3.6-5]. If it would take five years in total to receive a construction permit, including other approvals, it is necessary to start the preparation for the licensing procedures as early as possible. It is hoped that the site selection and the development of laws and safety review guidelines or their alternatives will be promoted. In addition, the Electricity Utilities Industry Law is applied in the case where electric power is generated in the nuclear reactor facilities. Thus, discussions are needed as to what laws and regulations should be applied to the experimental fusion reactor. In this section, the construction plan and work schedule of a light water fission reactor (LWR) are introduced as a reference for the case of a fusion reactor.

The building construction work and equipment fabrication work should collaborate with appropriate management and with the full understanding of the following responsibilities for the nuclear power plant construction:

a) Building construction and fabrication of equipment and systems are to proceed in parallel for an extended period of time.
b) A huge quantity of construction materials and components are involved and the construction period is very long.
c) The quality assurance (QA) requirements are extremely detailed. In addition, traceable QA records are required.

From the start of construction, 4-5 years are necessary to progress to the start of operations. To proceed with this kind of work effectively, detailed discussions in the planning stage prior to the start of construction, early finalization of the plant design, and careful process management are essential.

The planning stage takes a long time prior to the start of construction at a new site. It takes more than five years for the planning, and the details are substantially different from one plant to another. The power company (utility company) plays a major role in the planning stage. The power company orders equipment and systems from manufacturers after the regulatory safety review is finished and the reactor installation license is granted. The component manufacturers (plant suppliers) then have to design and construct the best plant possible under the given conditions. However, in many cases the design work is begun in cooperation with the component manufacturers in advance during the plant planning stage. The planning should include important milestones including material procurement, start of manufacturing, construction permit, and delivery. The following is an overview of the planning and construction in which component manufacturers are involved.

1) Plans of nuclear power plants

(i) Licensing:

For the planning, construction, and operation of a nuclear power plant, various licensing procedures have to be followed on the basis of relevant laws and regulations, in particular, the Reactor Regulation Law and the Electricity Utilities Industry Law.

After determining the site of the nuclear power plant, following the facility planning and the environmental review, the power company has to apply for the reactor installation license. The application has to be submitted to the Minister of International Trade and Industry based on the Reactor Regulation Law. The regulatory review for this licensing is the so-called Safety Review.

Examples of the structure of the application document for a reactor installation license are shown in Table 3.6-1 and 3.6-2. The volume of the Appendices is much larger than the main document. In particular, Appendices 8–10 must contain so many details that the power company has to prepare detailed designs well before actually placing orders with the component manufacturers.

After obtaining the reactor installation license, the power company has to submit a detailed construction plan to the Minister of International Trade and Industry and obtain permission for the construction plan (construction permit) based on the Electricity Utilities Industry Law. This application for construction permit can be divided into major construction processes because it is impossible to complete final detailed design for all construction tasks at the beginning of the construction phase.

New type reactors under development are not under the control of the Ministry of International Trade and Industry but instead are under the control of the Science and Technology Agency because these reactors are not for commercial operation. The process to obtain permission for the construction work of these reactors is the application for the "Permission for the Design and Construction Methods" based not on the Electricity Utilities Industry Law but on the Reactor Regulation Law. However, the Electricity Utilities Industry Law is also applied for reactors that generate electricity.

(ii) Basic Plan, Design, etc.:

The basic guidelines for the construction of nuclear power plant are determined based on the facility plan. Subsequently, the power generation capacity, the type of reactor and containment, the basic design of the plant facility, the layout plan of the site, the layout plan of components, etc., are determined. Then the specifications of the major facilities and equipment are fixed. The design of buildings and structures should then proceed in parallel with the design of equipment and components. However, the outline design had to be fixed before submission of the application for the reactor installation license.

(iii) Quality Assurance:

The power company is required to establish Quality Assurance (QA) Program, which covers activities from the design stage to the start of operations. The power company has to manage integrated control of the design, manufacturing, and commissioning of the components as well as licensing-related work for the nuclear power plant construction. Component manufacturers should submit their QA program plan to the power company and perform their QA activities for products of their own making and those procured. The power company oversees the status of their QA activities, checks records, and carries out on-site inspections.

(iv) Determination of procurements and orders:

In general, procurements are divided as follows and ordered:

a) Civil engineering: for each task of site grading, loading facility, foundation of building and structure
b) Construction: for each major building, auxiliary building
c) Equipment and components: for each major equipment and auxiliary equipment

The procurement of the major equipment and components should conclude with the start of the main construction work.

2) Construction process of nuclear power plant

The construction processes are roughly divided into those relating to reactor building, rad-waste process building (BWR), control building (BWR) or reactor auxiliary building (PWR), and turbine building. The construction period is divided into following four periods:

1st period: Civil engineering
2nd period: Building construction
3rd period: Installation of equipment and components
4th period: Commissioning

Construction work proceeds in the above sequence in cooperation with component manufacturing processes, governmental inspections, and so on. In planning the construction process, the power company develops a basic process and an integrated process chart that considers many conditions, such as the state of the plant, weather conditions, etc. Each component manufacturer develops detailed process charts according to the power company's process chart. Throughout the construction work, the processes are coordinated by cooperation among the companies and the power company.

(i) Civil engineering and building construction work:

The soil preparation is conducted at the site of nuclear power plant based on the overall site layout plan. According to Japanese practice, seawater is used as cooling water for the condensers. Thus, nuclear power plant sites are located along the seacoast. The excess graded soil is put to good use to reclaim land from the sea and enlarge the site.

The reactor building (containment) excavation starts immediately after the completion of all the procedures relating to construction are approved. At the completion of excavation work, the bedrock inspection is carried out as the first pre-service inspection of the plant.

The building construction work constitutes a major part of the nuclear power plant construction process. In order to shorten the construction period, it is necessary to adopt temporary housing, shortened processing methods, and coordinate all mechanical and electric work.

(ii) Mechanical and electric work:

During the construction work on the reactor containment, which is the critical path of nuclear power plant construction, all work needs to be coordinated well because large-scale work involving equipment installation and the building construction are proceeding in parallel. When the system integration nears completion and after completion of the installation of equipment and piping, flushing of the systems is conducted. The flushing uses air or water for the removal of unwanted objects from the system.

The objectives of flushing include:

a) Reduction of unwanted material that may become activated (radioactive) during plant operation
b) Prevention of the failure of the system due to unwanted objects
c) Removal of harmful elements from the system

Large quantities of filtered and purified water are used in flushing. Prior planning needs to be made to ensure the availability of raw water, and purifier equipment and storage tanks having sufficient capacity. A processing system for wastewater needs to be installed and operated to meet emission standards.

After the flushing, pressure tests and leak tests are carried out using pressurized water or air.

(iii) Tests and inspections:

Major tests and inspections are as follows:

a) Inspections in-factory: (as necessary) pre-service inspection by government and power company's inspection
b) Installation inspections: welding inspection based on the Electricity Utilities Industry Law including pre-service inspection
c) Single component tests: adjustment operation of each component after installation and before the system functional test
d) System functional tests: including pre-service inspections based on the Electricity Utilities Industry Law
e) Integrated inspections: conducted for each major step in collaboration of the power company and manufacturers
f) Start-up test (Commissioning)

For new type reactors under development, these tests and inspections are based on the Reactor Regulation Law. For reactors with electric power generation, the Electricity Utilities Industry Law is also applied.

(2) Organization of nuclear power plant construction [3.6-10]

In this section, a few examples of construction organization are presented as a reference for the study of fusion reactor construction. Each existing nuclear power plant in Japan was constructed by one of the domestic power companies. In the case where the construction project is under the control of an international cooperative organization, the methods of coordination and management are a matter of concern. Discussions should be started on this issue as soon as possible.

The following are examples of construction organization for the existing nuclear power plants as described in Appendix 3.6-1 which is attached at the last of this section:

(i) General:

The reactor facility construction involves a large number of contractors; not only component manufacturers (plant suppliers) but also other contractors such as for the building construction, civil engineering, and harbor construction. Each contractor has sub-contractors and allocates its tasks to them. For instance, a component manufacturer distributes some part of its tasks to related companies, sub-venders, and building companies.

(ii) Kansai Electric Power Company (KEPCO) / Ohi-4 (PWR):

The plant supplier is Mitsubishi Heavy Industries (MHI). Many building companies are contracted by KEPCO. In many cases of BWRs, Toshiba or Hitachi is contracted as a main plant supplier.

(iii) Japan Atomic Power Company (JAPCO) / Tsuruga-1 (BWR):

Tsuruga-1 is Japan's first commercial BWR power station. The plant supplier was General Electric (GE). Toshiba and Hitachi joined the construction as GE's sub-contractors to construct part of the equipment.

(iv) Chubu Electric Power Company (CEPCO) / Hamaoka-4 (BWR):

The contract for the construction was made with Hitachi and Toshiba separately as plant suppliers.

(v) Tokyo Electric Power Company (TEPCO) / Kashiwazaki-Kariwa-6 (ABWR):

The contract was made as a joint venture of Hitachi, Toshiba (representative), and GE as a plant supplier.

(vi) Power Reactor and Nuclear Fuel Development Corporation (PNC) / Monju (FBR):

The contracts were made with Toshiba, Hitachi, Fuji Heavy Industries, and MHI in parallel as plant suppliers. FBEC, co-sponsored by the four companies, coordinated technical issues among the companies. The contracts for the building construction were made with three joint ventures.

(3) Mastering of technology [3.6-8]

By looking back upon the maturing progress of LWR technology in Japan, the best ways to master technologies for the commercial application of fusion reactors may be seen. Fusion reactor development should consider events that occurred in the early stages of LWR development and construction, since similar events are expected in fusion reactor technology. There are many other fields in this new technology, but the LWR experiences still offer the best references. The following issues, summarized from the study of the development of LWR technology, should be considered:

a) Limitation of technology imported from foreign countries
b) Issues that emerge in construction and operation
c) Necessity of the accumulation of experience in construction and manufacturing (including manufacturing R&D)

One of the inherent aspects of "technology development" is the fact that the technology developed without the participation of engineers tends to become difficult to apply for practical use. Often difficult problems unexpectedly come up one after another during technology development. Problem solving constitutes a very important part of technological development, since an engineer can master better and more practical technology through this process. In this way, a new technology, although carefully developed, is nurtured for practical use. The acquired knowledge enables future improvements and advancements, and this process offers good opportunities to train engineers.

Importing technology from foreign countries is associated with the following two problems:

- Because engineers cannot experience the process of resolving unexpected problems, they might face difficulties in repair and improvement of the imported technology.
- Because the quality of technology (integral technology including manufacturing know-how, material resources, inspection criteria, etc.) is different between the importing and exporting countries, such technol-

ogy, in general, cannot be directly applied for practical use. To put such technology to practical use, adaptations and modifications to importing country's technological quality may be necessary, and in the worst case, the R&D must be redone from the beginning.

Fusion R&D still involves much basic research. It will take much work and a long time to produce a fusion reactor that is practical for commercial use. Throughout this development stage, fusion R&D will remain a national project. However, technology can be accumulated and transmitted to the next generation by involving component manufacturers in the manufacturing R&D processes. In particular, this aspect needs serious consideration for the key technologies and components of fusion reactors. For construction success, it is important to have engineers in the fields of plant systems engineering and project management coordinate manufacturing and construction activities. The participation of power companies may become necessary, during the process toward the commercial fusion reactor, to have them acquire experience in the construction and operation of fusion reactors. Throughout the development stage, the government, power companies, and component manufacturers (plant suppliers) should cooperate with each other to make improvements. This cooperation is essential to accelerate the process of making the fusion reactor available for commercial use.

It is thus very important that human resources and technologies in fusion related industries be maintained and enhanced. If adequate opportunities of manufacturing are not afforded to them, their technological expertise will soon decay. In the recent serious economic setback, component manufacturers were forced to shutdown "presently unprofitable" businesses. To maintain and enhance important technologies, it is essential that the companies be continuously challenged by new demands.

Technological development and enhancement in the case of LWR is presented in the following subsections. [3.6-1, 3.6-6, 3.6-7]

1) The beginning of LWR development

(i) LWR development in Japan

The first commercial reactors in Japan were built by importing established U.S. technology. In this period, participation in nuclear power plant construction as subcontractors of GE or Westinghouse (WH) enabled domestic manufacturers to become involved in the construction of nuclear power plants. Although domestic companies had experienced construction of research reactors and a power demonstration reactor, their technical level was much lower than U.S. companies and they had to study the basics, including quality assurance. The domestic companies entered into technology cooperation agreements with U.S. companies to master nuclear systems technology. The following is a brief summary of the development process for LWR (mainly BWR) technology in Japan.

In addition to the technology level problems in Japan, many problems existed in GE's quality assurance and in management of deliveries of components, etc. when BWRs were introduced into Japan. For instance, Japanese companies sometimes had to repair welding defects. There were cases of delivery delays, and the need to maintain the total construction schedule forced the Japanese companies to expedite construction. Japanese companies supported EBASCO, which was in charge of plant engineering for the first commercial BWR in Japan (Appendix 3.6-1), when the design work was delayed. They assisted with the design work to maintain the construction schedule. Consequently, Japanese companies studied technology through this effort.

GE supplied the major reactor components of the imported BWR plants. Domestic production of these components required Japanese companies to study GE's know-how and to repeat trial manufacturing several times to demonstrate the reliability of their products. Through their efforts, products from Japanese companies were finally accepted for use in domestic commercial plants.

(ii) Issues raised through operation experience

Although the U.S.-developed BWRs were well designed, Japanese companies faced inconveniences partly because they did not have nuclear power plant operational experience and partly because the following technical problems existed.

a) Issues on BWR that had to be resolved

- Waste processing system capacity was too small and the quantity of rad-waste was large.

- Radiation dose of workers was high, which made their work difficult and inefficient.
- Fuels had high failure rates.

In addition, many troubles arose, one after another, around 1974.

b) Initial troubles and their solution

- Stress corrosion cracking of stainless steel pipes in the primary system, etc. (BWR)
- Leaks in steam generator U-tubes, etc. (PWR)

These troubles reduced the capacity factor of these plants to less than 50%.

To solve these troubles, utilities and suppliers cooperated in investigating the causes of failures and developed countermeasures, including preventative measures, to minimize these frequent troubles.

2) Development of domestic technologies

Although the above problems were hard trials for the Japanese companies working on BWRs, the problems gave them an opportunity to develop their own technologies.

(i) The hard-earned lesson from the experience in importing technology led to the recognition of the importance of basic research and the establishment of organizations for R&D. (Recognition of limits on imported technologies)

- Research cooperation with GE (contract for new technology development on equal terms)
- Start of electric power cooperative research system: Support for suppliers' R&D and full utilization of the results
- Establishment of the Nuclear Power Engineering Corporation by the government

Until then, the Japanese government maintained an idea that R&D advancements for commercial reactors should be the responsibility of industry. The government had supported basic safety research for regulatory purposes mainly conducted in JAERI, but the government did not support R&D on LWR technology. However, in consideration of the above-mentioned troubles and their social influence, the government began to support R&D on LWR technological improvement. In fact, the government funded a special budget for the development of electric power sources in 1974 and established a budget for large-scale engineering demonstration tests on safety and reliability of LWRs.

(ii) Improvement and standardization program

There were similar issues in PWRs. The government, power companies, and plant suppliers reached a common understanding that they had to develop more reliable LWRs. Consequently, the "Improvement and Standardization Program of LWRs" was started.

The improvement in reliability and availability, and the reduction of the radiation dose to workers were the aim of the first period (FY1975-77) and the second period (FY1978-80) programs. During these periods, more than fifty proposals for improvements were submitted.

The results of the first and second programs were applied to new nuclear power plants being built. As the programs proceeded, conservatism in the design received attention. The number and volume of nuclear power plant components and buildings continued to increase. In addition, the influence of the oil-shock also made the cost of nuclear power plant construction increase sharply. Accordingly, in the third period of the program (FY1981-85), the government supported the development of a new type LWR and set up a Special Committee for Advanced Nuclear Reactor. The conventional LWRs were also improved and standardized.

The above-mentioned programs resulted in the Advanced BWRs, Kashiwazaki-Kariwa Unit 6 and 7, which started commercial operation in 1996 and 1997.

The accomplishments achieved are summarized as follows:

- Reactor fuel improvements reduced fuel failure rates to approximately zero.
- Radiation dose of workers decreased by more than one-order of magnitude. The number of periodic inspection processes decreased and the outage period shortened.
- Radioactive releases to the environment were significantly reduced (to 5% of natural radiation levels for gases and 0% for liquids).

• Plant operation was facilitated and controllability was improved.
• The construction period was shortened (shorter than 50 months).
• The capacity factor was improved, now higher than 80%.

3) Development of ABWR

(i) AEG (Germany) and AA (Sweden) independently developed the Recirculation Internal Pump (RIP) plant.

(ii) In Japan, the RIP system study (Toshiba) as a trust research on electric power and the study on the fine motion control rod drive (FMCRD) (cooperative study by Hitachi and Toshiba) began in 1977.

(iii) International cooperation for development of new type reactor: In the summer 1977, Toshiba, Hitachi, and GE agreed on the cooperation of a new type reactor development and established an advanced engineering team (AET). Then ASEA ATOM and Ansaldo (Italy) joined AET.

(4) Effect of mass production of plants [3.6-9]

In Japan, LWRs (PWRs and BWRs) are constructed at a rate of approximately one unit every other year. This pace is not high. Because the type and/or power generation capacity are different from one unit to another, mass production of LWRs is considered impractical.

In contrast, French nuclear power plants generate 80% of the total electric power for the country, and the unit cost of power generation by nuclear power is regarded as less expensive than other means of power generation. France benefits from standardization and serial production of plants. The 900 MW and 1300 MW class nuclear power plant types were standardized. Ten nuclear power plants of each type were constructed for 10 years, which meant one unit of each reactor type was constructed every year.

The situation of the fusion reactor is different from that of LWRs because the detailed design of the fusion reactor is still under development and thus is not fixed. The design of a fusion reactor will be applied for only one reactor plant. As the facility and equipment for manufacturing will be used for only one plant, the construction costs will be higher than when a standardized design is adopted later. The cost effectiveness of fusion power cannot be expected until fusion reactors are put to commercial use and their performance record is established. Improvement or standardization should be made after operational experience is acquired.

(5) Sharing of responsibilities between the ordering party and the supplier companies [3.6-11]

Formerly, the practice in Japan, except in the beginning, was that a single supplier company acted as a prime contractor and played a central role in the construction of every LWR plant. On the other hand, new type reactors under development have been constructed under multiple company cooperative agreements. The construction of the JT-60 took the same approach. Examples of this approach are shown in Appendix 3.6-1. It is expected that this approach will continue in Japan.

The construction contract is commonly categorized under a performance specification contract and a structure specification contract. The difference resides in the sharing of responsibility of plant performance between the ordering party and the contractor. In the most cases, a clear separation of responsibility cannot be made. Consequently, both parties share the responsibility, depending on each case, to ultimately produce the best performing plant. A comparison of the two kinds of contracts follows.

Performance specification contract:

The ordering party develops the basic design and performance specification documents in which the basic specifications and performance of each component are described. The contractor assumes the task of detailed design, including the design for manufacturing, trial manufacturing, manufacturing, single component tests, modifications if necessary, confirmation of performance, and delivery of final components. The contractor has the responsibility for the performance of components. This kind of contract is adopted for a subsystem of a significant scale or for a system.

Structure specification contract:

The ordering party assumes the responsibility for the basic design, detail design, design for manufacturing, and trial manufacturing of components. Then, after verifying compliance with the requirements, the ordering party develops structure specification documents, which include drawings for manufacture, specifications of materials and structures of components, manufacturing methodologies, etc., and awards a contract. The ordering party conducts single component tests. The ordering party is responsible for the component performance. In general, this contract is adopted for individual components.

A typical example of task allocation for each type of contract is shown in Fig. 3.6-1. In both cases, the design must be ready prior to the development of specification documents. In some previous cases involving a performance specification contract, engineers from contractors participated in the design activities under the responsibility of the ordering party. Structure specification contracts are rather common in the western countries, and in many previous cases research institutes conducted various design activities as the ordering party. Both contract types are compared in Table 3.6-3.

The choice of the type of contract generally depends not only on the levels and characteristics of technology but also on the capability and practices of the ordering party and of industry.

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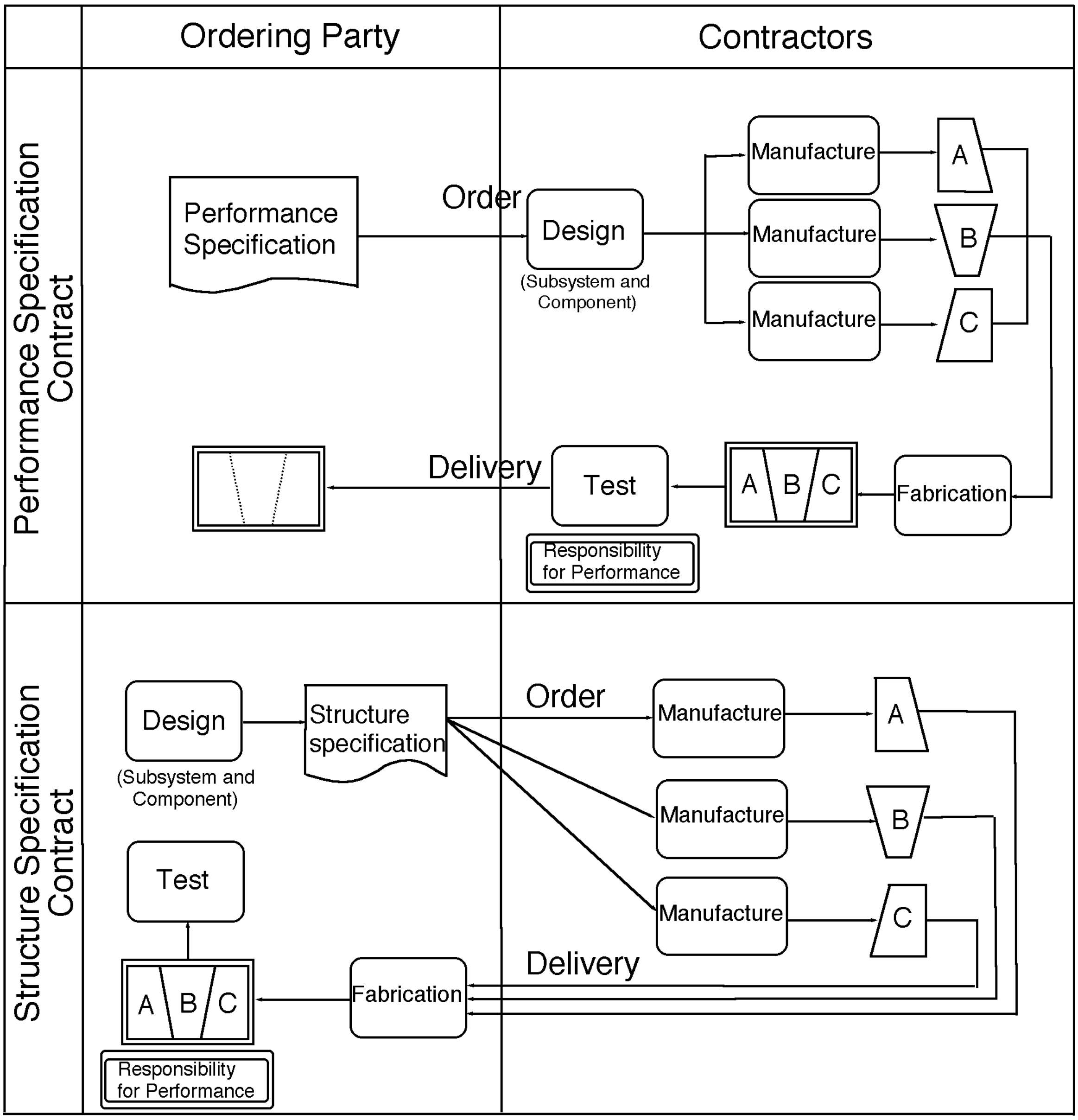

Fig. 3.6-1
Typical Example of task allocation for the ordering party and contractors in a performance specification contract and a structure specification contract

Table 3.6-1 Contents of an application document for reactor installation license (1)

[Main document]

1. Name of the applicant
2. Purpose of the usage
3. Type of the reactor
4. Name and address of the factory or establishment where the reactor will be located
5. Locations, structures, and equipment of the reactor and auxiliary systems
6. Construction plan of the reactor facility
7. Nuclear fuel materials to be used in the reactor and the quantity of fuel for annual use
8. Methods of spent fuel treatment

[Appendix 1]
Description of the purpose of reactor usage

[Appendix 2]
Description of the thermal power output of the reactor

[Appendix 3]
Description of the reactor construction cost and the financial plan

[Appendix 4]
Description of the procurement plan of the nuclear fuel materials for reactor operation

[Appendix 5]
Description of the engineering abilities for installation and operation of the reactor facility

[Appendix 6]
Description of the meteorological, geological, hydrological, seismic, social environmental, and other conditions with respect to the site

[Appendix 7]
Maps of the areas within 20 km and 5 km from the site

[Appendix 8]
Description of the safety design of the reactor facility

[Appendix 9]
Description of the radiation protection management against nuclear fuel materials and their contamination, and of the treatment of radioactive wastes

[Appendix 10]
Description of the categories, consequences, etc., of reactor accidents postulated to result from an operator error, a failure of machine or equipment, earthquake, fire, etc.

Table 3.6-2 Contents of an application document for reactor installation license (2)
(Appendix 8, 9, and 10)

[Appendix 8]

1. Safety design
   1.1 Safety design policy
   1.2 Conformity with regulatory guidelines on safety design
   1.3 Seismic design
2. Plant layout
3. Reactor and reactor core
   3.1 Overview
   3.2 Mechanical design
      3.2.1 Fuels (including mechanical design)
   3.3 Nuclear design
   3.4 Thermal hydraulic design
   3.5 Reactor dynamics

4. – 12. Description of systems

13. Operation and maintenance

[Appendix 9]

1. Basic policy of radiation protection
2. Radiation management in the plant
3. Radiation monitoring around the site and in the vicinity
4. Treatment of radioactive wastes
5. Results of radiation dose evaluation

[Appendix 10]

1. Introduction
2. Anticipated operational occurrences
3. Accidents
4. Major accidents and hypothetical accidents

Table 3.6-3 Comparison between performance specification contract and structure specification contract

	Items	(A) Performance Specification Contract	(B) Structure Specification Contract
1.	Required ability of contractors	Ability to develop detailed and manufacturing design, to manufacture components satisfying the specification, and to take technical risks.	Ability to manufacture components as specified in the specification and drawings. Contractors' risks are smaller than in the case of (A).
2.	Required ability of ordering party	Ability to manage a contractor that can satisfy the requirement as 1.(A).	Ability to develop manufacturing design, prototypes, drawings for manufacturing, and structure specification documents as well as to direct the manufacturing.
3.	The number of candidates of contractors	For established technology: many For technology yet to be established: limited	More than in the case of (A), even in the case of technology yet to be established.
4.	Cooperation between subcontractors to manufacture subsystems	The product quality of components tends to vary among contractors: quality should be specified in the specification document. Common standards for subsystem allocation and compatibility among the subsystem should be specified.	Although the number of problems would be smaller than in the case of (A), the ordering party's work load would be more than in the case of (A).
5.	Cost	The contractor has to incur the additional costs of detailed/manufacturing design, trial manufacturing, single component testing and modification. The cost to the contractor will be higher than in (B). In the case of having multiple contractors manufacture similar subsystems, the cost of manufacturing design and trial manufacturing will multiply.	The ordering party has to assume the additional cost of detailed/manufacturing design, trial manufacturing, single component testing and modification. The cost to the ordering party will be higher than in (A). For the case of having multiple contractors manufacture similar subsystems, the cost of manufacturing design and trial manufacturing will be lower than in (A).
6.	Manufacturing period	Manufacturing can be proceeded effectively from the detailed design to single component testing and modification. If one company manufactures a subsystem, the manufacturing proceeds more effectively because all the component design can be integrated to satisfy the overall performance specification. The manufacturing period is shorter than in the case of (B). However, if multiple companies manufacture the same components in the subsystem, the manufacturing period would be different among contractors and the total period would not be so short.	The manufacturing period tends to take longer than in the case of (A) because the processes of manufacturing design, trial manufacturing, manufacturing, single component testing, and modification are shared between the ordering party and contractors. If a number of components are included in the subsystem, dividing the order among multiple companies can shorten the total manufacturing period.
7.	Contract type and achievement of specified performance	For a contract including technology development, the contractor's risk would be bigger if the contract is a fixed-price contract. However, if a contract is a work order contract, the company's risk would not be so big (*2). The ordering party can obtain reliable components or subsystems, which satisfy the specified performance in a relatively short period.	Even for the contract including technology development, contractor's risk would be small. On the other hand, the manufacturing period would be longer than in the case of (A).
8.	Training of engineers	Engineers are trained to develop technology.	Next generation engineers cannot be trained because only the manufacturing department in the contractors will be involved (*1).

(1*) It is necessary that industry engineers participate in the design and trials after the basic design.
(2*) Fixed-price contract: The specification and price are fixed when the contract agreement is concluded. Price-readjusting contract: The specification and price are temporarily determined when the contract is concluded and the final price is determined based on the actual expenses at the completion of work. In contracts for a complex system, both contract types may coexist in the subsystems.

Appendix 3.1-6 Examples of construction organization for a nuclear power plant

1) General

- Power Company
  - Equipment (Plant) Suppliers — Related Companies
    - Subvenders
    - Nuclear Fuel Makers
    - Building Construction Companies (Basic Designs of Buildings)
  - Building Construction Companies
  - Civil Engineering Companies
  - Harbor Construction Companies

2) Ohi Unit 4

- KEPCO
  - MHI
    - Reactor Systems
    - Reactor Pressure Vessel
    - Core Structures
    - Fuel (MNF/NFZ)
    - Steam System
    - Turbine
  - Ohbayashi, Takenaka, Taisei, Kumagai, Kajima, Kokudo, Maeda, Hazama

3) Tsuruga Unit 1

- JAPRO
  - GE
    - AE(EBASCO)
    - Reactor Systems
    - Reactor Pressure Vessel (B&W / Hitachi)
    - Core Structures (Hitachi)
    - Fuel (GE, JNF, NFI)
    - Steam System
    - Turbine (Toshiba)
  - Takenaka, Kumagai

4) Hamaoka Unit 4

- CEPCO
  - Toshiba
    - Reactor Systems
    - Reactor Pressure Vessel (IHI)
    - Core Structures
  - Hitachi
    - Steam System
    - Turbine (Toshiba)
  - Kajima, Shiraishi, Kumagai, Takenaka, Shimizu, Hazama, Sato, Maeda

5) Kashiwazaki-Kariwa Unit 6

- TEPCO
  - Joint Venture
    - Toshiba
      - Representative
      - Reactor Systems
      - Reactor Pressure Vessel (IHI)
    - GE
      - Core Structures (Toshiba)
      - Fuel
      - Turbine / Generator
    - Hitachi
      - Steam System
  - Kajima, Shimizu, Takenaka, Kumagai, Hazama

6) Monju

- PNC
  - Toshiba
    - Reactor -Top Structure, Fuel Exchanger
    - Secondary Cooling System, Auxiliary Cooling System
    - Instrumentation in the Reactor Vessel
    - Steam-Water System & Turbine System
  - Hitachi
    - Primary Cooling System
    - Heater
    - Central Computer
  - Fuji
    - Reactor Systems
    - Waste Treatment System
  - MHI
    - Reactor Pressure Vessel
    - Reactor Containment Facility
    - Evaporator
    - Plant Control System, Safety System
  - FBEC
    - Technical Coordination
  - Fuel
  - (Ohbayashi • Taisei • Kajima JV), Civil Engineering JV

7) JT-60 (for Reference)

- JAERI
  - Hitachi
    - Total Coordination
  - Hitachi
    - Main Apparatus
    - Total System Control
    - NBI Ventiration System
    - Other
  - Toshiba
    - Power Supply for Poloidal Coils
    - 275kV Central Transformer System
    - NBI Power Supply System
    - Other
  - Sumitomo
    - Reserve Ionization Gas Injection System
    - High Frequency Wave Heater
  - Mitsubishi
    - Power Supply for Poloidal Coils
    - Other
  - Fuji
    - Power Network Panel for Control
    - Other

3.7 Issues to ensure competitiveness in the market

Unless fusion reactors prove to be a more attractive energy source than the existing energy sources, they will not become competitive in the market. Although the market is generally governed by the so-called market principle (in search of less expensive products), environmental aspects, such as global warming, have also become important measures of market competitiveness. The issue of high-level radioactive waste disposal may affect the social acceptance of fission nuclear plants and hence the market competitiveness. So, market competitiveness is not determined solely by the cost. However, if restrictions are imposed on emissions of carbon dioxide (CO_2) and on radioactive wastes, as may happen in the near future, CO_2 sequestration plants and radioactive waste disposal facilities would increase the cost of electricity, which will be reflected in the market competitiveness.

This section summarizes key issues to reduce the cost of electricity (COE) produced by a fusion power plant to ensure competitiveness in the market.

3.7.1 Key factors to determine the COE

The following three factors may significantly affect the COE of a fusion power plant;

- Capital costs and operational costs of the power plant
- Capacity factor (availability) of the power plant
- Safety of the power plant (location of plant, decommissioning costs)

The capital costs, operational costs, and availability directly influence the COE. A compact power plant is necessary to reduce the capital costs. Issues for establishing a compact fusion power plant will be discussed in the next section. To increase the availability of a fusion power plant, efficient replacement of blanket modules is essential. A capacity factor of 70-80% can be expected in a fusion power plant from the extrapolation of the present remote handling technology.

Safety in fusion power plants, which is essential to gain social acceptance of fusion power, plays an important role in reducing the COE. If a fusion power plant does not produce high-level radioactive wastes, the decommissioning costs of the plant will be reduced. When the safety of a fusion power plant is established and broadly accepted by society, construction of the power plants near large cities may be permitted, which would significantly reduce the transmission costs of electricity.

Being free from nuclear excursions, a fusion power plant is inherently safe. However, it contains a relatively large amount of radioactivity in the form of tritium fuel and activated material inside the plant. However, the biohazard potential is three orders of magnitude lower for a fusion power plant than for a light-water reactor plant during the operating phase. In the decommissioning phase of a fusion power plant, the biohazard potential after several tens of years of cool-down time is the same order as that of fly ash from a coal-fired fossil fuel plant (see Sec. 1.3.3). Decay heat in a fusion power plant is much lower than that in a light water reactor plant, which suggests there is no need for an emergency core cooling system for safety.

Although the present design study of a fusion power plant shows sufficient safety as described above, further efforts to improve the safety aspects (for example, a reduction of radioactive materials inside the plant, the realization of more efficient confinement of radioactive materials, and development of materials with low decay heat) should be made to increase its attractiveness and competitiveness in the market.

3.7.2 Issues toward a compact fusion reactor

The improvement of energy confinement time is the most important factor in realizing a compact fusion reactor. Confinement enhancement factors (H-factor: the confinement time normalized by the ITER89P scaling) employed in various fusion reactor designs are plotted against the major radius in Fig. 3.7.2-1. The ITER-FDR design (the major radius is 8.14 m), which aims at self-ignition, requires an H-factor of 2.5. It should be noted that an H-factor of 2-3 is sufficient for the design of a compact reactor and that a larger H-factor does not contribute to a size reduction of the device. As discussed in Sec. 3.1, these values of H-factor are obtained in present-day experiments and can be realistically expected in a future tokamak reactor.

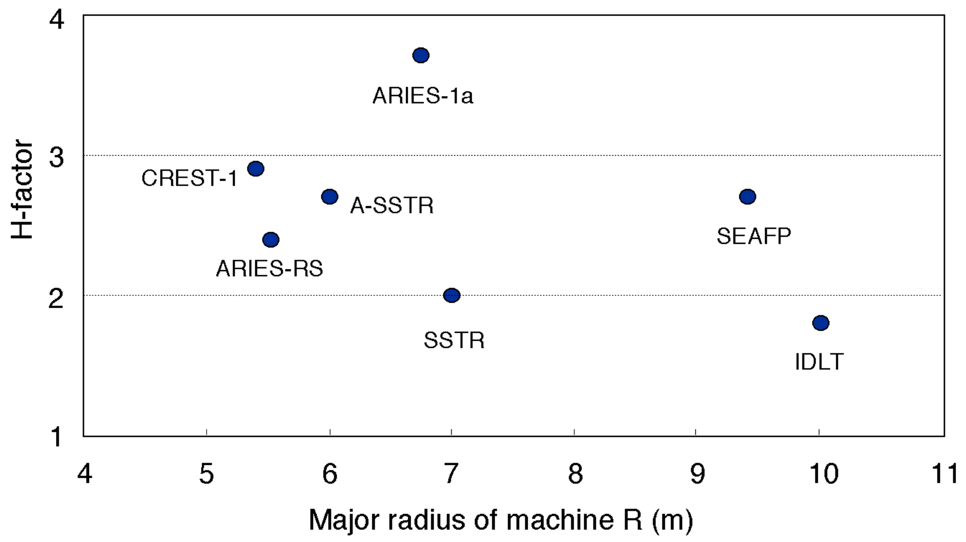

Fig. 3.7.2-1 Assumed confinement enhancement factor (H-factor) for various reactor design.

Fusion power can be approximately described by:

$P_F^{tot} = n_D n_T \langle\sigma v\rangle Q_F V_p \quad \beta_t^2 B_t^4 V_p \quad \beta_t^2 B_t^4 R^3$

where and R, B_t, and β_t are the major radius, the toroidal magnetic field, and the toroidal beta value (the ratio of the plasma pressure to the magnetic pressure: $\beta_t = p/(B_t^2/2\mu_o)$), respectively. For a constant fusion power, an increase in β_t and/or the magnetic field is the key factor to reduce the device size.

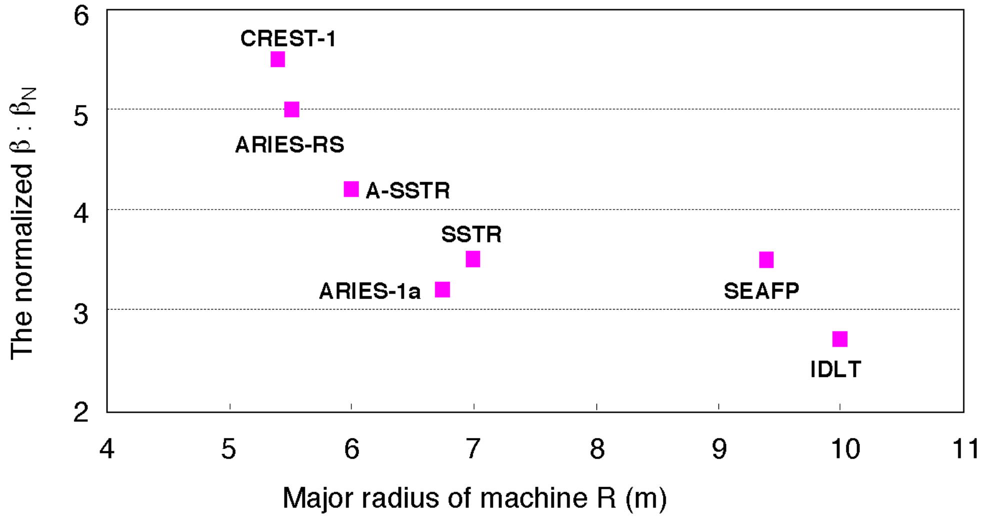

Fig.3.7.2-2 Assumed normalized β (β_N) for various reactor designs

There is an upper attainable limit for β_t. For a given magnetic field, the plasma becomes unstable when the plasma pressure exceeds a certain value. This is called the beta limit. Troyon has proposed a semi-empirical theory for the beta limit;

$$\beta_t \le \beta_N I_p/(aB_t)$$

where I_p(MA), a(m), and B_t(T) are the plasma current, the minor radius, and the toroidal magnetic field, respectively. The constant β_N is called the normalized beta value or the Troyon factor. This equation is valid only for a standard tokamak and is not applicable for a tokamak with an extremely small aspect ratio and for non-tokamak devices. This limit is derived from a theory based on ideal MHD stability. It is reported that the β limit is lower than the Troyon limit when the finite resistivity effect is taken into account.

The normalized beta values assumed in tokamak reactor designs are plotted against the major radius in Fig. 3.7.2-2. This figure indicates a higher normalized beta contributes to the design of a more compact tokamak fusion reactor. To design a tokamak reactor with the major radius smaller than 6 m for example, $\beta_N>4$ is required. A theoretical prediction suggests that a normalized beta value greater than 4, $\beta_N>4$, can be achieved with the resistive wall, which should be experimentally demonstrated to confirm the feasibility of a compact tokamak reactor.

Another key issue for compact reactor design is the high-field magnetic coil. In the ITER toroidal field coil design, a superconducting wire made from Nb_3Sn is employed where the maximum field on the conductors is 12-13 T. In a typical tokamak reactor design such as SSTR and ARIES, a maximum field of 16-20 T is assumed, with the expectation of a significant improvement in high-field superconducting materials, such as NbAl. Construction and assembly the central solenoid coil for ITER have been completed and a pulse operation test for a 13 T discharge will be carried out soon, as discussed in Sec. 3.2. Development of coils with a maximum field larger than 16 T will be undertaken in the near future.

An extremely compact fusion reactor with a significant improvement in core plasma performance requires a large increase in neutron wall load and divertor heat load. Development of an advanced first wall material and technology for efficient heat removal are also essential for a compact fusion reactor. Well-balanced development of both core plasma physics and reactor technology must be promoted to realize a compact fusion reactor.

3.7.3 More attractive concept of fusion reactor based on innovative technologies

Based on the present technology level, the COE of a fusion reactor would not be lower than that of other power plants. However, the economical attraction of a fusion reactor depends strongly on the development of reactor engineering technologies, such as structural materials. It has been pointed out [3.7-1] that the COE of a future fusion power plant could be made competitive with that of a fission power plant based on the assumptions of future advances in technologies and appropriate consolidation of the total plant. Development of fusion compatible materials, which may contribute to the reduction of radio-activated materials from the fusion plant, may make the future fusion reactor more attractive from the viewpoint of economics and safety.

One of the most important technologies to realize an attractive fusion reactor is considered to be the improvement of structural materials, especially low-activation structural materials. Development of applicable low-activation materials for a fusion reactor, such as vanadium alloys and SiC/SiC composite, would substantially reduce radio-activated materials and the time required before the dismantlement of the reactor core after the plant is removed from service. Development of a high-heat-resistance materials, such as SiC/SiC composite, make it possible to design a high-temperature blanket system that would operate at around 1,000ºC. Operating at this temperature would significantly improve the thermal efficiency by employing a gas turbine system.

A compact and high power plant is important to improve the economics of a fusion reactor. An increase in the toroidal magnetic field strength is essential to realize a compact and high power reactor from the projection from the present database on fusion plasma physics. Furthermore, minimization of the electricity consumption inside the plant requires a reduction of the refrigeration load of the superconducting coils as much as possible. The high field, high temperature superconducting magnet is essential to meet these requirements. Development of high-field, high temperature superconducting materials with a maximum field of >20 T at around 20 K is now in progress. Further technological progress will contribute significantly to the realization of a compact and economical fusion power plant.

Continuous studies of high-performance operation of fusion plasmas should be made because these studies will provide the possibility opening a new operation window for burning plasma in a low-field, high-confinement vessel. Such an advance holds the promise of still higher fusion power. A fusion reactor with a very-low neutron flux based on the D-^{3}He fusion reaction is a candidate scenario that offers both high safety and high public acceptance. However, this scenario will require further basic investigation of the plasma confinement scheme, since its requirements for fusion plasma and heat removal are much more challenging than of a D-T fusion reactor.

3.7.4 COE estimate of a fusion reactor

Here we estimate the COE of a tokamak fusion reactor with an electric output power of 1 GW. Although a power plant with a large output power is favorable for a fusion plant, a huge power plant may not be acceptable for the future market of electric power because the market tends to favor decentralization of power plants. But in any event, it is necessary to know conditions where a fusion power plant can be competitive with a power level of around 1 GW.

Figure 3.7.4-1 shows the normalized COE of a fusion power plant against various assumption of fusion plasma parameters [3.7-2, 3.7-3], where ~ 1 GW thermal output is assumed. The white part of each bar graph shows differences in thermal efficiency. The bottom of the white part corresponds to a thermal efficiency of 34-34%, which is the efficiency of a light water reactor / steam turbine, while the top corresponds to a 45% thermal efficiency of an advanced turbine system assuming He gas, liquid metal, or steam in a supercritical state. Improvement of thermal efficiency contributes significantly to the reduction of the normalized COE. The improvement of the Troyon factor, β_n, and the increase in the maximum field of the coils make it possible to increase the plasma pressure and the fusion power output. These improvements are effective in reducing the normalized COE by realizing a compact plant. This is illustrated at the bottom of Fig. 3.7.4-1.

For reduction of COE, there are three important factors: improvement of thermal efficiency, an increase in the magnetic field, and advanced plasma control (improvement of the Troyon factor). The requirement for improvement of the Troyon factor can be replaced partly by an increase in the toroidal magnetic field. For example, by increasing the magnetic field from 13 T to 16 T, the normalized value, β_n, can be reduced by 20% with the same plasma pressure (the same fusion power density). If we assume a fusion plant with a huge output power and with reduced construction costs due to progress in technologies, a further reduction of the normalized COE can be expected.

The normalized COE of a fusion plant depends strongly on the assumptions for the fusion plasma parameters. In Fig. 3.7.4-1, the Troyon factor β_n, which poses the upper limit of the plasma pressure, is used as a measure of a most important physics parameter for cost evaluation. A Troyon factor of ≤ 3 is assumed in ITER design (~ 2 in the design standard). The normalized COE of a fusion power plant with a similar Troyon limit as ITER is shown on the left side of Fig. 3.7.4-1. In this case, the normalized COE of < 1.5, which was discussed as a tentative goal of COE for a fusion power plant in the previous section, cannot be satisfied. The second column from the left in Fig. 3.7.4-1 assumes 50% of improvement in β_n compared with the previous case. As previously discussed, this condition on β_n can be relaxed by an increase in the toroidal magnetic field; for instance, the value of β_n can be reduced by 20% by increasing the magnetic field from 13 T to 16 T. Taking this into account, $\beta_n \sim 4$ may be required to satisfy the COEn < 1.5 with the maximum magnetic field of 16-20 T. If advanced operation with $\beta_n \sim 5$ can be expected, a fusion reactor satisfying the COEn < 1.5 can be designed with a maximum field of 13 T. There are various combinations of β_n and the maximum field that satisfy the COEn < 1.5. An improvement in the thermal efficiency is inevitable in any cases.

Theoretical predictions indicate that $\beta_n \sim 5.0 – 5.5$ can be expected when the conductive shell wall with enough surface area is set near the plasma surface and that $\beta_n \sim 3.5 – 4.0$ is the upper limit without the conductive wall. This wall stabilization effect on the steady-state high performance operation should be confirmed in experiments. The concept of conductive wall may strongly affect the design of the blanket and its maintenance. Conceptual design studies of a fusion power plant with blanket modules containing the conductive wall components were made in the ARIES-RS ($\beta_n \sim 5$) [3.7-4] and CREST ($\beta_n \sim 5.5$) [3.7-5] designs. For an increase in the magnetic field, a maximum field of 16 T can be realized under the extrapolation of the present technology. In SSTR [3.7-6], for example, the toroidal field coils with a maximum field of 16.5 T have been designed. Taking into account the fact that a superconducting material with a maximum field of 20 T has already been produced, the development of large magnets with 20 T is one of the attractive targets to improve fusion reactor economics. An example of such a high-field fusion reactor is reported in A-SSTR design [3.7-1].

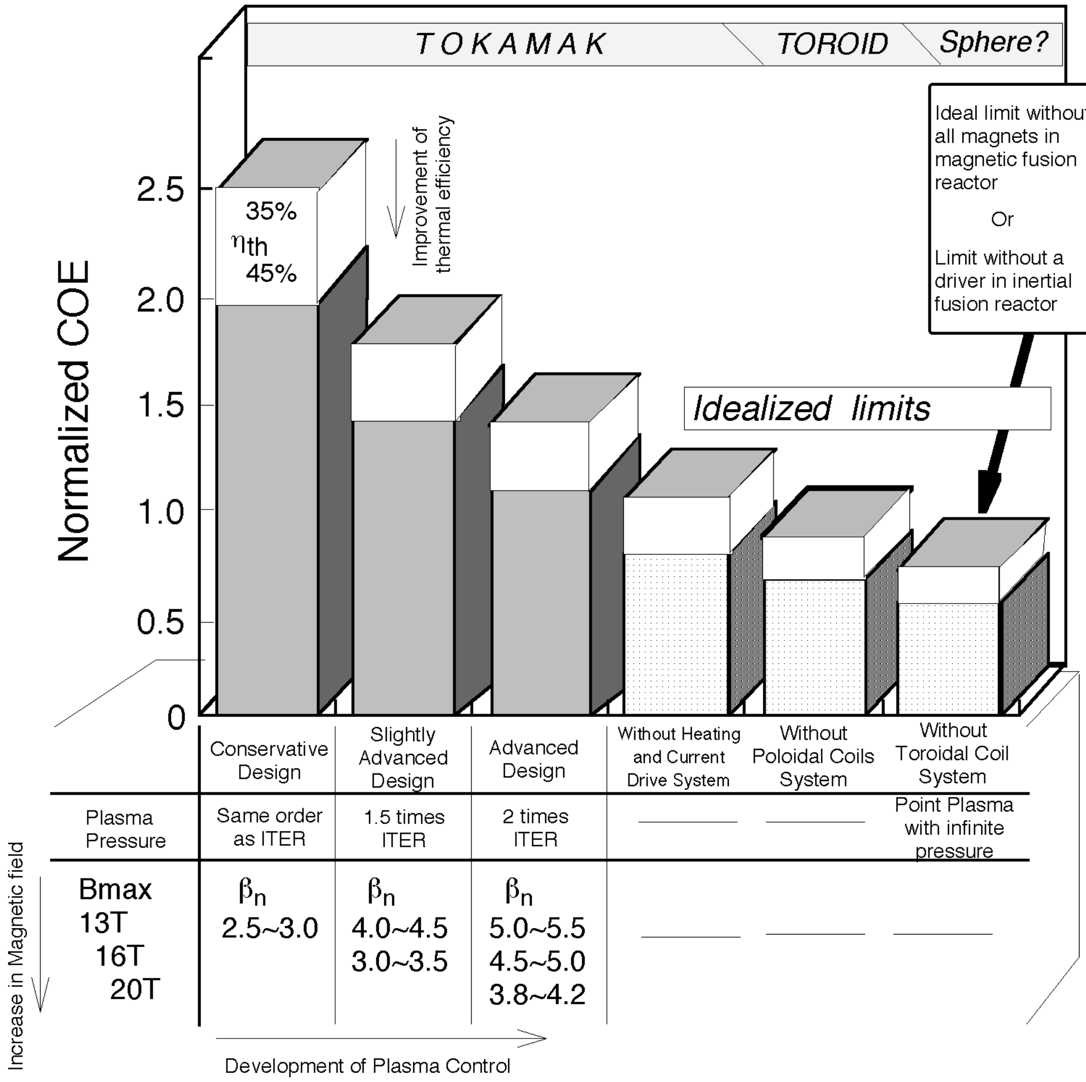

Fig. 3.7.4-1 COE estimate of fusion reactor with an electric output power level of around 1 GW

The three columns on the right hand side in Fig. 3.7.4-1 show ideal cases to emphasize the cost limits for power generation. But these cannot be realized in a real tokamak device. The ultimate limit is shown in the end of right hand side in Fig. 3.7.4-1, where a point fusion plasma with an infinite β value (the ratio of plasma pressure to the magnetic pressure) is assumed. In this case, the size of the device is determined by the limitation of the neutron wall load(*). The normalized COE is around 0.75 with a thermal efficiency of 35%. Improvements of the thermal efficiency, the availability (~ 75% in this case), an increase in the output power, and a reduction of the capital costs by well-developed technologies would further reduce the value of COE.

(*) For the case of point plasma with infinite pressure, a 20-MW/m^2 neutron wall load is assumed. The neutron wall load does not determine the device size for the other cases. For the case of the advanced design, an average neutron wall load of 5 MW/m^2 is assumed.

We can discuss the normalized COE of a fusion reactor based on the inertial confinement in Fig. 3.7.4-1. The normalized COE shown in the right hand side of Fig. 3.7.4-1 is equivalent with the normalized COE of an inertial fusion reactor without taking into account the cost of the driver. If we assume that there is no large difference in fuel expense and the cost of maintenance between the tokamak and inertial reactor, the capital cost for a driver in a 1 GW inertial fusion reactor should be less than the value corresponding to the COE = 0.75 to satisfy the tentative goal of COE < 1.5. By using the relationship between the capital cost and the COE shown in Sec. 1.3.6, the capital cost for a driver in an inertial fusion reactor should be less than 270 billion yen. Although this upper limit of the driver cost may be relaxed by improving the thermal efficiency, the order of this driver cost corresponds to the capital cost of the National Ignition Facility, which is now being constructed in the United States to demonstrate ignition in inertial (laser) fusion. Therefore, this upper limit of driver cost is a severe obstacle in realizing a cost competitive laser fusion reactor. A significant cost reduction of the driver may be one of the most important issues, especially for a laser fusion reactor. In other words, the economic aspect of a laser fusion reactor can be significantly improved when the required laser power for ignition can be reduced. Increasing the electric power output from an inertial fusion plant also contributes to a reduction of COE.

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3.8 Summary - Technological Prospects of Fusion Energy -

From the viewpoint of the "Technological Feasibility of Fusion Energy," the technological issues of a fusion reactor examined and analyzed in Sec. 3.1-3.7 are summarized as follows:

1) Fusion plasma technology: It is judged that self-ignition (Q ~ 20) and long burn in ITER are achievable. Physics issues on fusion plasma required for the demonstration reactor will be solved by studies in ITER on the long pulse control of high Q burning plasma, the advanced steady-state operation, and the high β experiments.
2) Component technology: Since key component technologies required in ITER sufficiently cover the key component technologies of the DEMO reactor, the technological basis of the fusion reactor will be established through the construction of ITER. The technology for the demonstration reactor requires technological enhancement from the present level in some issues, such as the development of high-performance superconducting magnets, which can be expected to be achievable by the steadily continuing R&D of applicable technology.
3) Blanket and Material Technology: For blanket and material technologies, which directly relate to electric power generation, promising materials such as low-activation ferritic steel are already available. High-performance materials that can withstand neutron irradiation of several tens to 200 dpa are expected to be developed by continuous optimization studies of material composition for reduced neutron irradiation damage. Blankets for power generation that can be used in the DEMO reactor condition, are expected to be developed through near-future studies and functional tests of test modules in ITER.
4) Safety technology: A fusion power plant is essentially safe because no nuclear excursion is possible and no high-level radioactive waste is produced. The safety technological basis for fusion reactors will be established through the safe operation of ITER. The safety technology required in the DEMO and commercial reactors is expected to be established by enhancing safety technology realized in ITER operation.
5) Operation and maintenance technologies: Operation and maintenance technologies are important for increasing plant availability, thus, making fusion reactors reliable power sources. Based on the operational experience in large tokamak facilities and the results of ITER engineering design activities, a capacity factor (availability) level equivalent to that in light water reactors can be achieved in a fusion reactor.
6) Technical Issues from the Viewpoint of Industry: The experiences of the manufacturing industry during the construction and improvement of fission reactors are very important for putting a fusion reactor to practical use. Utilization of these experiences is essential for increasing the feasibility of, and accelerating the process of bringing a fusion reactor to the market.
7) Issues to ensure competitiveness in the market: To be competitive in the market, safety should be enhanced taking advantage of the fusion reactor’s inherent safety, and a compact-sized reactor should be realized by enhancing the fusion plasma physics and fusion reactor technologies to reduce the COE. It can be considered that the fusion reactor has a good chance to become competitive in the market after the DEMO reactor phase.

It can be concluded that the technological goals of the fusion DEMO reactor aiming at demonstration of electrical power generation will be achieved by steady progress in the individual technological aspects described above.

PART 2

Extension of the Fusion Program and Basic Supporting Researches

CHAPTER 4. Extension of the program and basic supporting research

4.1. Significance of plasma physics research on ITER in the academic field

Plasma physics is the theoretical basis of fusion researches such as the ITER program. It has greatly progressed, together with fusion research, as a new frontier of science on the "fourth state" of matter, after the solid, liquid and gas states. Various phenomena involving the fourth state of matter, namely plasma and various research areas as an application of plasma physics are developed. These plasma-related areas have been recognized as a new field of science, not only for fusion research, but also for the advanced science of the 21st century. This is due to the fact that any matter is inevitably transformed into the plasma state when the energy density increases or the mutual interaction between the mater and an external field significantly increases. This reveals the complex characteristics of the non-equilibrium, and the usually non-steady state, behavior of plasma. Understanding such characteristics of plasma and acquiring plasma control schemes will provide the competitive capabilities for solving important problems necessary for the advanced science and technology in the 21st century, including astrophysics, beam and accelerator science, high-field quantum optics, and new matter creation.

High-temperature plasma, such as the burning plasma in ITER, is a many-body charged-particle system that exhibits double-fold nature, which is often observed in physics. It has the macro-scale "fluid character," which accompanies electromagnetic interaction, and micro-scale "particle characteristics," where the behavior of individual plasma particles plays an important role. An understanding of both characteristics is necessary to adequately describe the physical phenomena. In a sense, it is similar to the double-fold nature observed in quantum physics known as the "particle" and "wave" characteristics of photons. Accordingly, it is difficult to understand the phenomena in terms of only one double-fold characteristic. The plasma is a typical nonlinear and dispersive medium in which various physical processes develop as a result of the interactions between the externally applied strong magnetic fields and the electromagnetic fields produced by the motion of charged particles in the plasma. The freedom of the plasma interacting with the external electromagnetic fields and minute dissipation appertaining to high-temperature plasmas and the plasma property of retaining (memorizing) individual particle motion lead to the embodiment of various, complex, and plentiful phenomena on earth, which are essentially different from that of terrestrial phenomena and conventional fluid dynamics.

In fact, more than 90% of the volume of the solar system including the central burning region and radiative and convective zone is dominated by hydrodynamic or magneto-hydrodynamic phenomena. In high-temperature ITER plasmas, on the other hand, selective interaction between the collisionless energetic particles and various kinds of fluctuations, finite-inertial effects of electrons in the plasma, infinitesimally small dissipations that cannot be described by the fluid model and related dynamics of the magnetic field lines, and the inhomogeneous nature of the velocity distribution function caused by strong confining magnetic field often plays an essential role. Recently, it has been recognized in several physics fields as the multi time-scale phenomena that an accumulation of physical processes having microscopic time and spatial scales causes the formation of various kinds of macroscopic structures. From this point of view, it is necessary for high-temperature plasma physics research, which aims at the nuclear fusion, to be based on, or to regress to, the "first principle," that of constructing a theoretical basis, where "particle description" rather than "fluid description" and "micro-scale description" rather than "macro-scale description" are emphasized. Accordingly, plasma physics is intimately connected to the "fundamental disciplines" of basic physics and mathematical physics [4.1-1].

These intrinsic features of high-temperature plasmas have produced more important scientific outcomes than was originally expected. Since the latter half of the 1970s and the early 1980s, magnetic confinement fusion research encountered the emergence of various types of instabilities along with the related anomalous transport phenomenon. These phenomena limited the plasma confinement to a very low level and made it difficult to maintain the confined plasma for a long time. Hence, it was believed that plasma is a medium, which is very difficult to control. The concept of "profile consistency" of plasma was contrived in such a historical background. However, during the later 1980s and 1990s, owing to the progress of physical methods for the analysis of these instabilities, several scientific findings, which became a key influence the strategy of nuclear fusion research, were discovered. Initiated by these findings, nuclear fusion research came into a new phase and then exhibited the remarkable progress seen of late.

Examples of such scientific findings are self-induced (bootstrap) current originating from the small dissipation of plasma, and phase transition phenomena and improved confinement observed in the thermal transport barrier that appears in a narrow interior region of the plasma. Latter is related to micro- and macroscopic plasma instabilities and energy transport. These two findings, which were considered to be physically independent phenomena, were merged into a new observation of "self-sustaining character" in which the plasma intends to sustain itself by its self-induced current and improve confinement through various structure formations by itself. From these findings, the classical idea of plasma was abandoned, and it was strongly recognized that plasma is a physically interesting system that allows various types of structure formation on various time and spatial scales.

Such remarkable features are ascribed to the self-organization of plasma sometimes exhibiting dynamical behavior, which is produced by the competition between "fluctuations" that occur when we try to confine high-temperature plasma in a finite volume on earth and the "self-controlled function" of the plasma. Such physical phenomena are characteristic of the "solar system on earth" and are difficult to actually observe in the sun. It is considered that the confinement of the sun corresponds to a low mode (like an L-mode state) in tokamaks, where "fluctuations" dominate the major part of energy transport (anomalous transport). In the sun, large-scale convective cells, which are in a critical gradient state in a convective layer, play a similar role. Tokamaks have an intrinsic degree of freedom to suppress or break such fluctuations by several physical mechanisms, and this is considered to be one of the reasons that tokamaks manifest various distributions.

These phenomena exhibit unique characteristics as a complex system, which is difficult to explain in terms of a single physical model. In other words, as shown in Fig. 4.1-1, many different elements, such as the "structure of magnetic field," the "structure of fluctuations," the "structure of plasma flow," and the "properties of plasma dissipation and dispersion," are interacting with each other. It is thought that such features are crucial for realizing a compact, high-performance fusion reactor. In addition, understanding these phenomena and constructing a scientific basis will increase our understanding of existing classical physics, which describes the properties of matter and thus has great significance in scientific research [4.1-2].

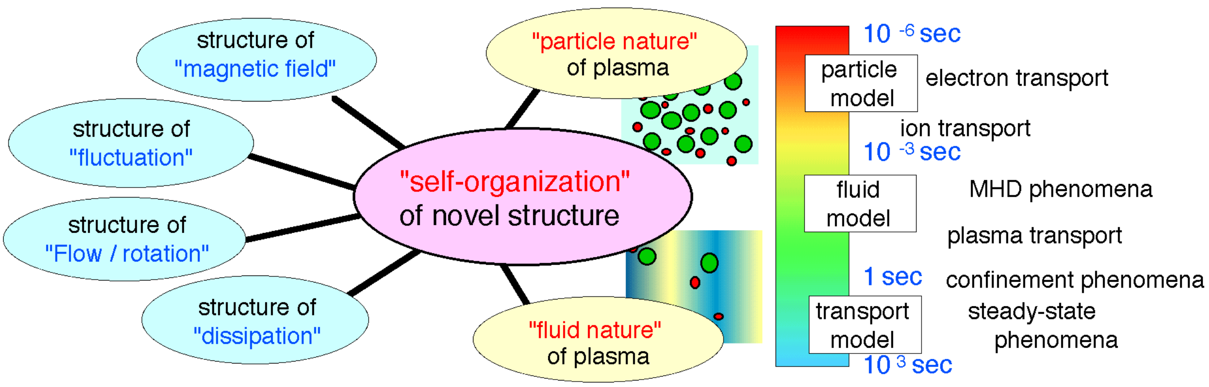

Fig. 4.1-1 Self-organization of plasma as seen in the thermal transport barrier formation and models in each stage that describes various time scales of plasma phenomena.

The features of high-temperature plasma, which ordinary solids and fluids are not endowed with, produce such various behaviors under the specific structure of the magnetic fields of a tokamak. Although the plasmas in the 20th century exhibited these interesting characters, they were sustained by external heat sources, such as particle beams and electromagnetic waves (externally sustained system). On the other hand, ITER plasma in the 21st century is required to achieve a high-energy gain, and it then will attain the features of a "self-sustained system" similar to the sun. In such a system, through interaction with the plasma, high-energy particles generated by the nuclear burning provide almost all the necessary sustaining energy required by the plasma.

Consequently, self-heating in the burning plasma and the resultant pressure increase yield various autonomous phenomena of self-organization and self-control, which are embodied as the "stationary sustainment" and "confinement improvement" in tokamak plasmas. Here, is it noteworthy to state that a nonlinear feedback loop is formed so that these features influence the burning plasma through self-organization (Fig. 4.1-2). The generation and effective control of highly complex and autonomous plasma are indispensable for realizing the stationary sustainment of improved confinement of the burning state with a high-energy gain. Realization of such phenomena is the mission of the ITER project, which is launching new themes of science. The ITER project will develop a new science of complex phenomena in plasma that attains the new freedom of burning, a phenomena that eluded 20th century plasma. Furthermore, it will develop new methods of controlling systems governed by the physics of self-sustained systems.

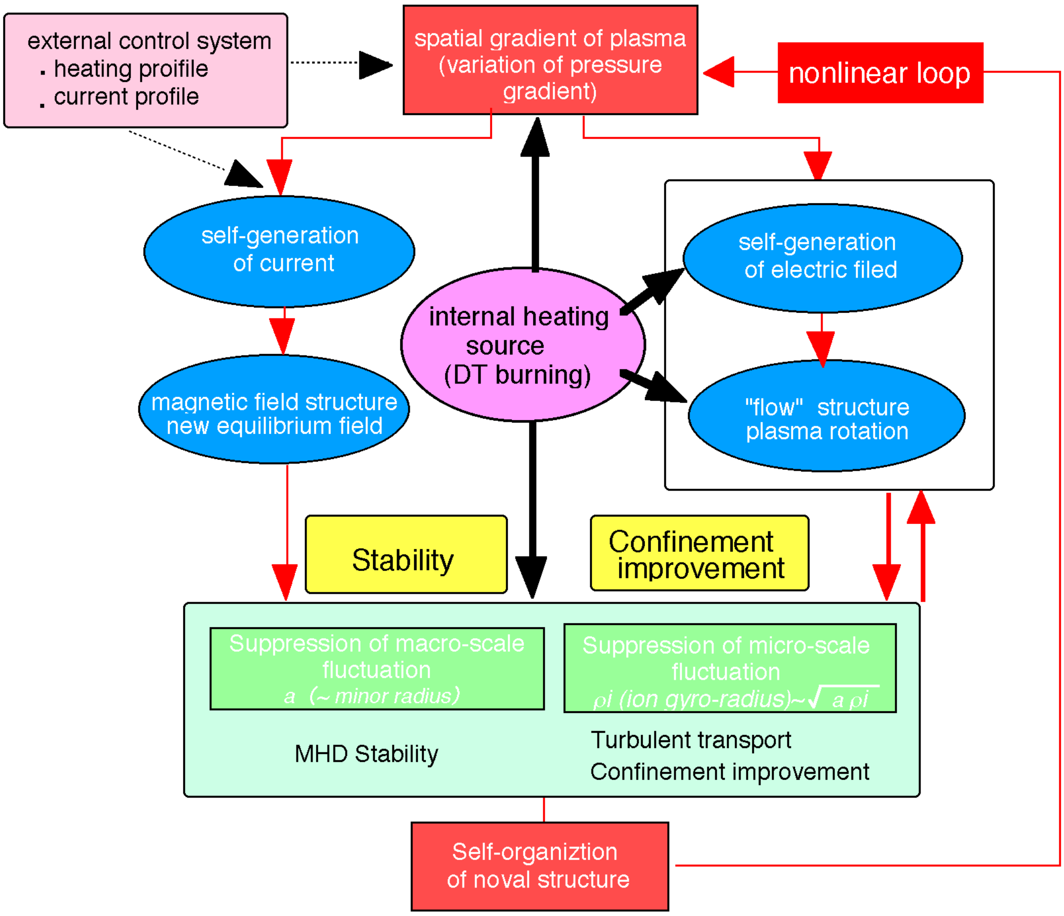

Fig. 4.1-2 Self-organization in burning plasmas and related nonlinear feedback loop

In the 20th century, science and technology were successfully developed by employing the well-defined and also well-systematized laws of physics, such as classical mechanics, fluid mechanics, electromagnetism, statistical mechanisms, quantum mechanics, and the theory of relativity, and applying them to engineering development. The successful application of these laws of physics to engineering was achieved by excluding inexplicable phenomena as much as possible, and by controlling the system from the external environment under the establishment of idealized and also simplified physical situations (externally-sustained system). Indeed, most science and technology achievements in the 20th century (except some examples like laser physics, and chemical and biological systems) were restricted to the regime of science where external control works well.

However, in the latter half of the 20th century, it was pointed out that most of nature is controlled by complex phenomena that are not explained by simplified and deterministic physical processes with a few degrees of freedom. A deliberate effort has been therefore devoted to the study of complex phenomena. Accordingly, since the 1980s, the nonlinear theory, the chaos theory, and the science of complexity have been rapidly developed. Furthermore, new research fields aimed at the study of properties of matter and self-organization phenomena are based on the above concepts and are considered to be the new basic science area to enhance science and technology standards in the 21st century.

The ITER project is regarded as a direct contribution to improving human society, namely "realization of an unlimited energy source on earth." In addition, from the above point of view, this project, which includes the new science of a self-sustained material state, will come to be recognized as one of the most important intellectual undertakings of humankind. The scientific knowledge of burning plasma obtained in ITER will contribute to the understanding of all material states in nature, an accomplishment left for the 21st century. In the field of space physics, for example, owing to the significant progress of the technology for astronomical observation, it has been pervasively recognized that plasma accompanied by magnetic fields plays an essential role in the formation of stars and galaxies, and in various phenomena occurring in stars like the sun. The scientific knowledge of turbulent phenomena, the dynamics of magnetic field lines in turbulence, and magneto-hydrodynamics will be clarified in the burning plasma in ITER, which will have small dissipation. This knowledge will also contribute to the scientific research of the properties of matter and space physics. Moreover, from the viewpoint of engineering, it is expected that the ITER project will stimulate the application of scientific knowledge for the mechanisms of autonomic and complex systems in industrial technology, mechanisms that were not understood in the 20th century.

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4.2 Other benefits of ITER science and technology

Fusion research and development, for which ITER is the core, is focused on the development of a new energy source. The research itself is essentially for the security of the energy source as described in Section 1.4. Below, other benefits of the ITER science and technology are described.

4.2.1 Significance of ITER fusion research in general science and technology

Fusion research is by nature a powerful driving force for the advancement of various fields of modern science and related technologies in addition to its ultimate objective of producing a new energy source for humankind. In other words, fusion research is pervasively contributing to the progress of science and technology in Japan. It has already occupied an important part of academic fields, particularly in the areas of plasma physics and engineering research. Moreover, developments relevant to Japanese fusion research are highly recognized in the international community. This research is driving corresponding research areas and propelling the advancement of industrial technology.

In fusion research, technological standards exceeding present capabilities are required in many areas of science and engineering as a result of R&D being performed for the fusion application. For example, development of a large strong-field SC (superconducting) coil has cultivated the new discipline of the SC material engineering, extremely low-temperature technology, refrigeration engineering, and high-power electrical engineering.

Fusion reactor engineering has accumulated vast amounts of scientific knowledge, technological information, and a fundamental database. It has also suggested the new area of scientific research in the past and re-activated the already established field of science with new concepts. Related examples, such as outstanding achievements that have contributed to the progress of other scientific and engineering developments, are listed in Table 4.2.1-1.

Table 4.2.1-1　Areas of science and technology related to fusion reactor engineering

SC magnet	Thermodynamics, Thermal Engineering, Cryogenic Engineering, Electrical Engineering
NBI	Electrical Engineering (mainly in Ion-Beam Engineering), Atomic and Molecular Science, Plasma Physics
RF	Electromagnetic Dynamics, Quantum Physics, Electrical Engineering
PFC	Heat Transfer Engineering, Mechanical Engineering, Chemical Engineering
Blanket	Metal Engineering, Material Chemistry, Thermal Engineering, Mechanical Engineering, Nuclear Engineering
Reactor Structure	Architecture, Civil Engineering, Mechanical Engineering, Instrumentation and Control Engineering, Robotic Engineering, Radiation Engineering
Fueling and Pumping	Vacuum Technology, Instrumentation Engineering, Electrical Engineering, Mechanical Engineering, Low Temperature Technology, Material Engineering
Tritium	Chemical Engineering, Electrical Chemistry, Physical Chemistry, Analytical Chemistry
Material	Metallurgy Engineering, Metal Engineering, Material Engineering, Solid-State Physics
Neutron	Accelerator Engineering, Applied Physics

The above achievements are not the result of basic research for its own sake—they were accomplished in the course of developments related to the advanced technologies involved in ITER R&D. Further progress would be expected during the construction and operation of ITER. In particular, ITER requires systems integration, the extent of which is enormous. ITER has a complexity never experienced before. Therefore, it is anticipated that ITER R&D will yield impressive advancements in the fields of systems engineering and control engineering.

Further progress in individual scientific and technological fields is expected as a result of participation in the ITER R&D program, which will undoubtedly raise the overall engineering standards. The R&D results, obtained in the most advanced scientific facilities, usually propagate quickly to other areas of high technological development and fundamental science but slowly to commercial products. Technologies developed and advanced in the fusion research program, such as accelerators, superconducting magnets, computer simulation, high-resolution measurements, impurity removal, and fabrication of large precision devices, will contribute to both fusion research and a wide variety of sciences and technologies, including other areas of physics, space development, material studies, and biology.

The technology to develop a specific device itself involves highly specialized engineering, and this provides feedback to fundamental science through the advanced nature of the development involved. In other words, the achievements produced in a high-technology mega-project like ITER embody new situations and environments, which then create new advancements in scientific research. Accordingly, fundamental science and applied technology are thereby stimulated, and they are extended and combined to conceive another science on a larger scale. This cycle in science development may be called "science chain." It is, above all, important as the infrastructure of society, which, in turn, enables the continuous progress of science, personnel training and education, and industrial technology. ITER can assume a substantial role in this science chain as fusion engineering achieves the burning plasma in ITER. The period of the science chain intrinsically has a very long time constant. Therefore, the acquired scientific knowledge and engineering skills are passed from one generation to the next.

The role of fusion research in the system of science and technology is summarized schematically in Fig. 4.2.1-1. ITER indeed has a great significance for both the development of fusion energy and for the advancement of science and technology because ITER is a comprehensive system that requires integrated technology. Fusion research will continue to influence the broad science area as a core, motivating advancements in science and technology, though indirectly, and coordinating cooperation between the world's scientific communities and industries for generations.

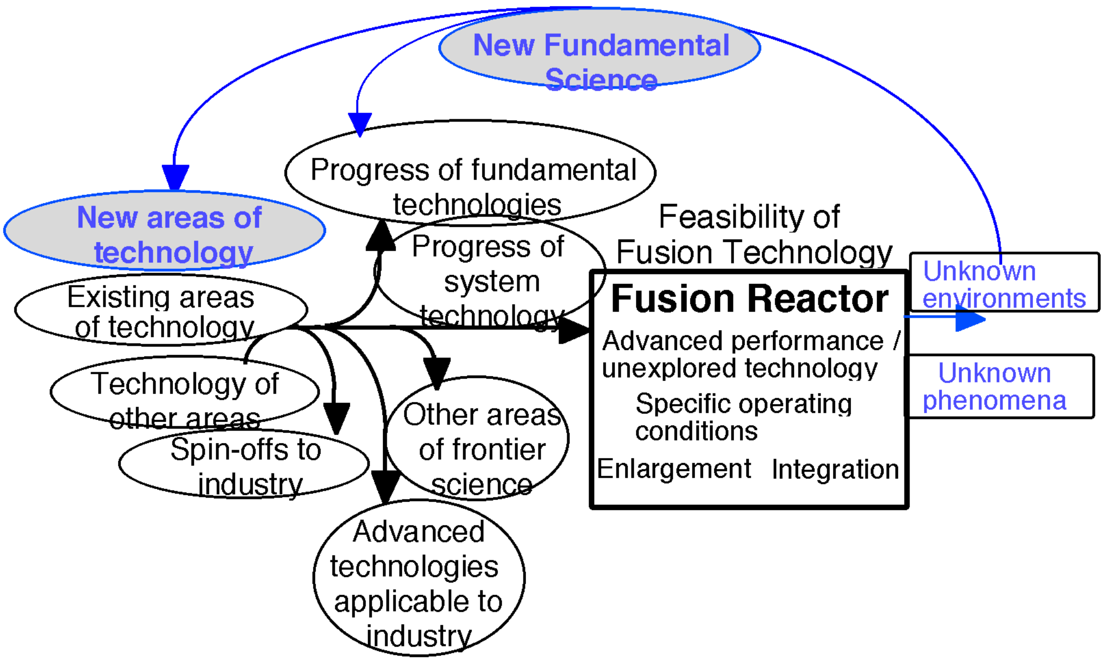

Fig. 4.2.1-1 Position of fusion research in the system of science and technology

4.2.2 Other benefits of ITER construction

Technologies involved in the R&D of the construction of ITER will be applied to fields other than nuclear fusion in the near future.

Spin-offs from technologies developed in fusion reactor engineering, especially those obtained during the ITER design activity phase, can be classified into three major categories.

1) The industrial application is now in use.
2) The achieved technology standards surpass industrial standards and the technology is available for industrial use.
3) The developed technologies involve engineering concepts that can be transferred to industrial applications in the future.

Most technologies in Category 1, which are typical examples of spin-off, have become industrial technology standards; and they are often patented, registered or produced under a license agreement for commercial use. On the other hand, the technologies in Category 2 are advanced and often regarded as a significant development. They are considered to be available for industrial applications within five or ten years.

R&D for advanced technologies of different fields is usually performed individually due to specific needs and engineering requirements. Therefore, the technological achievements seldom have influence in a fixed direction, but the results of R&D achieved in those fields interact with each other. The technologies in Category 3 have such characteristics. Accordingly, they potentially have the capability of being applied in other fields of research in the future. However, their impact is not readily apparent.

Typical examples of the three categories of technology are listed in Table 4.2.2-1. Superconductivity technology is applied to cable, refrigeration machines, and current supplies have broad applications, such as the super-conducting magnets for super colliders. Moreover, this technology has contributed much to reduce costs. The application of super-conducting technology to future energy storage systems is expected.

Table 4.2.2 – 1 Examples of spin-off technologies from fusion related R&D

Reactor engineering field	Industrial field	1) Practical application technology	2) Available technology	3) Common technology
Super-conducting magnet technology	Electrical and machinery	Super-conducting cable and device; Refrigeration machines	energy storage	Decontamination
Blanket	Electrical and machinery		Turbine; Liquid crystal	Neutron detector
Remote handling	Machinery	Robot	Piping; Maintenance; Welding	Atomic energy; Robotic control
Plasma facing components	Machinery		Boiler	Spacecraft
Supply and exhaust of fuel	Vacuum; Instrumentation	Mass spectrography		
Tritium	Semiconductor; Chemistry	Dry gas; Pump	Reprocessing; Hydrogen	Biotechnology
NBI	Semiconductor	Ion beam	LSI; Solar cell	Semiconductor manufacturing
RF	Ceramics	Electron tube	Ceramics processing	Radar; Communication; Electric power transmission
Material	Metal		Boiler; Turbines	
Neutron	Medical; Accelerator			
Plasma application	Waste disposal; Semiconductor			

The ion source used in the neutral beam injection system for plasma heating has the largest current and cross section available; it is applied to the production of semiconductors and liquid crystals. In Fig. 4.2.2-1, its operating principle is shown. The high-power mm wave source (gyrotron) developed for plasma heating is already applied to ceramic sintering. In addition, the gyrotron is expected to be used as a power source in new material processing technology, which employs chemical vapor deposition (CVD), aimed at the production of semiconductors, synthetic diamonds, and the surface processing of other materials. In the future, its application to high-resolution radar, and power transmission by microwaves may be possible (Fig. 4.2.2-2).

The high-vacuum technology and robotic engineering, which have been developed for remote handling in ITER, have potential applications as maintenance tools in nuclear power plants. The vacuum and tritium technologies have a newly developed vacuum pump and gas analyzer and these are anticipated to be available for industrial use in the near future.

In general, fusion reactor engineering is systems engineering that integrate the advanced technologies of broad scientific and engineering disciplines. It is clear that this field has made significant contributions in various fields, by providing

- an influence to commercial products,
- an influence to other fields of science as an advanced technology, and
- new knowledge and concepts to expand the bounds of fundamental science.

Most technologies in fusion research have either contributed to high-end performance or promoted R&D under extreme conditions. The cost for fusion R&D is generally high and the applications are restricted. However, the contribution of fusion technology to future technology in other fields seems substantial, e.g., fusion technology is often applied in high-energy physics and space science, which require technological standards that are much higher than those for industrial products. Nevertheless, such advanced technologies may have applications elsewhere in the near future. For example, new materials produced using space technology may become consumer products.

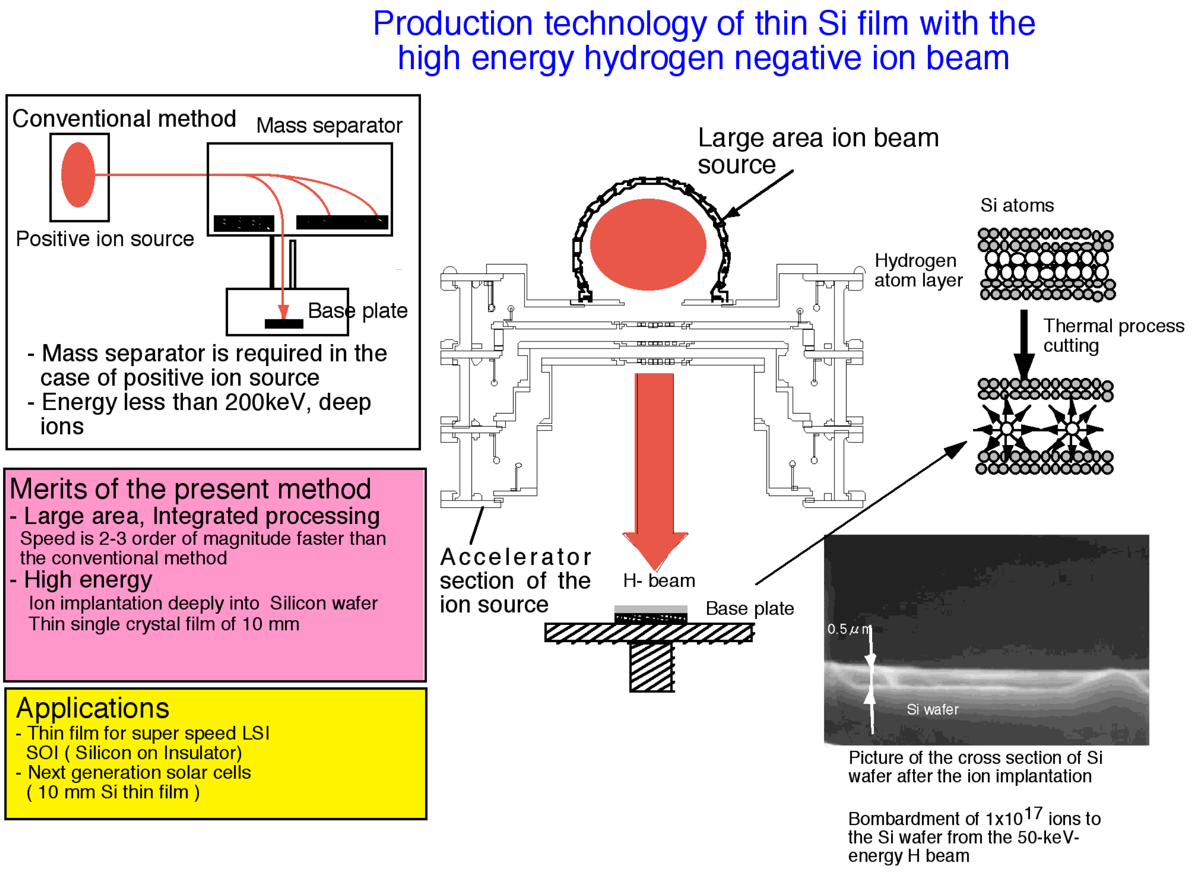

Fig. 4.2.2-1: Example of the industrial application of the negative-ion beam developed for plasma heating. One of the useful applications of the negative-ion beam system is the slicing of single silicon crystals. Injection of a large current ion (hydrogen) beam focused onto the single crystal silicon wafer delaminates the silicon thin layer by the deposition of hydrogen. The uniform single crystal layer having a 10-micron thickness was the first successfully produced in the world. It is expected to be used in high-efficiency solar cells and wafers of the next generation LSIs.

It should be noted that these results are attained for specific purposes, i.e., fusion application. Fusion research is not aimed at spreading its technology to other, or at influencing, basic research.

In addition to its direct contribution to science and technology, fusion research also provides a substantial educational impact. Many engineers and research scientists are being educated in universities, research institutions, and industry through research and development in fusion science. These engineers and scientists contribute to the progress in many other research areas. The universities need a stimulus to proceed with advanced research and the industry hopes to develop products with minimum investment risk. From these points of view, fusion has contributed to the progress of technology. In particular, international cooperation is very active in fusion research, such as ITER-EDA, which also produced international personnel exchanges. As a result, many distinguished foreign scientists remained in Japan for extended stays. Such exchanges stimulated the Japanese academic society and industry, and provided many beneficial interactions that grew to an intimate international network. This network includes not only the individual parties of ITER but also other countries that are interested in ITER. Fusion research indeed provides an excellent opportunity for Japan to take the initiative to provide a world-class scientific achievement.

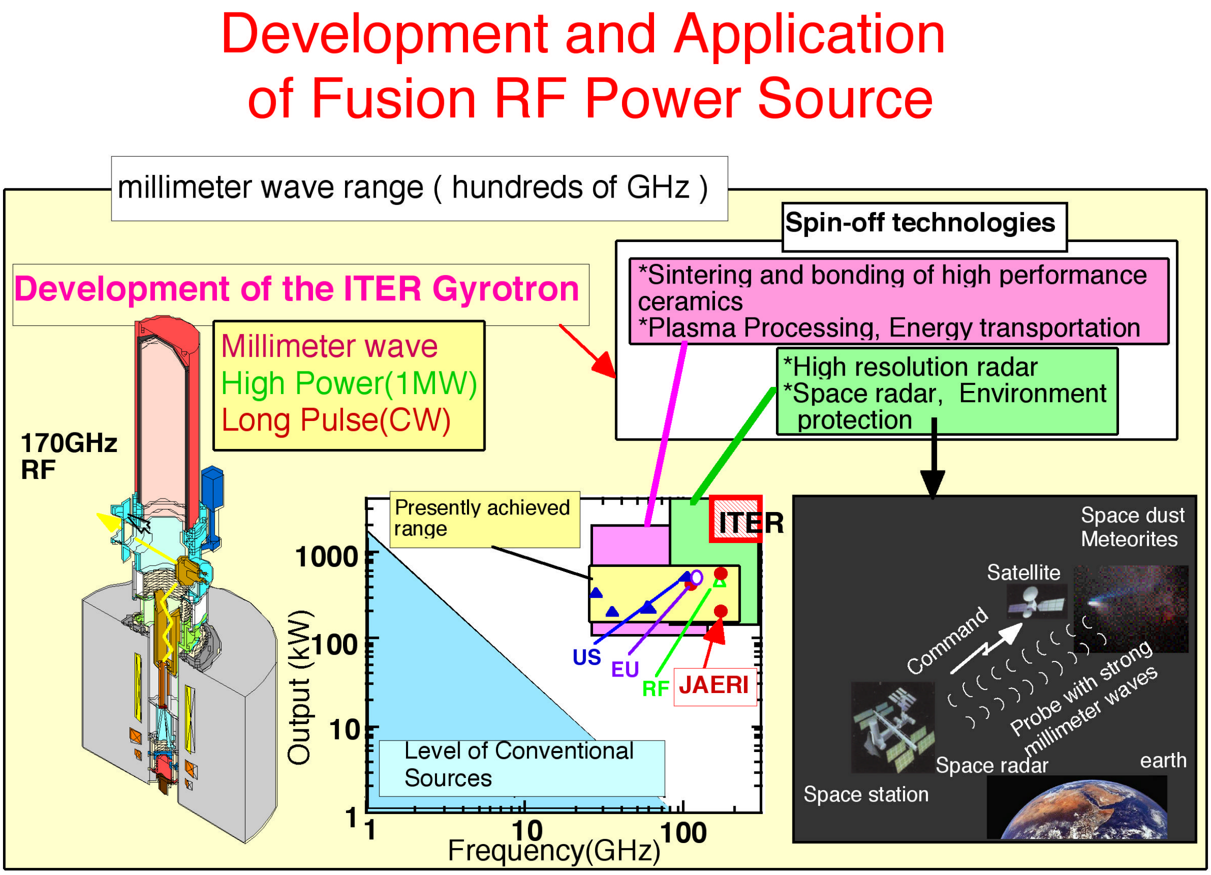

Fig. 4.2.2-2 The RF power sources developed for ITER and fusion research have an output power one order of magnitude higher than conventional ones. The high-power klystron is often used as an accelerator, and the gyrotron is applied to ceramic bonding and sintering techniques. The gyrotron will be used for processing semiconductor and new materials, such as diamonds. In the future, the gyrotron may be applied to space radar and RF power transmission by taking advantage of its characteristics, i.e., high power density, high resolution, and precise focusing.

4.3 Research on the advanced reactor systems and basic studies on materials and reactor engineering

4.3.1 Background and significance of research on the advanced reactor systems

4.3.1.1 Introduction

Research on various fusion confinement systems has been pursued since the early stages of the development of nuclear fusion. Several approaches, simultaneously undertaken in the past, were pursued in parallel without converging personnel resources or funding to select a specific confinement system. This was necessary because the most appropriate plasma confinement scheme remained unidentified as a candidate for the future fusion reactor. This is referred to as a multi-path approach.

Therefore, in the fusion research programs at Japanese universities about 20 years ago, numerous medium-sized experimental devices were built, including the tokamak, stellarator, cusp, bumpy torus, and REB machines at the Nagoya University Plasma Laboratory. A mirror device was constructed in Tsukuba University, and a reverse field pinch machine was built in Tokyo University. Other devices were a Heliotron and tokamak in Kyoto University, a laser fusion facility and FRC in Osaka University, and a superconducting tokamak in Kyushu University. Vigorous and intensive research examined these various confinement systems. Research on the reverse field pinch device has been carried out for many years at the Electrotechnical Laboratory, an agency of Industrial Science and Technology of the Ministry of International Trade and Industry. The Japan Atomic Energy Research Institute began to study the tokamak system in the 1970s. This configuration was then receiving worldwide attention. The Japan Atomic Energy Research Institute then initiated the construction and experiments of medium-sized devices. The large experimental device JT-60 was built based on the experimental results obtained.

Fusion plasma research has made rapid progress in the past 40 years. The fusion product (triple product of plasma temperature, density, and confinement time), which is an index of the fusion plasma performance, was improved by 7 or 8 orders of magnitude. Having excellent plasma confinement performance, tokamaks presently have the most prominent plasma parameters in the world. Accordingly, the tokamak device is employed for ITER, which aims at burning plasma research.

Although the fusion plasma research performed to date has resolved the characteristics of high-temperature plasmas, and achievement of the self-ignition condition is presently within our capability, the fundamental physics involved with fusion plasma is not adequately understood, and reactor design must largely resort to the semi-empirically obtained scaling and extrapolation. Moreover, high-temperature plasmas exhibit very complicated behavior, such as the self-organization phenomenon. It is possible that in the future that very-high-performance plasma, which surpasses our present prediction capabilities, could be developed by the application of our accumulated knowledge in high-temperature plasmas. On the other hand, we would have to consider the fusion reactors that use advanced fuels, such as D-D and D-^{3}He. They show properties superior to those of the DT reactors presently considered, especially in the sense that they do not require tritium breeding and produce fewer energetic neutrons. Since high-temperature plasma research has many subjects in common with various scientific research fields, such as space plasmas, scientific interactions have become more and more active in the broad research areas. Furthermore, advanced technologies on plasma production, control, diagnosis, etc., which were cultivated in fusion research, have been transformed into the innovative plasma application technologies used in the semiconductor industry and other fields. Scientific and technological spin-offs from high-temperature plasma research have been observed abundantly and additional benefits of fusion research can be expected in the future.

From the viewpoints mentioned above, the significance of fusion plasma research with various non-tokamak confinement devices has been specifically recognized as much as the importance of the tokamak in fusion reactor development. Based on such a background, intensive research on non-tokamak confinement systems has been continued. Aiming at the development of a more attractive fusion reactor, the key words *stead-state*, *compact*, *high beta*, and *advanced fuel* have been used mainly in universities in our country for many years. Therefore, the practical approaches of proceeding with fusion research with such an advanced reactor system[1] (the so-called alternative system) have been mainly discussed in the Fusion Sectional Meetings (the present Atomic Power Sectional Meetings), the Specific Research Promotion Subcommittee, the Science Council, together with plans for carrying out nuclear fusion research at the universities in Japan. Below, the significance of research on the advanced reactor system is summarized, according to the discussions at the above meetings and the sectional meeting reports.

4.3.1.2 Arguments on future fusion research at universities

[1] The advanced reactor systems here are plasma confinement systems other than tokamak, such as the helical system, the inertial fusion system, and so on. They are named as so-called alternative systems.

The Nuclear Fusion Sectional Meeting of the Specific Research Promotion Subcommittee, the Science Council presented a report on future fusion research at universities on February 14, 1986. The report stated, referring to the status of fusion research in our country and role of university research programs, that fusion research at universities in the future… (sentences omitted)… should be securely pursued, in order to verify various concepts to demonstrate steady-state and high-efficiency plasmas and to realize compactness and simplification of fusion reactors at each stage of development of their systems. Accordingly, the report concluded that university research programs should consider the following as the next steps. (1) For the open-end system and the inertial nuclear fusion program, advanced research should be organized, utilizing existing devices for the present. (2) For the internal current devices, advanced research with small- and middle-size devices should be coordinated with much flexibility, mindful that the large tokamak program is being undertaken. (3) The external conductor system should contribute to the comprehensive understanding of the toroidal magnetic confinement plasmas, taking advantage of the current-less steady-state plasma research. As a result, it was proposed that a large helical device with an external conductor be considered suitable as a new large-scale program at universities. The Ministry of Education accordingly established the National Institute for Fusion Science in 1989 in Toki City, Gifu Prefecture, and the construction of the large helical device (LHD) commenced in 1990.

After the LHD device was completed and the plasma experiments started in 1997, arguments on coordinating the future research plan at Japanese universities were intensively presented in the Atomic Power Sectional Meeting of Specific Research Promotion Subcommittee, the Science Council, the Ministry of Education. In the meeting, each related university organization reviewed the status of its fusion research activities, the results, and future plans. Intensive discussions intended to identify the role of advanced and fundamental fusion plasma research, its accomplishment by related research institutions, and the coordination of fusion technology research took place. Thereupon the report "About the Development of the Nuclear Fusion Research at the Universities" (January 19,1998) was prepared.

In this report, the significance and the future plan of fusion research at the universities in Japan were summarized as follows.

The university program in our country adopted the helical system as the main device for fusion research. In addition, basic science research on various confinement systems, such as the physics involved in the toroidal magnetic confinement system, the magnetic mirror system, the inertial confinement system will also be pursued at the universities to acquire the basic knowledge necessary for the systematic understanding of fusion plasmas and construction of fusion reactors.

The progress and status of the research, issues of concern related to proceeding to the reactors, and the plan for the future of helical devices, the tokamak system, and other plasma confinement systems (the magnetic confinement systems with toroidal current except the tokamak, the magnetic mirror system, and the inertial containment system) are described in the report. Finally, the report comments on the fusion research with such a diversified scheme, as follows.

As described above, the R&D efforts at Japanese universities are focused on the systematic understanding and broadening of the knowledge of plasma physics, which is necessary for the control of high-temperature plasmas. The universities do not aim at world-record fusion parameters, in contrast to the Japan Atomic Energy Research Institute where development of the reactor with relevant plasma is the ultimate objective. However, the universities will pursue investigations on the complex behavior of plasmas and unresolved high-temperature plasma physics. The systematic understanding thus obtained is indispensable for the coordination and advancement of fusion plasma development and for finding the solutions to realize a compact and efficient reactor core. Newly acquired knowledge in plasma physics would produce substantial benefits to other fields of science and innovative research topics could be deliberated. Furthermore, most new technologies, such as various plasma production schemes, plasma heating technology, fuel injection technology, plasma diagnostic technology, and high-voltage power control technology, developed along with the experimental nuclear fusion research will be extensively applied to industrial fields where their impacts will be substantial.

4.3.2 Status of the advanced reactor systems and their prospects for use on the fusion reactor

4.3.2.1 Helical system [4.3.2-1, 4.3.2-2]

The concept of the helical system, categorized with the toroidal-magnetic-confinement devices, is based on the concept of the stellarator by L. Spitzer in the1950s. The various optimization schemes of this magnetic configuration have been hitherto experimentally examined; its historical background is shown in Fig. 4.3.2.1-1. Due to the complexity of three-dimensional non-planar-axis toroidal geometry, optimization of the helical system does not lend itself towards a unique solution so far as the various aspects of physics and engineering, such as plasma confinement, MHD stability, and engineering feasibility, are concerned. Therefore, there exist quite a number of variations in magnetic configuration. However, it should be noted here that the substantial progress in recent computing technology has facilitated the optimization of complicated three-dimensional structures.

The characteristics of the helical system are summarized as follows, in comparison with the axi-symmetrical toroidal-current- confinement-system, such as tokamaks.

General advantages:
- Steady-state capability is inherently provided
- No current disruptions
- Less circulating energy, which leads to an efficient reactor concept
- Equipped with a natural divertor, which is effective in impurity control
- The control of the magnetic configuration from the outside is straightforward

Drawbacks:
- Theoretical prediction is relatively difficult due to the non-axi-symmetric magnetic configurations
- Particle confinement capability is relatively low due to its non-axi-symmetric geometry
- The helical coil system is complicated from an engineering viewpoint

The fact that the current-less plasma is produced in the helical system should in particular be emphasized in tokamak research. The helical system does not require the non-inductive current drive facilities for steady-state operation, and it does not have instability-induced rapid current decay (current disruption), both of which are critical issues in the development of a tokamak reactor. This is certainly a substantial advantage.

The fundamental difference of the helical system in comparison with the tokamak device is that the confinement magnetic field, which produces the current-less plasma, is sustained only by the external coil in the helical system. However, there are many similar features in the helical and tokamak devices, as they are both categorized as toroidal confinement systems. Therefore, the comprehensive understanding of plasma physics observed in the helical and tokamak devices would contribute much to the general picture of plasma properties in the toroidal system. It is expected that such effort would also contribute to the ITER program. The confinement time and operation density strongly depends on plasma current, and the current distribution is important for the equilibrium formation and is critical for the stability of the tokamak plasma. Investigations on the current-less helical system are expected to complementarily redound to the general and universal understandings related to the role of plasma current in tokamak research.

Helical devices planned, constructed, and operated worldwide that are designed to optimize plasma properties from different aspects are listed in Fig. 4.3.2.1-1. The largest device is the LHD (Large Helical Device) of the National Institute of Fusion Science (NIFS), which started operation in March 1998. LHD has a major radius of R = 3.9 m, and all helical and poloidal coils employ superconducting materials. A similar-sized device, Wendelstein VII-X, is under construction in Europe (Germany). Smaller devices (R = 1-2 m) are CHS and Heliotron-J (Japan), the Wendelstein VII-AS (Germany), TJ-II (Spain), HSX (the United States), Uragan-3M (Russia), and H-1 (Australia). Intensive research on plasma confinement, and stability are carried out at present. The quasi-axi-symmetric devices with extremely reduced helical ripple, such as the proposed QAS and QOS, expected to proceed with optimization research from a different approach.

The following is a survey of research conducted on helical systems with emphasis on LHD in Japan. The construction of the LHD device is based on the recommendations of Fusion Sectional Meetings (the present Atomic Energy Sectional Meetings), the Special Research Coordination Subcommittee, the Science Council. These recommendations, "On the Future Fusion Research at Universities," were issued on February 14, 1986. The construction of LHD started in 1990 at NIFS. The facility was completed as planned in December 1997, and the first plasma was successfully produced in March 1998 following numerous pre-operational tests. The principal research objectives of the LHD project, which will complement tokamak research, are the achievement of high-temperature and high-density plasma, plasma transport research for extrapolation to reactor plasma, achievement of an average beta value higher than 5%, studies of MHD characteristics, investigations on divertor performance, and research on steady-state operation, α-particle simulation, and complementary research to tokamak and comprehensive understanding of plasma properties in a toroidal device.

One and half years after the start of plasma operation in March 1998, the stored plasma energy was increased successfully, as shown in Fig. 4.3.2.1-2. This result is due to an increase in the magnetic field and the heating power as well as the conditioning of the vessel wall and divertor. Plasma having a higher temperature and density is anticipated in the future using increased heating power and pellet injection. The specifications of the LHD device are shown in Table 4.3.2.1-1, while the plasma parameters attained to date are listed in Table 4.3.2.1-2. It should be emphasized here that the energy confinement time is 50% higher than predicted and an average beta value exceeding 2% has already been attained. Figure 4.3.2.1-3 is a comparison of the confinement time between the empirical scaling used at the time of the LHD design and the experimental results. An energy confinement time of about 1.5 to 2 times higher than the prediction from the empirical scaling was obtained in the LHD device. An improved confinement mode was obtained. This mode is similar to the H mode in a tokamak and had a remarkable confinement improvement at the plasma periphery. The impurity accumulation was not an issue of concern in the experiment employing additional heating with NBI, ECH, and ICRH, where effective plasma heating was confirmed. Further experimental investigations with increased heating power are planned.

The 1 GJ superconducting coil was successfully operated, and the fabrication of the helical windings within a dimensional accuracy of 2 mm is indeed a noteworthy technical achievement. The development of heating devices, such as the long-pulse gyrotron oscillator and the negative-ion source, as well as the development of diagnostic instruments, such as FIR interferometer, high-performance Thomson scattering apparatus, and heavy-ion-beam probe were also completed.

Heliotron-E, CHS, and Heliotron-J are medium-sized helical devices in Japan. Heliotron-E was a reference prototype for the design of LHD, and it has provided outstanding results in the PoP (Proof of Principle) experiments conducted in this helical device. However, it was shutdown after the start of LHD experiments. The aspect ratio of CHS was reduced, as was the machine size, which produced distinguished experimental results related to plasma properties in this helical system. For example, the relation between the toroidal rotation and the neoclassical viscosity of plasma was resolved, and the effect of the radial electric field shear on fluctuations of the electrostatic potential inside the plasma and in the internal transport barrier were clarified by the heavy-ion beam probe diagnostics. These results are directly relevant to the physics of improved confinement, such as the H mode in a tokamak, and they contributed much to the comprehensive understanding of plasma properties in the common toroidal confinement system. Furthermore, an average beta value of 2.1%, which was the highest in the world for high-beta research in a helical system, was achieved.

Heliotron-J is a helical device having 3 dimensional axis and helical windings of L = 1 / M = 4. As shown in Fig. 4.3.2.1-4, two sets of toroidal field coil were installed. The bumpiness of the toroidal field is controlled by changing the current ratio of the coil set, which produces a quasi-poloidally symmetric magnetic configuration. It is anticipated that the bumpiness of the magnetic field may be able to reduce the neoclassical diffusion and improve the MHD equilibrium and stability properties.

4.3.2.2 Inertial confinement fusion system [4.3.2-3]

Inertial confinement fusion systems are based on the principle that intense laser beams and/or strong ion beams irradiated to a solid fuel pellet having a diameter of a few millimeters will instantaneously generate a high-temperature, high-density plasma state for a fusion reaction. Figure 4.3.2.2-1 shows the parameter range of plasma density and confinement time in both the inertial fusion system and the magnetic confinement fusion system. It is easily seen that the parameter range of each system is substantially different. In the inertial fusion system, the time duration of the fusion reaction is very short (10^{-11} sec), and it is necessary to compress the plasma and raise its density to a value 1000-10000 times higher than that for solid materials. In this section, the laser fusion system is highlighted as it provides the most advanced research results, employing the laser as a driver of the inertial fusion. The recent status of the research as well as the difference and / or complementary capabilities of the inertial fusion to the magnetic confinement system are reviewed in this subsection.

In the1980s, large-scale laser apparatuses of the several ten kJ class were constructed in Japan, the USA, and France, and laser inertial fusion research made substantial progress. Figure 4.3.2.2-2 shows the plasma parameters achieved to date in laser fusion research. The GEKKO-XII machine in the Osaka University has successfully attained a highly compressed solid density approximately 600 times greater than solid material, which is a world record. Based on this outstanding result, the next step for the program, a facility named the National Ignition Facility (NIF), which aims at the self-ignition, was recommended and its construction started in the USA in 1997. This facility is composed of 192 laser beam lines with a total energy of 1.8 MJ. The fusion output in this program is expected to be more than 20 MJ (an energy gain of more than 10) for a single implosion. A similar scale device, named Laser Mega Joule (LMJ), is planned in France.

The laser fusion process is as follows. (1) Irradiation of laser power (laser beams are irradiated uniformly to the surface of a tiny spherical pellet). (2) Acceleration (the pellet surface explodes, which accelerates the pellet particles inward). (3) Compression (a compressed core is formed, composed of the hot spark point and the high-density main fuel portion). (4) Ignition and burning (ignition starts at the hot spark point and fusion burning propagates into the main fuel). Two schemes of laser irradiation, direct and indirect, are proposed. In the former, the multiple laser beams irradiate the fuel directly, whereas the laser beams are converted into X-rays that in turn irradiate the fuel in the latter. Also in the compression and ignition processes, two schemes are proposed, namely, central ignition and fast ignition.

Central ignition is a method where the central part of the compressed core (hot spark point) is heated to the ignition condition by compressive force. In this method, however, the width of the mixture layer would have to be suppressed to within a permissible limit, as Rayleigh-Taylor instability is induced at the boundary region between the hot spark point and the main fuel. On the other hand, fast ignition is the method in which the peta-watt laser beams are injected into the center of the main fuel continuously, so that the hot spark point is sustained to heat the main fuel, even though the initial hot spark point disappears in the fluid mixture process. The necessary laser energy and the target gain for the direct and indirect irradiation methods and for the central ignition and for the fast ignition are shown in Fig. 4.3.2.2-3. The NIF and LMJ facilities mentioned above are aiming at central ignition with the indirect irradiation method. On the other hand, the fast ignition method requires a super-high-power laser apparatus with an output power of peta-watts to sustain the hot spark, although several hundred kJ is adequate simply to achieve a substantial target gain. Owing to recent progress in short pulse compression techniques, a huge glass laser of the100 tera-watt class is being constructed in the Osaka University, which concentrates on the performance of high-gain inertial fusion research by the fast ignition method.

In the inertial fusion method, direct control of the fusion output is possible by controlling the number of injected fuel pellets and the repetition period of the pulsed laser. Therefore, stabilized power control or variable power control is expected to be straightforward. It is also possible to design a real-time load following for the permissible range of thermal design. Although the disruption phenomena observed in tokamak operation does not exist, a malfunction in the pellet injection system may occur in the inertial fusion system. In addition, misalignment of the optical equipment can also reduce the output power. Nevertheless, it seems possible in principle to sustain the output power by the controlling the repetition period of the laser pulses.

The reactor designs for inertial confinement fusion and magnetic confinement fusion are substantially different, and it is not simple to find common features from a plasma physics aspect. In addition, many engineering requirements are specific to inertial confinement fusion, such as the development of high-power and high-repetition-rate laser apparatuses and the production of a homogeneous pellet. However, in the area of reactor engineering and technology, the inertial and magnetic confinement systems have several common issues of concern. Table 3.2.2.2-1 lists important issues in the reactor technology of the inertial fusion system, many of which are similar to those for the magnetic confinement fusion system, such as ITER.

The high-power laser technology developed for pellet implosion, as well as other R&D devices, such as the peta-watt class laser, the semiconductor excited solid laser, and the KrF lasers, are expected to be implemented not only in the field of fusion but also in broad industrial applications. The feasibility of accelerating particles by a high-power laser has already been examined, which is of great interest to scientists involved in high-energy physics research. In addition, the physics involved in the hot and high-density plasma produced in the laser fusion is relevant to astrophysics, where the state of supernovas and gamma-ray burst phenomena are being investigated. Accordingly, much attention is paid to this area of research, called *laboratory astronomy*. In particular, light and X-ray radiation have a significant role in this field and a new area of science named *radiation fluid dynamics* is being investigated. Also in the divertor region of ITER, the radiation process has an important role, one that is intimately related to the atomic processes and radiation studies investigated in the inertial fusion research and radiation field research.

4.3.2.3 Mirror system [4.3.2-4]

The mirror system is categorized as being in magnetic fusion research, and is referred to as a plasma-confining scheme that makes use of the spatial variation of the magnetic field strength. An electric field may also be applied at the open end of the magnetic field line. The most important concern for the mirror system is to reduce the plasma loss from the open-ended straight field line. The most recent investigation effort is mainly on the tandem mirror configuration, where a couple of small mirror coils are located at both ends, being intended to reduce the end loss by electrostatic potential formation. Figure 4.3.2.3-1 shows the coil arrangement of the tandem mirror device as well as the spatial profile of the magnetic field line and the electrostatic potential, where the hot, high-density plasma is confined in the central cell, the central region of the machine. The hot plasma drifting along the magnetic field line is bounced back toward the central cell at both ends of the mirror field. Nonetheless, some plasma is not reflected and escapes to the anchor and the plug regions. As shown in Fig. 4.3.2.3-1, controlling and optimizing the magnetic-field and electric-field profiles at the anchor and plug regions confines both ions and electrons. The optimization of the magnetic field configuration is straightforward and is accomplished by appropriately arranging the coils. However, it is not so simple to form and control the electric field (potential) due to the properties of the local plasma.

The mirror machines presently in operation are GAMMA 10, HIEI, and the QT-Upgrade in Japan, AMBAL-M and GDT in Russia, and HANBIT in Korea. GAMMA 10 is one of largest devices among them, and it has produced numerous mirror research world-class results. This tandem mirror magnetic fusion research, which employs an electrostatic potential, is focused on experimental programs aimed at the suppression of the end loss and the formation of the electrostatic potential. This research is presently at the proof of principle (PoP) stage.

In GAMMA 10, an ion temperature of 10 keV was achieved at the central plug region with ICRF heating. Neutron production due to a thermonuclear reaction was also observed. An electrostatic potential of about 1 kV is formed at the central plug. However, the electron temperature is about 100 eV and the plasma density is rather low, around $2 \times 10^{18}\ m^{-3}$. Since the necessary electrostatic potential for potential plugging is 5 to 10 times larger than the electron temperature, it is necessary to form a potential of more than 100 kV for the fusion reactor.

The main characteristics of a linear fusion reactor are the simplicity of the magnetic field configuration, and the simplicity of the machine construction, installation, and decommissioning. The mirror machine is equipped with a stationary magnetic field coil, and it is therefore easy to operate in a steady state. Another technical possibility pointed out for the mirror system is its application as a strong neutron source. Investigation of high-beta fusion plasmas using advanced fuel is also a suggested application.

The significance of the radial electric field is identified as a key issue for the improvement of plasma confinement in recent tokamak research, whereas the plasma potential has been considered very important and of major concern in mirror research for many years. The magnetic field line in the tokamak divertor, which is one of the main concerns in ITER, has an open field structure. This configuration is very similar to the field line of the mirror device and has common issues for investigation in plasma physics in the direction of magnetic field (say, potential formation and the plasma flow).

In the construction and research of the mirror device, the heating apparatuses, such as the neutral beam injector and gyrotrons, as well as the advanced diagnostic instruments, such as the microwave and soft X-ray detector, have been developed. The technology involved in the development of the heating and diagnostic instruments can be broadly applied to the other areas of fusion experiments, such as tokamak experiments.

4.3.2.4 Reversed Magnetic Field Pinch System [4.3.2-5]

Reversed magnetic field pinch (RFP: Reversed Field Pinch) is a toroidal confinement system, where the direction of toroidal field in the central region of the plasma is opposite that at the edge. It was experimentally discovered in the ZETA device in the1960s that such a reversed magnetic field configuration was stable. After the theoretical verification of its stability, many RFP devices were constructed and operated in the world. Intensive research on the characteristics of RFP plasma and the improvement of plasma performance were pursued in various devices. Figure 4.3.2.4-1 shows the presently operational large RFP devices (TPE-RX in Japan, MST in the United States, RFX in Europe), small devices in Japan and an RFP based reference reactor design. The characteristic device parameters of the large machines are such that the plasma major radius is 1.5 - 2.0 meters and the plasma current is about 1 MA.

RFP devices are similar to tokamak devices in the sense that both devices have axi-symmetric-toroidal geometry with toroidal electric current. However, RFP devices can carry a toroidal current more than 10 times larger than the tokamak devices. Should an RFP-type fusion reactor be constructed with a toroidal current similar to that of a tokamak-type reactor (Ip = 10 – 20 MA), the required toroidal field strength can be reduced to 10% of the tokamak-type reactor, which means that a toroidal coil made of normal conductors can be used in the reactor. It also means that the construction cost of the fusion device would be greatly reduced due to the simplification and size reduction of the system.

The ultra low q (ULQ) configuration was discovered experimentally in the REPUTE-1 device (Japan), which has a plasma configuration similar to a tokamak. In spite of the fact that theoretical studies performed in the past had identified no stable equilibrium in this configuration, the Meta-stable configuration was found by controlling the safety factor profile in the plasma. The optimized safety factor profile is similar to that of reversed magnetic shear in a tokamak. Therefore, discovery of this stable configuration pioneered the path for the steady-state operation of the tokamak device.

The RFP plasma has a characteristic feature that both energy confinement time and density increase with plasma current (τ_E - $Ip^{1.5}$, n_e - Ip). Figure 4.3.2.4-2 shows the product of the density and confinement time as a function of plasma current. The figure suggests the significance of proceeding with extended research with a larger toroidal current. Typical plasma parameters of the RFP devices are such that electron and ion temperatures are several hundred eV, and the beta value is about 5-20%. These values are 1 to 10% of that for tokamak devices with a similar size. The longest confinement time achieved to date in the RFP devices is τ_E ~10 msec in MST. It is therefore necessary to pursue further the research related to the plasma confinement mechanism and to improve the confinement performance.

The equilibrium and stability of RFP plasma can be explained by the MHD relaxation theory developed by J.B. Taylor. According to this theory, the plasma is relaxed to an equilibrium and stable state as a whole under the condition that the helicity of the whole system is conserved. The theory is consistent with the experimental results. This self-organization phenomenon is induced by the rearrangement of the magnetic field lines (magnetic reconnection). The theory of magnetic reconnection was investigated systematically by H.P. Furth and his colleagues, and the results obtained are being applied to research of the tearing mode that plays a significant role in current collapse and the limitation of the attainable beta in a tokamak. It was also found that the so-called Dynamo Effect, the mechanism by which the plasma itself produces the magnetic field, exists in the relaxation process. This process is pervasively observed in various physical fields other than fusion research, such as in geomagnetic field production and in solar flares. The RFP plasma research plays an important role in the sense that only RFP has the self-organization phenomenon that can be controlled in the laboratory and is able to have detailed measurements made on it.

4.3.2.5 Spherical torus [4.3.2-6]

The spherical torus (ST) can be considered as either an ultra-low aspect ratio tokamak (aspect ratio A (= R / a) of regular tokamak is greater than 2.5 whereas the ratio is less than 1.6 for ST), or a compact torus with a weak toroidal field. The structure of the magnetic field line in an ST is shown in Fig. 4.3.2.5-1. The ST has the possibility of achieving a substantially high beta value, as its magnetic field lines wind around the so-called good curvature region. Fig. 4.3.2.5-2 shows the beta value attained experimentally in the START device in Europe. The START device achieved the very high beta value of 40%, in contrast to the currently operated tokamak devices with ≤ 10%.

At present, experiments are performed in MAST (in the United Kingdom) and NSTX (in the United States). These devices are a size larger than that of the START device. The objective of these devices is proof of principle research of ST plasmas. The device size of MAST and NSTX is major radius / minor radius = 0.7 m / 0.5 m (MAST), 0.85 m / 0.68 m (NSTX), and the target plasma current is 1-2 MA. In Japan, small ST devices (major radius R = 0.3-0.4 m), such as TS-3 / 4, TST-M / 2, and HIST, are under construction or in operation. The objectives of these devices are to perform the advanced plasma research, such as the ST generation by plasma merging and RF-heating as well as helicity injection.

Although the compactness of the spherical torus is an attractive feature for a nuclear fusion reactor (major radius will be less than 3 m), it also poses engineering design difficulties, such as the design of the central conductor, which produces the toroidal magnetic field, the high neutron wall load, and the divertor heat handling capability. The ST reactor can employ a normal conductor instead of a superconductor for the central solenoid. The degradation of the copper conductor installed in the narrow region in the torus center, due to the neutron radiation, is indeed an issue of concern. In addition, the ST intrinsically has the characteristic of having a high neutron wall load due to the small size of the device. The application of ST as a volume neutron source to examine the functions of the blanket module is being discussed.

4.3.2.6 Compact torus [4.3.2-7]

Spheromac and field reversal configuration (FRC) are called compact torus (CT) as a whole. In a spheromac, the poloidal and toroidal magnetic fields are generated by the application of toroidal and poloidal plasma current in spherical plasma, respectively. In the1980s, medium size spheromac devices, such as LASL and S-1, were constructed and operated in the US. These devices successfully produced plasma with an electron temperature of several hundred eV, but the program was terminated because it was difficult to stably sustaining the spheromac configuration. In Japan, small size spheromac devices were constructed and operated in the past. A recent effort in this field has been focused on the merging of toroidal plasmas and studies of magnetic reconnection. Also in the US, it has been decided to construct a medium size spheromac device in the Livermore National Laboratory.

FRC is a magnetic configuration in which the plasma is confined only by the poloidal field produced by the toroidal current applied in the plasma. The so-called theta-pinch scheme is generally employed to produce the plasma current in the toroidal direction. In principle, the confinement of the very-high-beta plasma is possible in FRC. FIX, TS-3 / 4 (both in Japan), LSX (the United States) are typical FRC devices. The important parameter for FRC is the S-value ($S = a / \rho_i$), i. e., the ratio of the plasma minor radius to the ion Larmor radius. The value of S = 2-3 has been obtained in the present experiment, but S has to be around 40 for the reactor. The evaluation of plasma properties under such S-value is necessary for application to reactor plasma. It has been demonstrated FRC can be produced and then transferred in the axial direction. Therefore, it is possible in FRC reactors to spatially separate the formation section and burning section.

Although the CT relevant research is still at the stage of the proof of principle, CT has the potential capability of being realized for the ultra-high-beta fusion reactor, which is advantageous due to the simplification of this type of fusion reactor. An advanced fuel fusion reactor concept with CT, such as the D^3He fusion reactor, has also been proposed.

4.3.2.7 Internal conductor torus machine [4.3.2-8]

An internal conductor torus machine has been highlighted as a device that seeks the confinement of super-high-beta plasmas. In the RFP / ULQ plasmas, the electron flow is relaxed to the force-free configuration by the MHD activity, and in principle they are low beta plasmas. On the other hand, according to the theoretical prediction based on the MHD relaxation in a two-fluid plasma model, in which the ion flow is also taken into account, the squared sum of the plasma beta and the flow velocity normalized by the Alfvén velocity is constant and perpendicular to the magnetic surface. This is considered as a generalization of the Bernoulli principle, in which it is understood that the sum of the static pressure (plasma pressure) and the dynamic pressure (the pressure of the plasma flow) is constant. Therefore, it is suggested that inducing high-speed plasma flow in the edge plasma region can produce the equilibrium configuration that accommodates the high-beta plasma in the central plasma.

An internal conductor torus machine is suitable for experimentally investigating the new MHD relaxation, which takes advantage of the two-fluid effect. The poloidal field is generated by the circular current in the internal conductor, and the plasma is rotated in the toroidal direction at high speed by the radial electric field in the plasma. Here, a large radial electric field can be induced by a slight non-neutralization (about 10^{-4}%) of the plasma. As the poloidal field decreases with the distance from the internal conductor, the ExB drift velocity becomes large, and the high-beta plasma can be confined near the internal conductor.

The Proto-RT device (Japan) is constructed and operated to study the possibility of super-high-beta plasma confinement with the internal conductor configuration. It has been successful so far, producing non-neutral plasma by the injection of electrons, which formed an electrostatic potential of about 500 V. However, the electric current inlet and the support structures cross the magnetic surface, as the internal conductor is a copper coil. Therefore, an internal conductor device that employs magnetic levitation of an internal conductor made with superconducting materials is being designed.

The magnetic levitation internal conductor device was constructed and operated from the late 1960s to early 1970s in the Levitron and Spherator. In these devices, the magnetic well effect or stabilization by the magnetic shear was expected, and relatively good confinement characteristics were documented. Nonetheless, they were shut down, seems promising. Since internal conductor was regarded as unfavorable also from a reactor-engineering viewpoint and tokamak seems promising. However, the super-high-beta plasma confinement scheme, which is based on a different concept, was recently proposed in the US. A magnetic levitation internal conductor device LDX is now under construction there. In the LDX device, super-high-beta plasma is confined by the substantial magnetic compression effect of the dipole magnetic field line.

Plasma flow plays an important role in the H mode of tokamak plasma, and it has been noted that the two-fluid MHD relaxation phenomena mentioned above are related to H mode physics. In addition, a recent inter-planet satellite observation indicates that super-high-beta plasma (100% and more of the beta value) exists on Jupiter. It is possible that the confinement mechanism of tokamak super-high-beta plasma is correlated with this observation.

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[4.3.2-8] Zensho Yoshida, "Research of the Advanced Confinement System at the University- Nuclear

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Table 4.3.2.1-1 Specification and the present status of LHD

	Target Phase I (II)	Operation (1^{st}- 3^{rd} cycle exp.)
Major radius	3.9 m	←
Plasma radius	0.5-0.65 m	←
Plasma volume	20-30m^3	←
ℓ/m, ι (0)/ ι (a)	2/10, < 0.5/1	←
B_0 / B_{max}	3/6.6 T (4/9.2 T)	2.75 T
LHe temp.	4.4 K (1.8 K)	←
Magnetic energy	0.9 GJ (1.6 GJ)	
Plasma duration	10 s – CW	0.5-35 s
Heating power		
ECRH	10MW	0.9 MW (84 GHz,168 GHz)
NBI	180 keV/15 MW (20 MW)	150 keV/4.2 MW
ICRF	3 MW (12 MW)	1.0 MW

Table 4.3.2.1-2 Attained parameters up to 1999 by LHD

High electron temperature (ECH+NBI)	Electron temperature	4.4 keV
	Ion temperature	2.7 keV
	Confinement time	0.06 sec
	Electron density	0.53 x 10^{19} m^{-3}
	Absorbed heating power	1.8 MW
High ion temperature (NBI+ICRF)	Electron temperature	3.3 keV
	Ion temperature	3.5 keV
	Confinement time	0.09 sec
	Electron density	1.0 x 10^{19} m^{-3}
	Absorbed heating power	3.9 MW
High confinement (NBI)	Confinement time	0.3 sec
	Electron temperature	1.1 keV
	Electron density	6.5 x 10^{19} m^{-3}
	Absorbed heating power	2.0 MW
Maximum stored energy		0.88 MJ
Maximum β-value	Volume average β-value	2.4% @1.3T
Maximum density		1.1 x 10^{20} m^{-3}
Long pulse operation	NBI	80 sec (0.6 MW, 1.5x10^{19} m^{-3}, Ti ~ 1.6 keV)
	ICRF	70 sec (0.85 MW, 1.0x10^{19} m^{-3}, Ti ~ 2.0 keV)

Table 4.3.2.2-1 Most important issues in inertial fusion reactor technology development

Issues	Contents	Common aspect to MCF/differences
Driver interface	Neutron streaming Thermal protection of the final mirror	Neutron streaming to the NBI apparatus
Plasmas facing materials (liquid metal flow)	Width of the liquid metal curtain ≥ 20 cm. Difficult protection of the upper and lower solid structure	Target (width=2.5 cm, velocity=20 m/s) cooling by the liquid lithium, as planned in IFMIF
Plasma facing materials (solid wall)	Ultra short thermal shock in solid first wall	Shorter pulse than the disruption and many repetitions
Fast vacuum recovery after the burning	Fast (1/3 sec) exhaust of the burning products, evaporated liquid and solid surface	Fast exhaust of the liquid and metallic materials
Tritium fuel processing	Refinement and isotope separation from the debris and exhaust gas	Recovery of tritium from the debris
Collection of blanket tritium in the blanket	Tritium recovery from the liquid and solid lithium compound	There are common parts to the tritium collection from MCF blanket
Fuel target production and injection	Production of highly spherical DT fuel and highly precise injection control	Number of common features to the MCF pellet injection, however, more precision required in ICF
Vacuum chamber	Many ports are necessary for the laser injection	Application of the double wall vacuum chamber may be possible in ICF
Remote control maintenance technique	Necessary for the maintenance of the components in the reactor chamber and other auxiliary facilities	Maintenance of components which do not interact with liquid may be unnecessary? Possible application of the remote maintenance technology for ITER
Low activation Materials	Lower activation favored from the view-point of environmental safety	Many common issues, such as SiC, although the required performance is different
Counter measure to earthquakes	Necessary for the integrity of beam optics	Developed technologies for ITER are applicable
Diagnostics coolant accident	Diagnostics of ultra-high density plasmas with ultra-short pulse technique	Similar instrumentation and radiation monitoring technology
Safety system	Coolant off-normal event tritium release	The tritium confinement technology is common

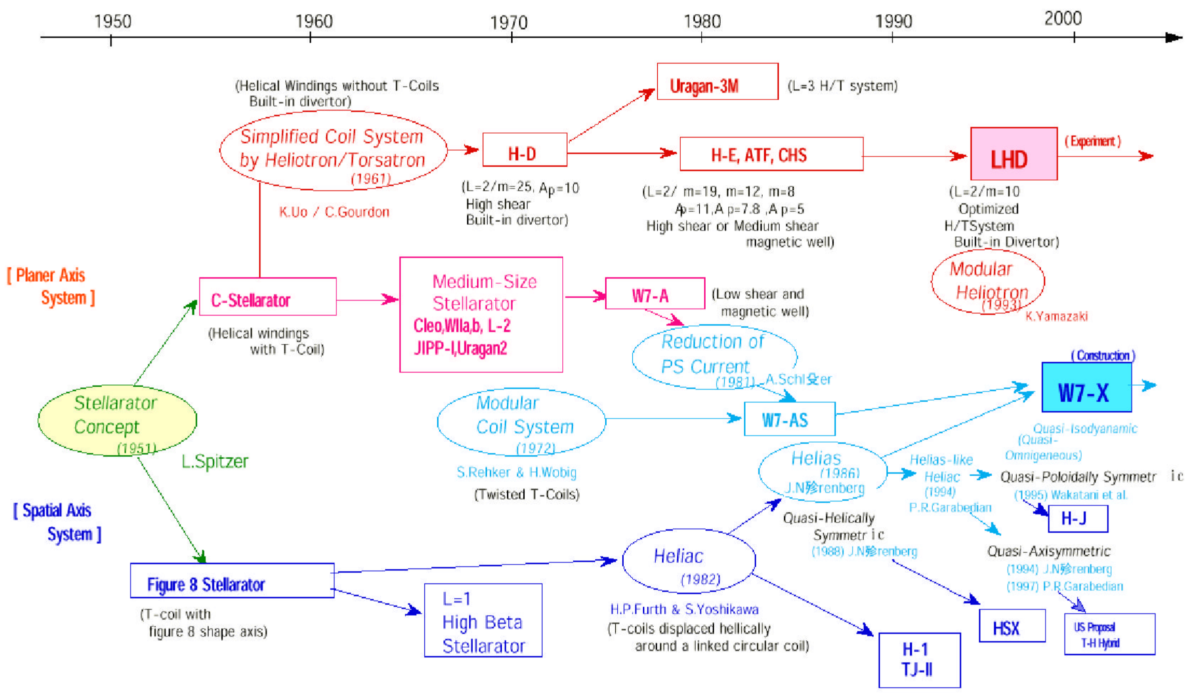

Fig. 4.3.2.1-1 Development of helical confinement concept

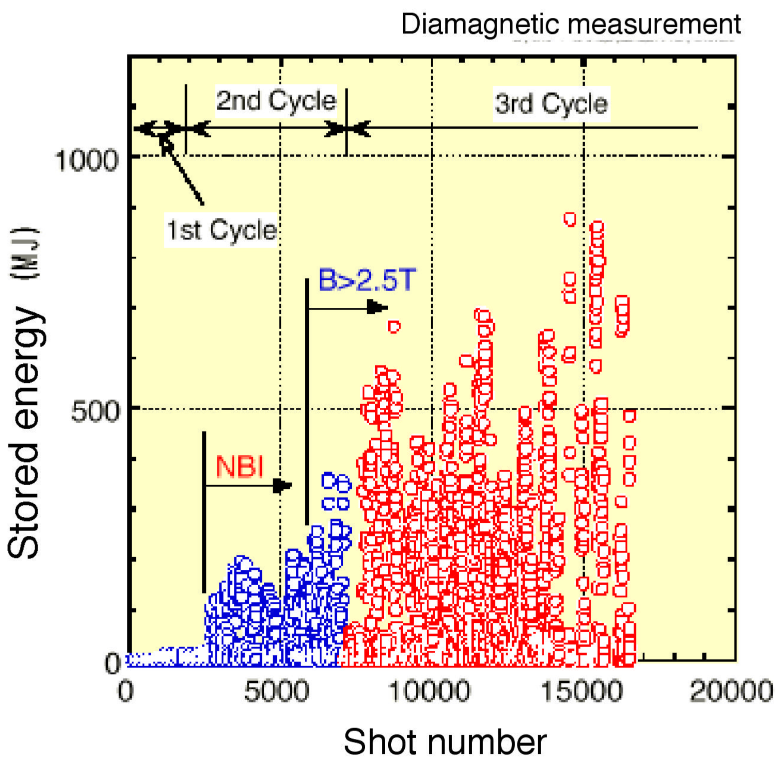

Fig. 4.3.2.1-2 Improvement in plasma stored energy with succeeding discharges of the LHD

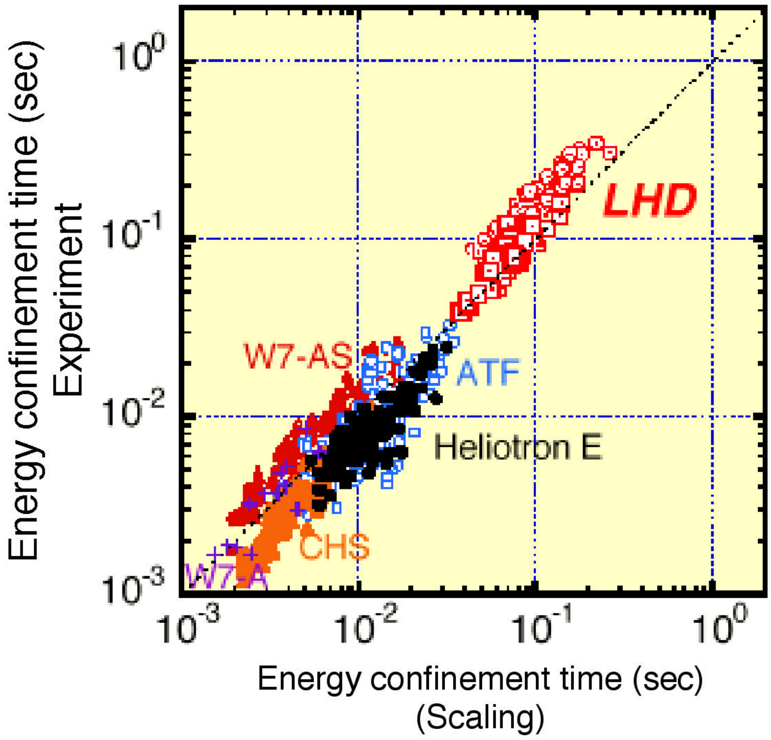

Fig. 4.3.2.1-3 Comparison of energy confinement time between empirical scaling from a medium-size helical system and LHD

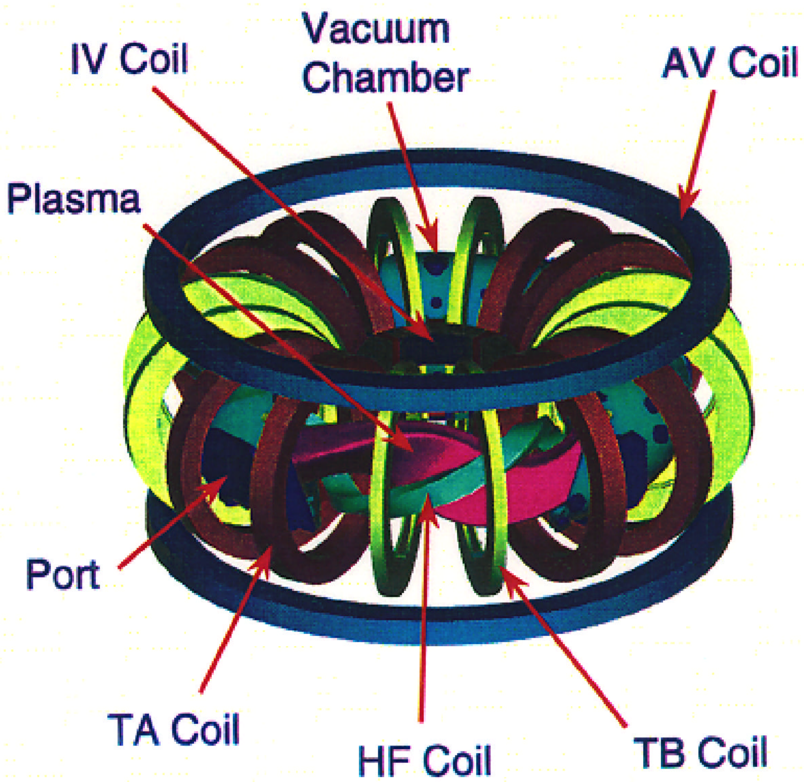

Fig. 4.3.2.1-4 Coil system of Heliotron-J (HF Coil: helical field coil, TA Coil/TB Coil: a two-set toroidal field coil, IV Coil/AV Coil: poloidal field coil)

Difference between inertial fusion and magnetic fusion

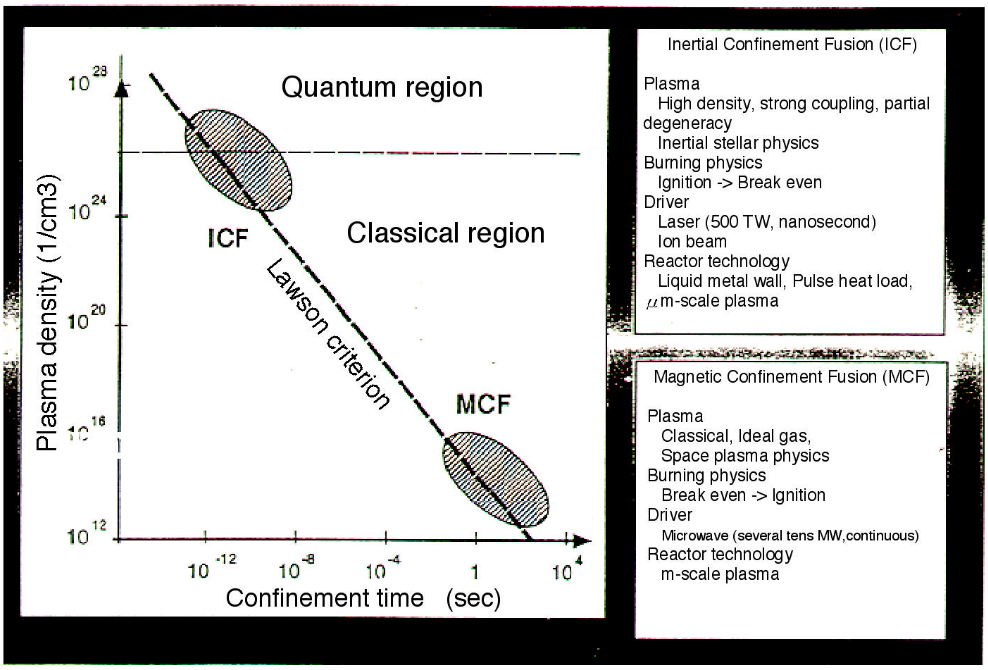

Fig. 4.3.2.2-1 Differences of parameter ranges between the magnetic fusion and inertial fusion confinement schemes

Approach to fusion ignition

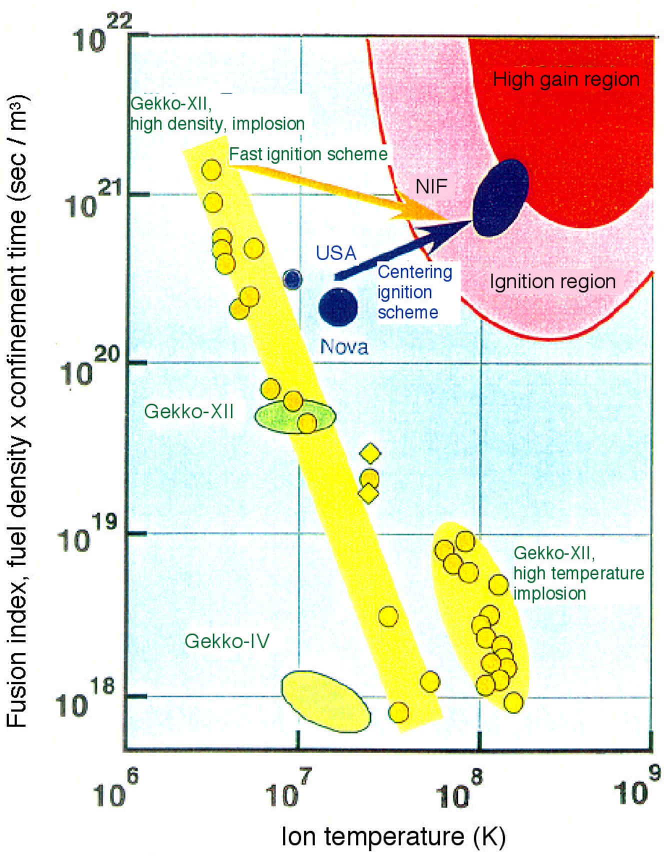

Fig. 4.3.2.2-2 Direction of the inertial confinement fusion research towards self-ignition

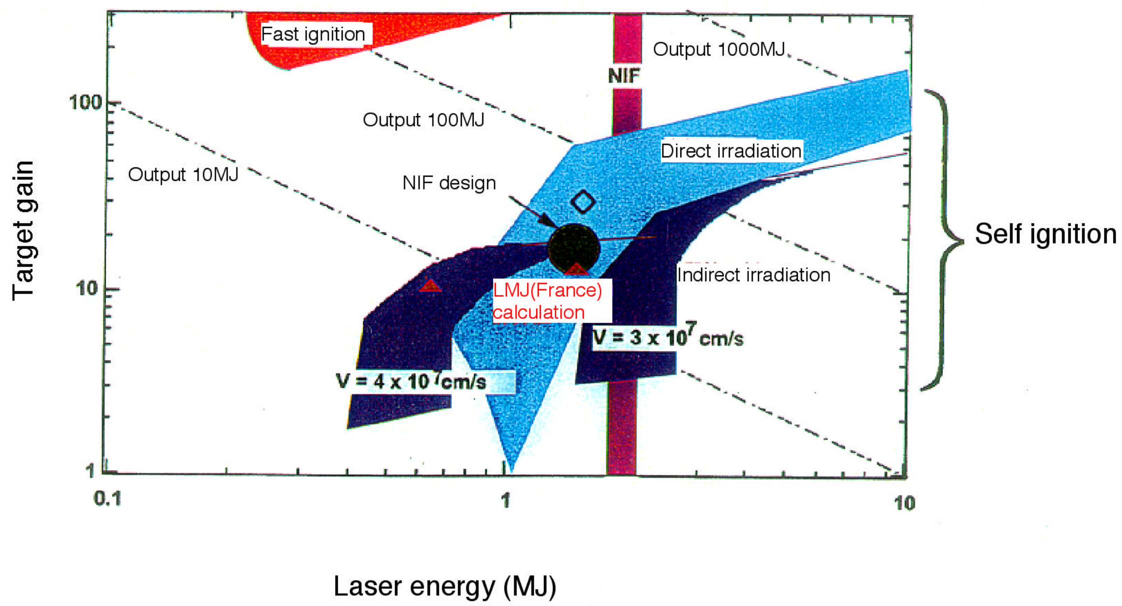

Fig. 4.3.2.2-3 Laser energy and target gain required in various scenarios in the inertial fusion scheme

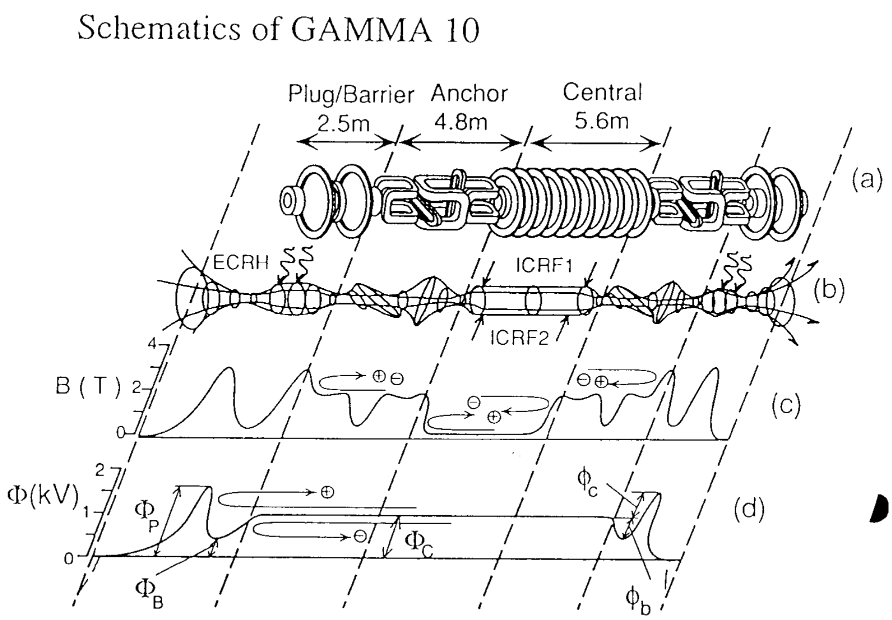

Fig. 4.3.2.3-1 The machine configuration of Gamma 10 tandem mirror system and axial distribution of magnetic field and electrostatic potential

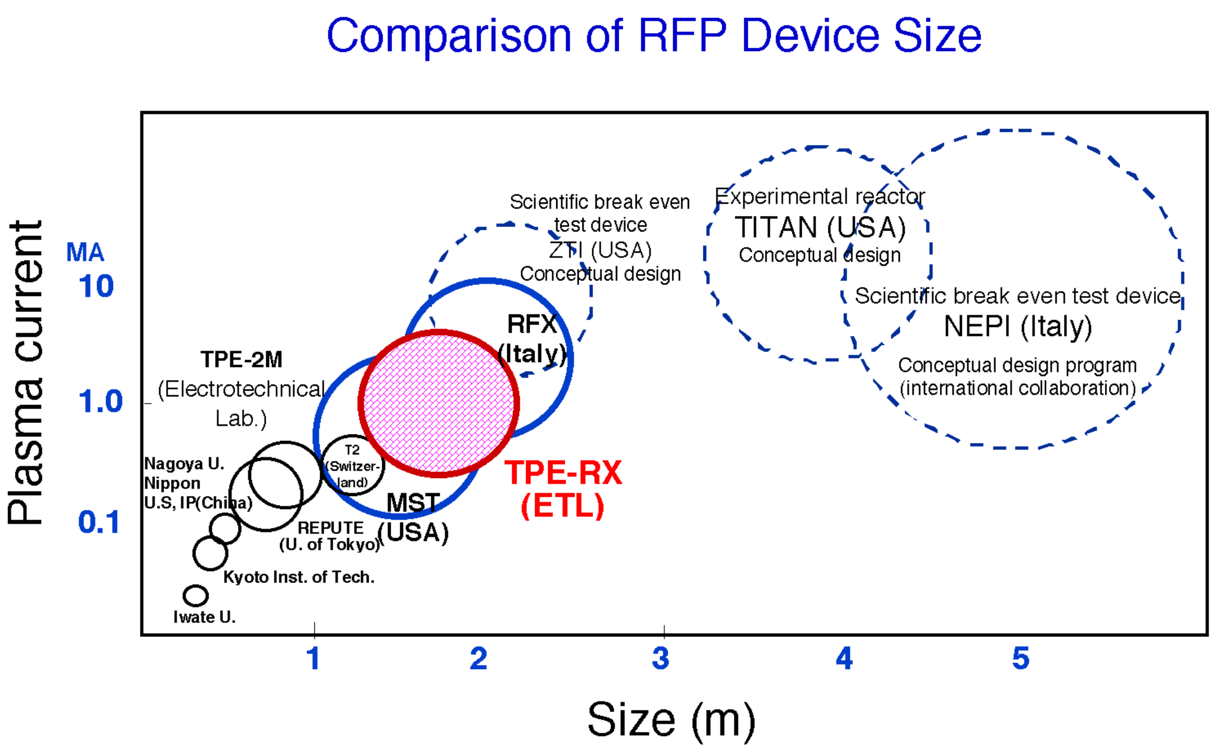

Fig. 4.3.2.4-1 Comparison of the size and plasma current of the world's RFP devices, including those under conceptual design

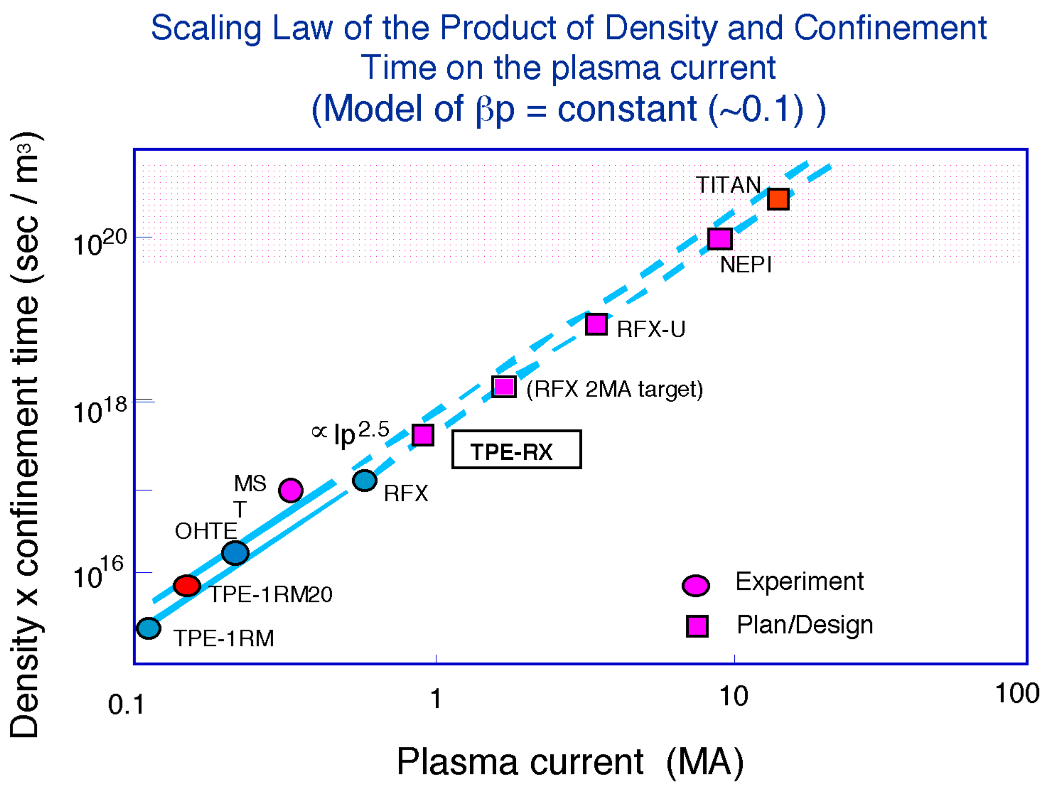

Fig. 4.3.2.4-2 Scaling law of the product of density and confinement time as a function of plasma current

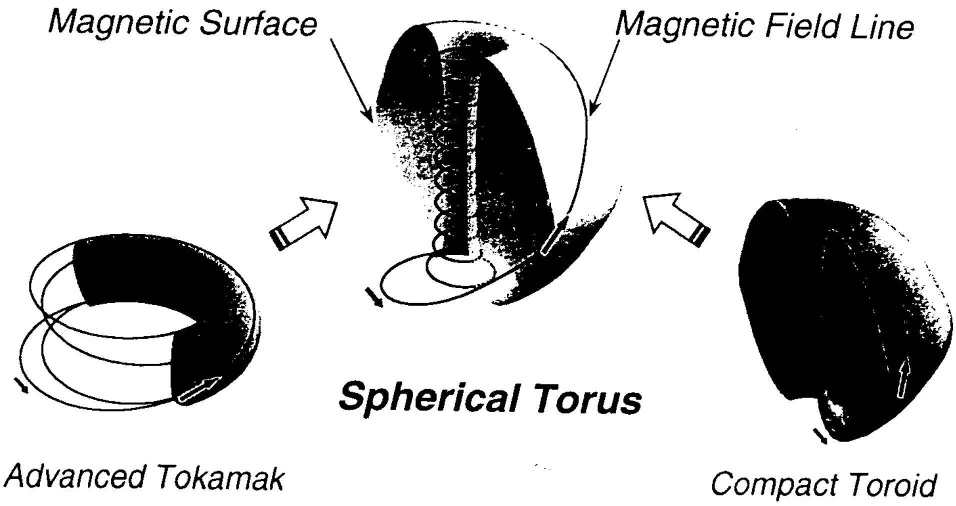

Fig. 4.3.2.5-1 Comparison of the magnetic field structures of ST, tokamak, and CT

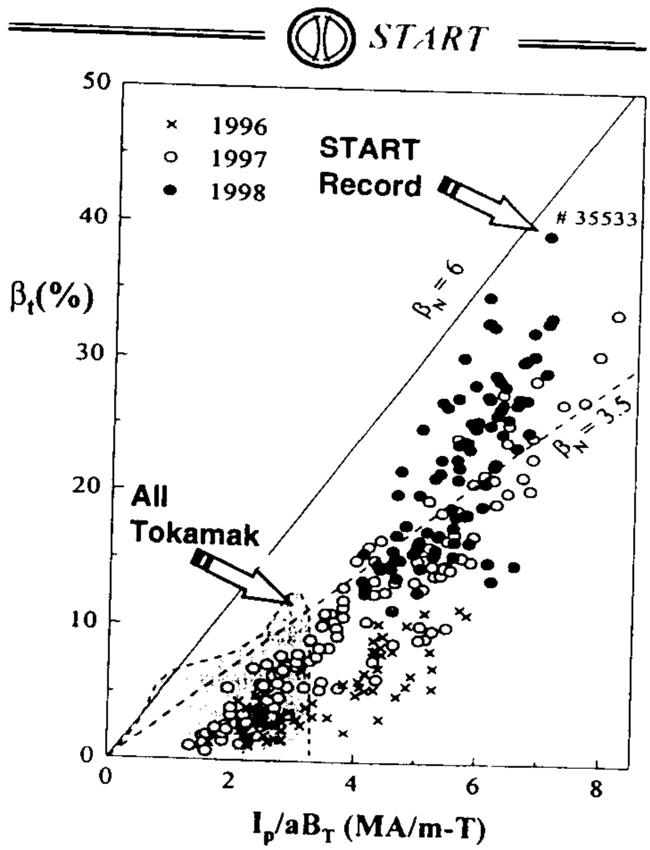

Fig. 4.3.2.5-2 Comparison of the experimentally achieved beta-values in START (ST) and in present tokamaks

4.3.3 Basic research on materials and fusion engineering

4.3.3.1 Introduction

The strategy of material development toward the experimental reactor ITER and the demonstration reactor DEMO is described in Section 3.3.2 and the strategy of the fusion engineering development for fusion reactors is presented in Section 3.2. In this section, the academic and basic aspects of fusion engineering research are described. Fusion engineering research consists of various engineering fields, including the in-vessel material engineering, structural material engineering, blanket engineering, tritium science and engineering, superconducting magnet engineering, electromagnetic structural engineering, tritium biological engineering, thermo-structural engineering, reactor design engineering, system safety engineering, neutronics engineering, and engineering related to the inertia fusion. Basic studies on the "Promotion of the Comprehensive Systematization of the Fusion Research" and on the "General Research on Sophistication and Information Network of Fusion Science" were conducted under the budget of the Ministry of Education from FY1991 to FY1995. The network between the National Institute of Fusion Science (NIFS) and universities was prepared under this activity. Also proposed was the organization of a new framework consisting of the fusion system safety group, the material and fuel group, the electromagnetics group, and the inertia fusion group. In these groups, collaborative research and information exchange has been carried out to contribute to the fusion engineering research. In the material field, for example, the research network has been reinforced via the collaborative research involving fission reactors, and specific research tasks were defined, common samples were prepared, and a database was compiled as joint work.

The research tasks of fusion engineering have also been reviewed under the activity of Science Council of Japan. The main issues of fusion engineering and the preparation of a research center were proposed in the 16th Fusion Research Committee in the report titled "Preparation of the Collaboration Research Center for the Fusion Reactor Engineering" that was published in July 1996. The report was based on a study conducted in the Subcommittee of Fusion Engineering. In the activity of 17th Nuclear Science Research Committee / Fusion Research Specialist Committee, "Restructuring and Systematization of Fusion Engineering" has been studied in the Subcommittee of Fusion Engineering. Recommendations on the basic study of fusion engineering were produced as an external report of the Science Council of Japan.

Taking advantage of the characteristics of universities, various basic research on fusion engineering has been conducted. These research activities support the basis of fusion engineering for the experimental reactor, ITER, and contribute to the continuous training and retention of human resources. In the following sections, the major issues that were confronted in the research network are presented. The research activities related to materials of fusion reactor are described by introducing some examples.

Major issues of fusion engineering in various fields are presented in Reference 4.3.3-1.

4.3.3.2 Research tasks of fusion engineering

(1) In vessel material engineering

- Assessment of the characteristics, including irradiation, of high-Z materials
- Methods to decrease the heat load to the divertor
- Research on the removal of helium ash and on the particle balance
- Assessment of the plasma facing components using LHD and TRIAM-1M and the development of methods for conditioning and coating the first wall

(2) Structural material engineering

- Assessment and development of advanced low-activation materials
- Improvement of irradiation tests using fission reactors
- Improvement of simulation models of material behavior
- Development of a simulation technique using molecular dynamics, etc.
- Basic study on the shape effects to establish a small sample test method
- Study of the irradiation effects under the complex operational environment
- Basic study on the irradiation defects by ion- and electron irradiation

(3) Blanket engineering

- Assessment and preparation of the physical, chemical, and mechanical database for breeding materials
- Study the transport mechanism of tritium from blanket
- Study the irradiation effects of solid breeding materials and neutron multiplier material
- Evaluate ceramic coating for the liquid breeding materials
- Examination of the practicality of molten salt (F/Li/Be) as the blanket system

(4) Tritium science and engineering

- Development of the tritium permeation barrier and improvement of tritium confinement
- Development of the tritium permeation materials and processing technology
- Development of the technology to remove the tritium contamination and the technology of tritium waste treatment
- Improvement of technologies to purify, separate, and recover tritium
- Development of dry processing technology

(5) Super-conducting magnet engineering
- Research on high-function superconducting materials, such as high-temperature superconducting materials, and application of those materials
- Improvement and stabilization of the electromagnetic characteristics of superconducting materials
- Safety and protection of magnets
- Study of required mechanical conditions of large-scale superconducting magnet structure
- Cool down and heat transfer of superconducting magnets
- Radiation resistant materials for the structure of superconducting magnets

(6) Electromagnetic structural engineering
- Improvement of reliability of instruments against electromagnetic destructive phenomenon
- Assessment of structural reliability by eddy current analysis and electromagnetic analysis
- Improvement of the electromagnetic nondestructive inspection
- Clarification of the coupled phenomenon of electromagnetic structures and its application to the design

(7) Tritium biological engineering
- Research on the dynamic state of tritium in the dome environment
- Research on the repairing mechanism of radiation damage
- Effects of long-period, low-dose rate exposure of tritium
- Basic study on the effects of environmental tritium on human beings

(8) Thermo-structural engineering
- Feasibility study on the DEMO reactor using advanced cooling methods
  (solid-gas mixture flow, liquid metal flow, liquid metal double layer flow, etc.)
- Development of the advanced high-heat-flux system
- Basic technology for the development of advanced thermal flow system
- Safety of thermal flow and system design

(9) Reactor design engineering
- Innovation of reactor design by introducing new reactor concepts
- Improvement of safety and environmentally-friendly reactor concepts
- Improvement of reliability in fusion engineering by introducing new technologies in various reactor concepts
- Establishment of a universal procedure for the economic assessment of various reactor concepts

(10) System safety engineering
- Establishment of the basic concept on the safety of a fusion facility
- Assessment of the system safety analysis of a fusion facility
- Research and development of the engineering safety equipment

(11) Neutronics engineering
- Updating the nuclear database including 14-MeV neutron data
- Editing the special objective files for fusion
- Preparation and improvement of the computer codes for radiation transfer and nuclear design
- Remote utilization of the codes and computer system
- Development, design, and construction of the neutron irradiation facility
- Experiments on neutronics engineering, such as the measurement of nuclear data, benchmark experiments to evaluate the neutronics design
- Development of the technologies for irradiation experiments and diagnostics of radiation

(12) Engineering of the inertia fusion

- In addition to the tasks described above, identify the unique engineering fields related to inertia fusion engineering
- Development of high-power, high-efficiency, long-life driver, and their mutual interaction
- Development of the manufacturing technology of high-accuracy pellets and the technology to inject them at high speed
- Design of the reactor chamber, such as the liquid metal wall and beam ports
- Safety research on the inertia fusion reactor system

4.3.3.3 Research activity on fusion reactor materials

Among the issues indicated in the previous sections, materials research is one of the common subjects that does not strongly depend on the method of plasma confinement. Further, a relatively long lead-time is required for the development of these materials. Therefore, it is recognized as a subject that needs to be addressed in the early stage of development of the demonstration (DEMO) reactor. In particular, studies of irradiation effects on structural materials have been conducted based on fundamental studies at Japanese universities to understand the mechanisms of the phenomena. The fundamental aspect of the mechanistic study is one of the fields in which universities have superior experience. These research activities had been promoted by Science Research Funds for Fusion Science Studies from the Ministry of Education. University researchers successfully gained fundamental understandings of the phenomena and provided remarkable results. After this initial period, a joint project among the universities was performed based on Japan-US collaboration on fusion with the linkage to the joint research using reactor irradiation research facilities in the Institute of Metals of the Tohoku University at Oarai and researches at various university chairs.

The RTNS-II project, which is a Japan-US collaboration program aimed at performing an irradiation experiment using the D-T fusion neutron source of RTNS-II, was started as the first phase of a research project for the irradiation effects test of fusion reactor materials. The irradiation experiment on pure metals and simple model alloys at temperatures from liquid helium temperature to higher temperatures have been conducted successfully to provide fundamental understandings on the mechanisms related to the development of damage microstructures and the fission-fusion neutron correlation of irradiation hardening. The compiling of knowledge on the irradiation effects on ceramics is a major achievement of this project.

The second phase of the research for fusion reactor materials of the Ministry of Education FFTF/MOTA project was carried out using the Fast Flux Testing Facility (FFTF) to irradiate various structural materials and ceramics at elevated temperatures. This research effort was focused on the fundamental understanding of the relation between the development of microstructure damage and changes of the mechanical properties including the development of a method to simulate the effect of the production of He atoms under the irradiation by fusion neutrons. The major achievements are (1) the understanding of swelling and the change of the mechanical properties by alloying elements and eduction for vanadium alloys and various steels, (2) development of the dynamic tritium trick method to introduce He atoms into the alloys and (3) the fundamental understanding of the irradiation effects of ceramic composite materials and bonded metal-ceramics materials (Reference [4.3.3-2]).

The JUPITER project has been conducted using the High Flux Isotope Reactor (HFIR) to acquire knowledge on the cumulative and dynamic effects of irradiation on mechanical properties and the transport phenomena that are thought to have a significant effect during service in a fusion reactor. In situ measuring experiments of electro- and thermal conductivity during irradiation were also performed. The analysis on the dynamic effect of temperature change during the irradiation has also been carried out for typical low activation structural material (Reference [4.3.3-3]).

The research on the irradiation effect of materials is expected to be completed in the third phase of the project. In the next phase, integrated research on "materials and systems" issues will be studied aiming at advanced blanket by the joint works with other fusion technology areas.

References

[4.3.3-1] Report for the Ministry of Education research funds (A), “For the advanced fusion researches and the networking,” Final report for the Reactor Engineering Committee, May 1996 (in Japanese).

[4.3.3-2] Report for the FFTF/MOTA collaboration, “Current Status of the Fusion Reactor Materials,” NIFS, May 1995 (in Japanese).

[4.3.3-3] Report for the FFTF/MOTA collaboration, “Dynamic and Synergistic Effect on the Fusion Reactor Materials,” NIFS, December 1997 (in Japanese).

4.4 Education and cooperative structure

Very close cooperation among the industry, government, and universities is important in coordinating a large-scale and long-term R&D program in a balanced way, such as is the case of fusion research. In particular, it is indispensable to (1) secure human resources and provide the necessary education, (2) coordinate the cooperative structure between the universities and research institutions, and (3) ask for the support of industry, which can substantially enhance the ultimate capabilities of the program.

It is also important to explore the human resources aspect, facilitating the research and educational programs at universities and research laboratories. In addition, it is necessary for successful ITER development to maintain the scientific research standards and engineering R&D methods for a substantial period of time, to continuously provide research topics to sustain an adequate number of scientists in fusion research, and to pass down the acquired knowledge in physics and technology to future generations.

4.4.1 Human resources required for the ITER program

The specialists of various fields are required to proceed with the ITER program. The proportion of the human resources required in individual fields is first analyzed during the construction and operation phase, based on the constitution of the ITER JCT members (Table 4.4.1-1) [4.4.1-1]. Here, it is assumed that a similar number of personnel are directly or indirectly involved in the ITER program from each Party. It does not mean that a contribution of 300 people is adequate for the construction of ITER. The allocated number of the specialists in the various fields is to be predicted, and the number of total core staff is assumed to be 300 (in the organization including the supervisors and labors on the production lines).

Table 4.4.1-1 Predicted distribution of necessary staff for ITER classified by their original specialty fields

Specialty Fields	Present (JCT)	Construction Phase	Experimental Stage
Physics (Plasma)	14 (14%)	20 (7%)	90 (30%)
Physics (Engineering)	5 (5%)	15 (5%)	15 (5%)
Mechanical engineering	23 (23%)	60 (20%)	30 (10%)
Electrical engineering	17 (17%)	60 (20%)	30 (10%)
Nuclear Power engineering	19 (19%)	60 (20%)	40 (13%)
Chemistry	3 (3%)	10 (3%)	8 (3%)
Information and Instrumentation engineering	8 (8%)	30 (10%)	60 (20%)
Architecture and civil engineering	3 (3%)	25 (8%)	3 (1%)
Administration & Affairs	7 (7%)	20 (7%)	22 (7%)
Total	99 Persons	300 Persons	300 Persons

Note: Specialists of the material field are not shown explicitly, as they are considered to be involved in mechanical, electrical, and nuclear engineering fields.

(1) Human resources required for reactor engineering

The group leaders, managers, and the supporting staff are supposed to be well acquainted with the fusion facilities and experienced with the development, construction, and physics operation of the fusion devices in their own Party, and the portion of this core staff would be larger than 20 to 30% of all the personnel involved in the engineering field. Other engineering staff members having relevant expertise, specialized skills, and technical experience in the fusion research and development are not necessary. The supply of laborers with the required expertise is largely left to the industry.

In the early phases of the experiment, 50% of the core engineering staff, who have experienced the construction and initial operation, would have to remain in the program. However, this percentage may be reduced to 30% with the progress of experiment.

In addition to the JCT core staff, capable engineers are necessary from industry. However, they have been fostered in fusion R&D activities performed to date, including the ITER program. In the case of the Japanese contribution to ITER, more than 60% of the personnel was sent to JCT from industry. Moreover, approximately 40 persons are newly involved in the engineering design of the fusion device every year. Inclusion of those who actually worked on the production of components used in the fusion device would further increase the total number of engineers. Therefore, it is considered that a large number of engineers are available to readily participate in some part of the ITER construction work, provided that they could be continuously engaged in relevant technology areas. In this case, it is not necessary for the most of the engineers to be experienced with fusion related engineering, it is adequate if they have specialized knowledge and experience in designing and manufacturing the advanced products in their own field of mechanical, electrical, and nuclear engineering. In this sense, the securing the human resources in industry is possible, should the present schedule for the start of construction be maintained.

(2) Human resources required for burning plasma research

Those who would be involved in burning plasma research are required to be acquainted with plasma science and capability of coordinating the experiment activities and analyzing experimental results. They should be selected from each Party's experienced physicists.

The human resources necessary for the construction of ITER can be sought for among the personnel or organization involved in the current activities related to fusion plasma research and engineering. Finding these personnel would not be a serious issue.

However, a key issue is whether the human resources secured worldwide would continue to coordinate the physics operation, plasma control, and data analysis in the long term, 10 years or more after the first plasma is produced in ITER. The plasma anticipated in ITER is indeed much larger than that of the present large tokamaks. Furthermore, it is necessary to explore the unprecedented field of full-scale nuclear burning and the high-efficiency non-inductive current drive. Therefore, the fostering of scientists who have the profound knowledge and comprehensive understanding in plasma physics as well as device technologies is indispensable. In general, scientists in the field of fusion research have been mainly trained in research institutions by being involved in the advanced science related to the large devices, after having been educated at universities in basic science and experienced with proof of principle experiments with small- or medium-scale devices. The advancement of the scientific research with broad research bases as well as the establishment of procedures to secure capable physicists for the experiment is as important as achieving the ultimate objectives in a large-scale program, such as ITER.

In addition, it is also important to proceed with the advanced research with an ancillary device, one that has machine parameters are as close as possible to ITER, to secure and train the required human resources. The startup phase of the ITER experiment would thereby be well organized, which would expedite the success of achieving the initial objectives. If the research activities at JT-60 and JET are terminated, where prominent scientific results have been produced in the relevant reactor regions, the ITER program would suffer from a worldwide deficiency in human resources immediately after the completion of ITER construction, and it would likely be difficult to smoothly start the experiment.

In the case of Japan, the procedure to foster scientists is well established at present and the scientific standards could be retained in the future as large devices, namely JT-60 and LHD as well as the small- and medium-size devices are in operation at the Japan Atomic Research Institute and universities, and the policy to continue research and to provide education and scientific experience is in place. However, the impact of losing the above facilities is tremendous. It is an urgent issue of concern for our country to determine which experimental devices would be used to cultivate the required human resources, in order to retain a strong initiative for the ITER program.

Furthermore, it is indeed vital for Japan to endeavor to maintain the number of capable scientists needed. Therefore, it is necessary to attract junior scientists to fusion research by activating and maintaining basic research programs at the universities in our country.

4.4.2 The role of universities for securing human resources

Scientists from the next generation will mainly perform the experiment in ITER as well as the construction and operation of the DEMO reactor. Therefore, it is prerequisite to continuously try to secure the personnel, who would be involved in fusion research, as it will take another few decades for the development of the commercial fusion reactors. Continued progress in fusion research is necessary during this entire period. From this viewpoint, the role of universities is significant. The status and future prospects on the capability of providing the environment to encourage students to become involved in the fusion research at universities are described in this subsection, based on the statistics of the past several decades.

4.4.2.1 Overall trend during the 1970s and 80s

A systematic investigation was performed 10 years ago using a questionnaire provided to those who were involved in the plasma science and fusion research fields at that time [4.4.2.1-1,4.4.2.1-2]. Here, the number of students receiving Ph.D. degrees in the previous 20 years was examined and compared with that in the US. Figures 4.4.2.1-1(a) and (b) show the evolution of the number of students in plasma science and fusion research in Japan. The number of undergraduate and graduate students steadily increased, however, the increase of students earning Ph.D. degrees is less prominent. A similar trend was also observed in the number of students earning Ph.D. degrees in the US.

The fraction of Ph.D. students who chose fusion research as their career is shown in Fig. 4.4.2.1-3. It was 50-70% and almost constant or slightly decreasing in the previous 20 years.

4.4.2.2 Overall trend during the 1990s

As a result of the recent diversification of research in the universities and the increasing number of graduate students, it is sometimes observed that students in the same lab work in different fields of science, which makes it difficult to clearly identify the number of students involved in plasma science and fusion research. Therefore, statistics based on this type of questionnaire are not practical. To get a clearer understanding, the number of contributions to the annual meeting of the Japan Society of Plasma Science and Nuclear Fusion Research, which has taken a major role in domestic fusion research since it was established in 1983, as well as the Atomic Energy Society of Japan and the Physical Society of Japan were polled. However, fusion related research is also performed and discussed in other societies, such as the Energy and Resources Institute of Japan, the Japan Society of Applied Physics, the Cryogenic Association of Japan, the Institute of Electrical Engineers of Japan, the Japan Society of Mechanical Engineers, the Japan Inst. of Metals, the Vacuum Society of Japan, the Japan Radiation Research Society, the Japan Welding Society and the Laser Society of Japan. These organizations were also polled.

(1) Evolution of the number of Japan Society of Plasma Science and Nuclear Fusion Research members

The evolution of the number of members, including students, who belong to the Japan Society of Plasma Science and Nuclear Fusion Research, is shown in Fig. 4.4.2.2-1(a). The number of the total member doubled to approximately 2000 from 1000 in the past 15 years. It is noteworthy that the number of student members increased to 400 from 100, and the increase in the recent years is remarkable. The number of students who became regular members or withdrew from society activities is shown in Fig. 4.4.2.2-1(b). The reduction of withdrawing students and the increase in members in 1987 is ascribed to the fact that many students registered as members in response to stimulation by the society office.

The evolution of scientists engaged in fusion research, investigated by the Statistics Bureau, Management and Coordination Agency, is shown in Fig. 4.4.2.2-2, which is an excerpt from the "Report on the Energy Research in Relevance to Science and Technology Research and Investigation." It indicates that the number of scientists in the fusion research field is almost constant, remaining at around 1000 persons for the past 15 years, which is an apparent contradiction with the number of members who belong to the Japan Society of Plasma Science and Nuclear Fusion Research. Recalling that the number of presentations has not increased as much as the increase in society members, which will be discussed later, it seems that the number of scientists in the fusion research field may not have increased substantially in the last 10 years. The increase in the registered members of the Japan Society of Plasma Science and Nuclear Fusion Research seems to be due simply to the fact that quite a number scientists who were not originally involved in the plasma and fusion research became members of the society.

Student members are mainly composed of graduate students, and this directly reflects the number of students who are interested in plasma science or fusion research. Figure 4.4.2.1-1 and Fig. 4.4.2.2-1 have an overlap of a few years, during the period in which the number of graduate students was increasing. The significant increase after 1993 seems to be the result of an increase in the total number of graduate students, which was intended and encouraged by the Ministry of Education under its policy to further emphasize education at graduate school; "emphasized graduate school" (the related reorganization was carried out in the Engineering Department of Tokyo University in 1992-94).

The increase in the number of regular members registered in the society is an index that indicates the increase of the scientists in the fusion research. As shown in Fig. 4.4.2.2-1(b), it was around 30-40 in the last few years, and neither a remarkable increase nor decrease is observed. It is consistent with the results shown in Fig. 4.4.2.1-3, which indicates that a similar trend was also maintained in recent years. On the other hand, the number of students who withdrew from the society has increased in the last few years. This indicates that a significant portion of the graduate students, which had increased in number after the reorganization of graduate schools, have chosen a career other than fusion research.

The descriptions above can be summarized as follows. Although the number of graduate school students increased, owing to the reorganization of the graduate schools, the number of scientists in the fusion research area has not increased. This may be ascribed to the fact that the positions or opportunities to work in fusion research have not increased. Other possible approaches to investigate would be to refer to the list of members who belong to the Nuclear Fusion Reactor Construction subcommittee of the Atomic Energy Society of Japan or the "Fusion Network" organized in the universities. The Nuclear Fusion Reactor Construction subcommittee of the Atomic Energy Society of Japan was formed in 1991 and is composed of 347 members at present. The "Fusion Network" is concerned with fusion science, reactor engineering, and plasma science, and is composed of 408 research groups having a total of about 1190 research scientists and approximately 1940 students.

(2) Evaluation of the number of presentations to the society meeting by young research scientists

One of the useful indices, which are relevant to the activeness of the scientific research, is the number of contributions to academic societies. The contributions from the universities, which are in practice supporting basic fusion research, are above all important. A certain portion of the contributing scientists is graduate students and junior members of research staffs. In this subsection, the activeness of fusion research is reviewed, based on the total number of academic society presentations and the fraction of university contributions. Here, invited presentations were excluded, and the contributions from graduate students were highlighted. As it was hardly possible to evaluate the number of presentations made only by the graduate student, a fraction factor was hereby introduced. The number of contributions was multiplied by the fraction factor to estimate the portion relevant to the work performed by graduate school students. The presentations from Institute of Plasma Physics, Nagoya University and National Institute for Fusion Science were excluded, as senior research scientists normally gave them. However, the presentations from the Nagoya University graduate school for integrated research was included.

The number of presentations made at the Japan Society of Plasma Science and Nuclear Fusion Research meetings is shown in Fig. 4.4.2.2-3. The total number of presentations increased from 150-200 in the early phase, when the society founds, to 200-250 at present. Since fiscal year 1997, only a general meeting has been held every year, and this resulted in an increase in presentations to 320-360. The fraction of university contributions is around 60%, and has been almost constant for the last 15 years. The fraction increases to 75-80% when the institutions promoted by the Ministry of Education, such as the Institute of Plasma Physics and National Institute for Fusion Science, are included. Figure 4.4.2.2-3 also shows that the number of presentations slightly increased from 1992 with the graduate school student increase fostered under the national policy of "emphasized graduate school" and the start of the ITER-EDA in 1992. However, the change in proportion of presentations from universities is subtle and not obvious.

A similar analysis was performed on meetings organized by the Atomic Energy Society of Japan, where many of the presentations are focused on the reactor engineering and include fusion reactor topics related to the fields of reactor materials, blankets, divertors remote maintenance, safety issues, reactor design, tritium handling, ITER design, and inertial confinement fusion. The number of presentations at the society meetings and the portion of university contributions are shown in Fig. 4.4.2.2-4. The total number of presentations is 80-90, and substantial changes were not observed during the last 10 years, while the proportion of the university contributions is as small as 50-60%. The reason for the small university contribution is that construction and operation of large-scale plant facilities are mainly addressed by the Japan Atomic Energy Research Institute and commercial industries. Graduate students and research scientists who participate in the Atomic Energy Society meetings can be potential research staff members when the construction of the large-scale fusion experimental facilities, e.g., ITER starts, and they may also be coordinating the relevant R&D activities for the demonstration reactor, DEMO, in the future. Therefore, an increase in research scientists in this field is strongly desired for continuous progress in the field of reactor technology.

The session "Plasma Physics and Fusion," organized by the Physical Society of Japan, plays an important role in expanding the range of fusion research and providing an occasion for the exchange of information. The presentations made at the 1999 general meetings (held twice, in the spring and autumn) were categorized into 3 areas, basic plasma science, magnetic fusion plasma, and inertial fusion plasma, and number of presentations in each field is plotted in Fig. 4.4.2.2-5. It is readily seen that nearly half the contributions were concerned with basic plasma science. The number of presentations at Physical Society meetings and the university contribution portion are shown in Fig. 4.4.2.2-6. The number of total presentations exceeded 200 each fiscal year and was almost constant for the last 10 years. The proportion of presentations from universities was 70-80%. Therefore, it is conceivable that the Physical Society meeting is regarded as an important opportunity to make university presentations. The research scientists and students who participate in the Physical Society meetings can be the potential experimental plasma scientists when the operation of the large-scale fusion experimental facilities, e.g., ITER, starts. Accordingly, human resources for fusion plasma research seem to be secured. A word of warning, however, quite a number of fusion plasma scientists tended to change their career to be involved in basic plasma science in the recent years. Therefore, is important to sustain the scientific standards and activeness of fusion plasma research by expanding the range of fusion research and through the exchange of information among the related fields of research.

4.4.2.3 Summary

The number of graduate students in the field of fusion research has steadily increased, although the direct comparison of the tendencies in the 1970-80s and the 1990s is not possible, since the survey methods employed were different. However, the increase in the number of fusion research scientists somewhat saturated in recent years, which indicates that quite a number of fusion scientists in universities and industry have changed their career to other fields of science. However, this attests to the fact that fusion development has broad extensions, sharing common physical issues of concern with other fields. Capable human resources having basic knowledge in fusion science are potentially either in other fields of science or in industry. The results of the efforts related to securing fusion research scientists contribute much to benefit society, and that acknowledges the importance of the fusion research and the need to obtain the broad support of the public. To continue such an effort, which is understood to be the mission of universities, it is necessary that attractive and advanced research be performed in the universities. Accordingly, a firm foundation has to be prepared, where exploratory and advanced research is performed at universities, and this must also include the fusion program.

The expected role of universities is not only to vigorously contribute to the fusion plasma research but also to explore new scientific areas of research and to strive for breakthroughs in technology. Therefore, the accomplishments achieved at universities should influence other fields of science. In practice, the number of presentations related to the fusion and plasma science presented at scientific societies not directly concerned with fusion plasma research has increased. For example, semiconductor-processing technology was originally developed in the course of fusion plasma research and later applied in the semiconductor industry. Fusion plasma technology will have many similar spin-offs in the future.

As a result of the expansion of the fusion research to other fields of science, the activeness and attractiveness of fusion research is sustained, and the number of graduate students in fusion research has increased. Should this interpretation be correct, the solicitude about the loss of fusion plasma scientists is not necessary. Instead, we should convince ourselves that capable human resources with a basic knowledge in fusion science are available. It is above all important in the future to exchange information among related fields of research so that the understanding and support of the public is gained and maintained.

However, it should be noted that if the loss of fusion plasma scientists in universities and industry continues, sustaining viable research activities in universities as well as incorporating and maintaining the technological standards in industry will become difficult.

Furthermore, recalling that a reduced portion of undergraduate students show an interest in fusion plasma research in recent years, it is important to sustain the attractiveness of fusion research with a positive and sincere attitude. In addition, the Japanese fusion community should also be preparing for the situation of decreasing population and lack of interest in science among school children.

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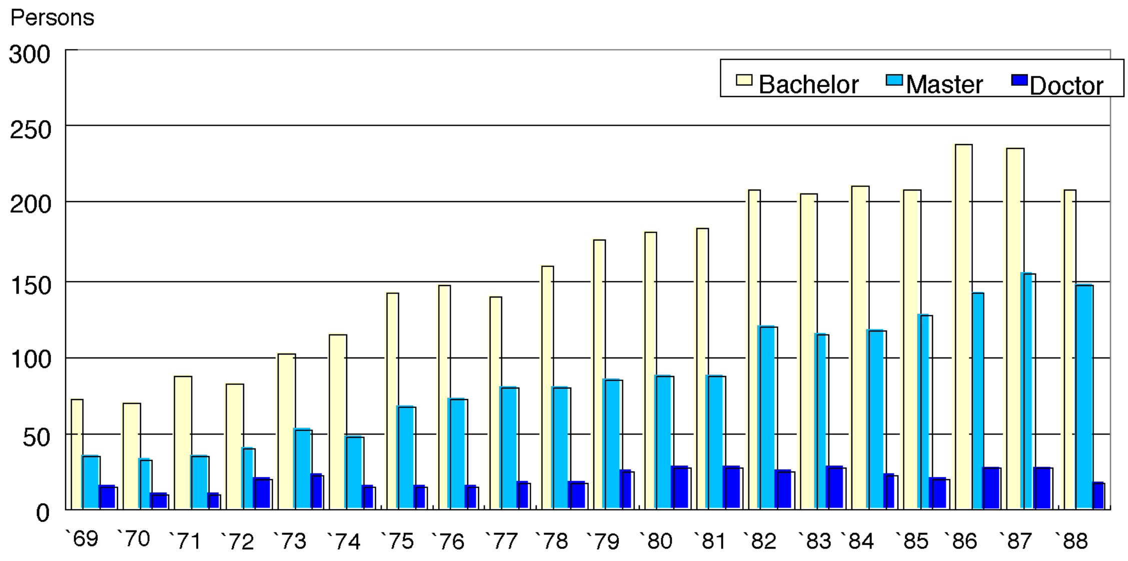

Fig. 4.4.2.1-1(a) Transition of the number of students in the fields related to plasma science and nuclear fusion research

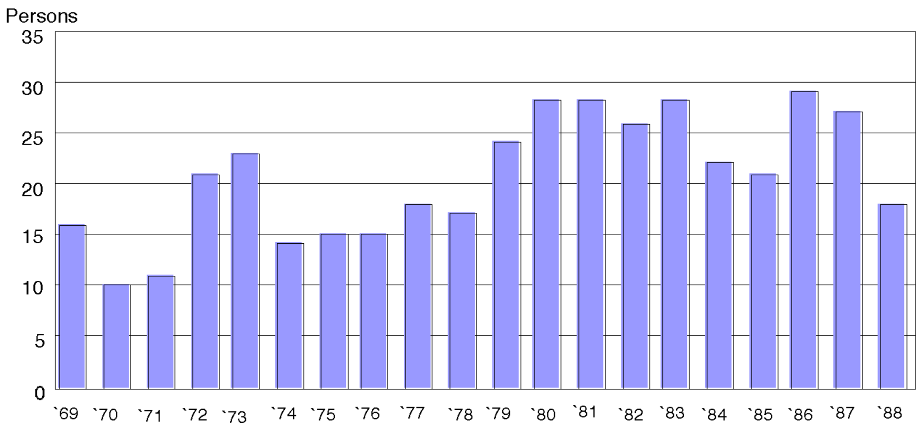

Fig. 4.4.2.1-1(b) Transition of the number of graduate students in doctoral courses in the fields related to plasma science and nuclear fusion research

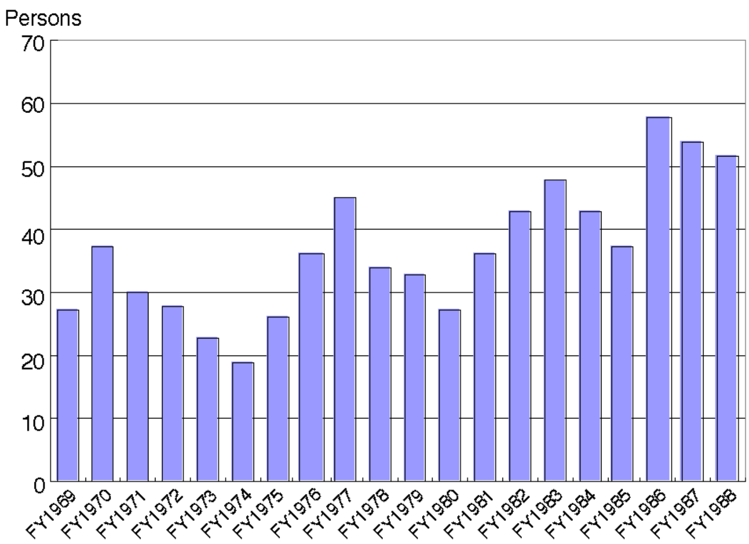

Fig. 4.4.2.1-2 Evolution of the number of doctoral degree holders in five universities (MIT, PPPL, Univ.of Maryland, Univ. of Wisconsin, UCLA) in the fields related to plasma science and nuclear fusion research in U.S.A.

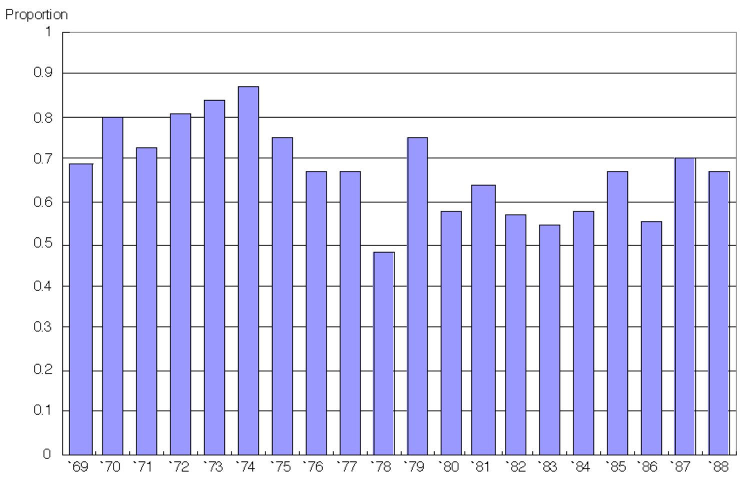

Fig. 4.4.2.1-3 Proportion of doctoral degree students who found positions in the nuclear fusion professional field

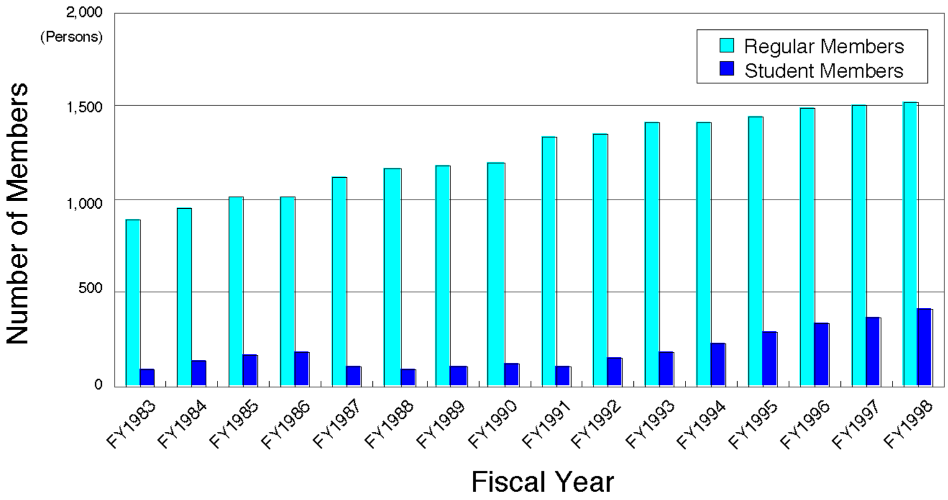

Fig. 4.4.2.2-1(a) Tendencies of Members of Japan Soc. of Plasma Sci. and Nuclear Fusion Research

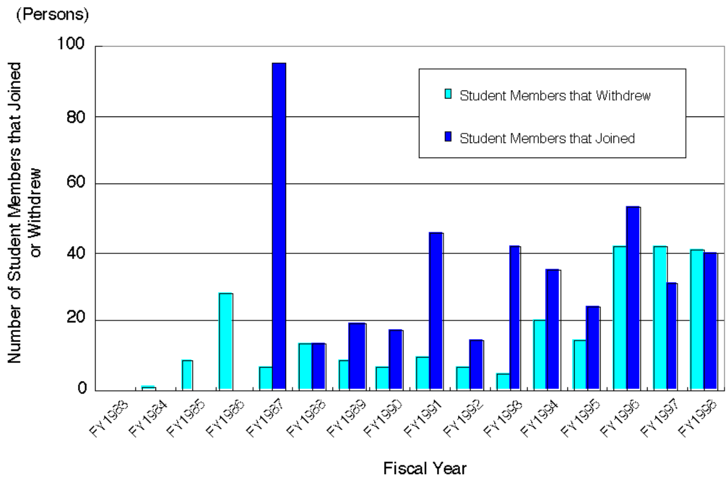

Fig. 4.4.2.2-1(b) Tendencies of student members of Japan Soc. of Plasma Sci. and Nuclear Fusion Research

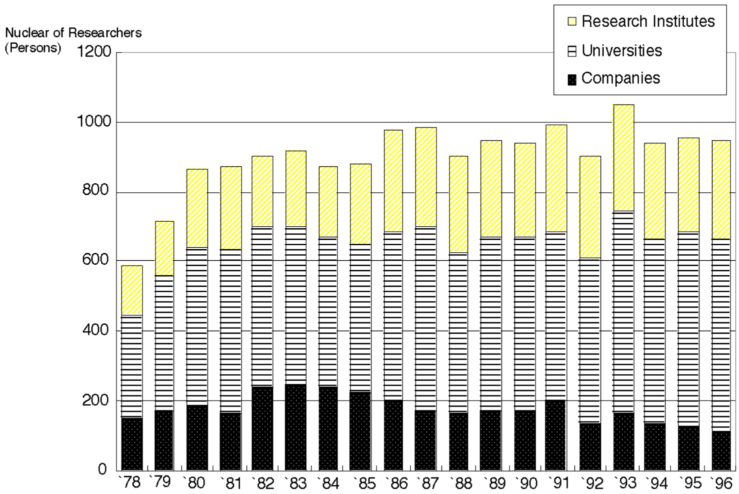

Fig. 4.4.2.2-2 Evolution of the researchers in nuclear fusion research

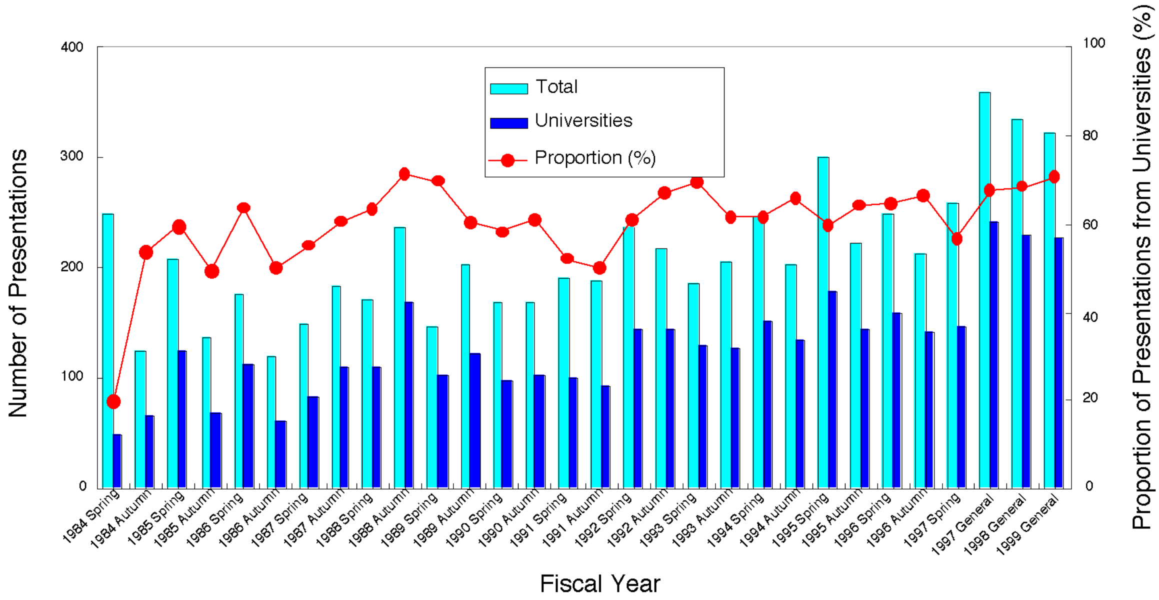

Fig. 4.4.2.2-3 Number of general presentations at meetings held by Japan Soc. of Plasma Sci. and Nuclear Fusion Research

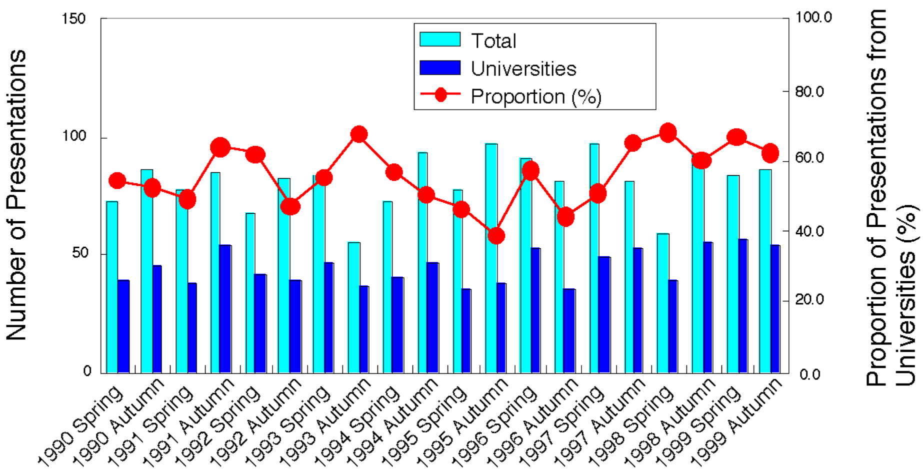

Fig. 4.4.2.2-4 Number of general presentations in meetings held by Atomic Energy Soc. of Japan

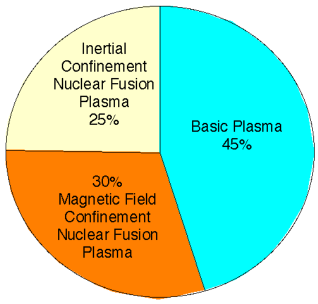

Fig. 4.4.2.2-5 Presentation categories of meetings organized by the Physical Society of Japan (FY 1999)

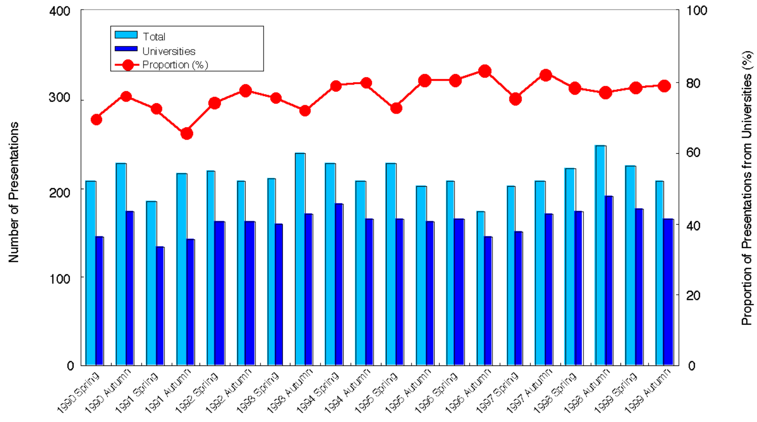

fig 4.4.2.2-6 Number of General Presentations in The Meeting held by Physical Soc. of Japan

4.4.3 The cooperative structure between universities and research institutions

The cooperative structure between universities and research institutions in Japan spans several decades. The Nuclear Fusion Round-Table Committee (President: Hideki Yukawa, 1958), Atomic Energy Commission (Nuclear Fusion Specialty Subcommittee, 1958) and Nuclear Fusion Ad Hoc Committee of Science Council of Japan (1959) were founded in the early phase of the fusion research in Japan, where a coordinated framework of fusion research programs was intensively discussed. Accordingly, cooperative research has been performed. However, the scale of fusion research has become enormous. Therefore, further effort to coordinate the collaborative work between the universities and research institutes is presently required in order to reduce the development risks and to maximize the efficiency of available resources.

The following collaboration framework has been coordinated between the Japan Atomic Energy Research Institute and universities in regard to ITER activities.

(1) Contracts based research and investigation related to the ITER / EDA
As one of the approaches for the universities to contribute to the ITER / EDA, research topics are defined by the Japan Atomic Energy Research Institute and universities conduct investigations after the specific topics have been reviewed by the ITER / EDA Research Collaboration Committee organized in the university. The number of topics approved and investigated for fiscal years 1993 to 1998 was 130.

(2) ITER physics R&D
In ITER physics R&D, the physics committee coordinates all activities and directs the activities of seven specialist sub-groups. The specialist sub-groups are required to conduct investigations directly relevant to the ITER program. The universities and the Japan Atomic Energy Research Institute participate in this activity as equal partners. In some of the sub-groups, databases are produced to improve the ITER design.

(3) Participation in meetings related to ITER
- The ITER technical advisory committee is composed of 4 members specialized in fusion research from each Party. In the case of Japan, 3 out of 4 members are selected from universities, and one of them is the chairperson of the committee. The university staff participates in the special working group meetings and the blanket module test program, which are organized by the IC (ITER Council).
- 223 technical meetings and workshops were organized in 6 years, and many university staff members participated in these meetings.

In the ITER / EDA agreement, the Japan Atomic Energy Research Institute is designated as the representative institution in Japan, and the role of universities and industries has been to support the engineering design activities in collaboration with the Japan Atomic Energy Research Institute. To successfully proceed with the construction of ITER, however, it is necessary that all research institutions, universities, and industrial partners work together as a single entity. Accordingly, all cooperating partners should secure the necessary human resources and be prepared to send them to the ITER legal entity.

4.4.4 From the viewpoint of industry

The focus of fusion research presently is in transition from the demonstration of scientific feasibility to the demonstration of engineering feasibility, and the role of industry has become increasing more and more important. Through the manufacturing components used in the fusion device, the industry contributes to resolve the energy issue. The industry also recognizes the significance of the construction of ITER as a challenge to the innovative technology in advanced and unprecedented fields of science [4.4.4-1, 4.4.4-2, and 4.4.4-3].

The industry was responsible for all construction procedures of the fusion devices in the past, including the detailed design, manufacture, performance testing, and even the implementation of the relatively new technologies. However, for the construction of ITER, development of a totally new and significantly more advanced production technology is required. For example, the production and quality control technologies of the superconducting coil material used for ITER has been developed as a result of intimate collaboration among industry, the Japan Atomic Energy Research Institute, and the universities. In addition, considering the fact that it takes more than 10 years from R&D production to the completion of construction, scheduled training of the engineers is necessary to retain the technical standards of quality. This is important not only for ITER but also to be prepared for the demonstration reactor, DEMO. The following is a description on securing human resources in industry

(1) Sharing the technical achievements of the international cooperation (ITER) activity

It is not sufficient for ITER relevant technology development to simply introduce the results of technical achievements developed by another Party. Instead, engineers, research scientists, and other necessary personnel in industry should be involved in the manufacturing process to extend their own technical capabilities.

Along with the hardware technologies directly relevant to the ITER tokamak machine and the auxiliary facilities, and the systems engineering software technologies, is a large part of the hardware technology that is classified either as core technologies or as core components. Providing this hardware and software requires unprecedented approaches and requires engineering R&D. Therefore, the engineers themselves will have to be deeply immersed in the manufacturing processes to acquire the necessary skills, as mentioned in (2) and (3). Also, in regard to the software technology, the core part of the systems engineering algorithm would have to developed by engineers who have experienced being involved in organizations that are concerned with the tokamak design and construction and physics operation.

On the other hand, technologies classified as conventional technologies or technologies for conventional components are already established in the industry of each Party, and it is possible to share the result of technical achievements by exchange of information.

(2) Significance of the individual technology developments

A manufacturing technique cannot be acquired without experience. In this case, opportunities to learn the necessary manufacturing skills should be provided to engineers based on the following reasons.

1. Limitation in the exchange of information by the inter-company barriers
   Even if the information related to the final results obtained by a manufacturer, namely A Corporation of the X Party is transferred to the another manufacturer, namely B Corporation of the Y Party, the B Corporation will be unable to produce the specified components in general. In most of the cases, the key information is not transferred, because of the inter-company barriers.
2. Limitation in the quality of information on technologies with multiple solutions
   Even if adequate information is provided from the A Corporation to the B Corporation, the B Corporation may be unable to produce the specified components if the relevant manufacturing technology is not available at the B Corporation.
3. Significance of individual technology development
   Even if the adequate information is provided from the A Corporation to the B Corporation together with the detailed guidance on manufacturing, the B Corporation may not be able to provide the products to the customers with a quality assurance sheet. It is necessary for the engineer in the B Corporation to experience all production procedures, i.e., the background of the specifications, the R&D, the design, the selection of materials, the detailed procedures of manufacturing, such as the working method and connection method, the inspection of the component and the integrated performance test. Corporate in-service training includes all of the above and only through such complete training are capable engineers produced.

(3) Accumulation and transfer of technical information in industry

The results obtained in the process of technology development are either accumulated as drawings and document sheets or in the memory or intuitive sense of engineers. In particular, the latter is passed down from generation to generation in a person to person style, thus continuous recruitment of the engineers in the relevant field is necessary. Once the knowledge is lost, it requires a vast amount of investment for recovery. Especially in the unprecedented case of the field of engineering in fusion technology, the impact of the loss of information is enormous.

(4) Industrial support of the technology and the human resources for fusion research

In the case of our country, the major companies have actively contributed to the progress of the fusion research from the early stages of development. The method of such research coordination has been recognized and highly regarded by our overseas partners. Obtaining engineers involved in fusion research in the industry is generally performed by two methods. (1) One is to have the engineers dispatched to the research lab to be involved in integrated fusion research technology, and (2) the other is contract based collaboration.

With the former, necessary preparation is completed, and engineers in the industry are assigned to the ITER / EDA and other fusion research activities. For example, in the Naka fusion research establishment of the Japan Atomic Energy Research Institute, about 100 of the 400 persons involved with ITER / EDA have been dispatched from Japanese companies. These engineers return to their companies a few years after their assignment began, and other engineers are dispatched to replace them. As for the ITER joint central team, more than 60% of the personnel sent overseas from Japan originally belonged to the private enterprises. Thus, approximately 40 persons are engaged in integrated fusion research every year. When ITER construction starts, they are expected to take an important role in coordinating the on-site labor and in training the less experienced engineers, along with those who have been involved in the manufacturing of the components for ITER and other fusion devices.

The latter presumes the continuity of the contract order from the research lab. However, after the modification of JT-60 and completion of the construction of LHD, another new large-scale device in not within the present planning, and neither is the time of the start of ITER construction obvious. This is a particular issue of concern in that the number of personnel in fusion related departments have been decreasing as a result of the restructuring of the intra-company organizations in the middle 1990s and the reduction of new contract orders. This situation seems to be serious for ITER construction if the trend continues. It would take considerable time to recover a large number of engineers at the start of ITER construction. In addition, the loss of transfer of technical knowledge from one generation to the next becomes quite significant. The only solution to overcome this problem is to make an early decision regarding ITER construction, clarify its schedule, and continuously provide the contract orders for the necessary R&D and design studies before the start of the construction. Thus, the loss of engineers in the industry could be prevented, and the human resources would be secured, even if the start of ITER construction were delayed due to authorization of the ITER legal entity and the safety review of the host government. Furthermore, time is provided to train the less experienced engineers in the industry.

(5) Conclusion

Fusion relevant technology development has been performed in Japan under the national fusion development program and it has been coordinated among the national research institutions and universities. Intensive effort has been devoted to implementing both the world's leading scientific personnel at universities and outstanding production technology in the industry. For involvement of industries in ITER construction, further improvement of the technology standards and an increase of experienced engineers are required based on the accumulated knowledge hitherto acquired and considering the issues pertaining to human resources described in the previous subsections.

In order to coordinate the corporate in-service training effectively, it is necessary to clearly define a concrete strategy, one in which the milestones of the progress in engineering is described in detail corresponding to the technology development requirements. In regard to sustaining the technology standards over the generations, it is favorable to continuously construct facilities of appropriate size at appropriate intervals. Therefore, it is desirable to make continuous investments in the development of the fusion relevant technologies in Japan, under the elaborated long-term national fusion development program, the scope of which is extended to a period even after completion of the ITER program. The international collaboration is also an important issue. The steady progress of a coordinated program also contributes much to retaining and attracting new human resources.

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4.5 International Cooperation

4.5.1 Reviews of International Cooperation

(1) Purpose of International Cooperation

When research and development program becomes so large in scale that it taxes the resources and personnel of a country, international cooperation is very effective in minimizing the risk and in conducting fruitful research and development efficiently. Especially in the field of research and development for fusion energy, international cooperation is indispensable. With the steady progress of fusion plasma research, the time has come to construct a large experimental device and conduct large-scale activities in it. In addition, the international community requires Japan, which has achieved major advances in science and technology and is a major economic power, to make a positive contribution to society.

(2) Status of International Cooperation in Research and Development of Fusion Energy

Reflecting the situation mentioned above, international collaborations are now conducted vigorously in various forms. In multilateral cooperation, large-scale collaborations with financial contributions by each collaborator are being implemented, like ITER activities and the exchange of information or activities for fruitful discussion such as the Fusion Energy Conference and expert meetings under the International Atomic Energy Agency (IAEA). Furthermore, collaborations are widely being carried out under frameworks of the International Energy Agency (IEA), including cooperation among the three large tokamak facilities in the world, collaborations on plasma confinement systems other than tokamak like stellarator and reversed field pinch, collaborations on radiation damage in fusion materials, and collaborations on the environmental, safety and economic aspects of fusion power. Japan is involved in bilateral cooperation with the United States, the EU, Canada, Australia, Russia, China, Korea, and other countries. Especially in cooperation with the United States, the Coordinating Committee on Fusion Energy is established under an agreement on cooperation between both governments, and collaborations are conducted in four frameworks, a personnel exchange program, joint planing, joint research projects, and joint research for plasma physics. The joint research projects include joint research for advanced plasma confinement using the Doublet III, a tokamak facility located in United States, joint irradiation experiments of fusion materials using the FFTF and HFIR reactors, joint experiments on tritium safety technology using the TSTA tritium processing facility, and a collaborative program on a data link. Many researchers from JAER, ETL, NIFS, the universities and institutes, and technicians from industry are involved in these collaborative researches. The main plasma confinement devices in the world with which fusion research and development is being carried out and the institutes that are collaborating with Japan are shown in Fig. 4.5.1-1. The frameworks of international collaboration in which Japan is now involved are shown in Fig. 4.5.1-2.

(3) Background of Expansion of International Cooperation

These international collaborations are strongly connected to the level of progress in research and development in Japan. In the field of fusion, all research and development conducted has been open to the world since the early days, about 1960, so discussion and exchange of information has flourished from the beginning, but the well organized international collaborations only started about 1980.

Those collaborations did not start because the advanced nations sought efficient solutions against financial difficulties caused from escalation in scale. In the late 1970s, these advanced countries vigorously strove to enlarge the scale of fusion research and development, not because they felt they were facing financial difficulties, but they had strong intentions to proceed with international competition by learning quickly other nation’s technologies, especially world-leading technologies. In this context, advanced nations undertook international cooperation for their own benefit, and they did not recognize any nation without sufficient potential worthy of collaboration with them. Actually, in early 1970s, the United States would not accept Japan as a partner of any collaboration (4.5.1-1). Just prior to 1980, we came to be accepted, and, later, the United States choose us as a partner in every collaborative program. This fact shows that they could no longer ignore the Japanese policy, such as the Second Program, of making achievements, as well as in fusion research at institutes and universities in Japan (US-Japan cooperation in fusion research and development began in 1979).

As mentioned above, international cooperation began as a scheme for the exchange of information, and extended to bilateral and multilateral collaborative research including personnel exchange, and to projects aiming at joint construction of facilities, such as the ITER project. Now, international cooperation has matured well, and we should use it to the best advantage in a systematic way.

(4) Results of International Cooperation

Scientific and technical results of the international collaborations carried out to date now can be found in reports of specific collaboration efforts and in publications (in great detail) of scientific societies. This paragraph below provides an overview of the results from the viewpoints mentioned below.

1) Results which could not have achieved without international cooperation, regardless of the scale of costs

One of the typical examples of this category is the cooperation on the chronic tritium release experiment in Canada, which was conducted in the cooperative program on environmental, safety, and economic aspects of fusion power under an IEA framework agreement and in Japan-Canada cooperation. This experiment could not be executed in Japanese territory, but through the cooperation much valuable data was obtained.

Significant results in fusion plasma research were realized by experiments with tritium at JET and TFTR, though it was not beyond expectations based on the outcome of D-D experiments.

In the field of fusion technology, collaborative research on fusion materials, using the Fast Flux Test Facility (FFTF) and the High Flux Isotope Reactor (HFIR), remarkably raised the potential of Japan in material science. In the design activities for the International Fusion Materials Irradiation Facility, which are now under implementation under the framework of the IEA, it was very beneficial that the results of the design and R&D of United States' Program on high flux neutron source (FMIT) was able to be put into practical use, and that such facilities as the French deuterium ion beam test facility equipped with a neutron shield was made available to us. The required capability in such a facility did not exist in Japan.

2) Collaborations that were effective in reducing the financial burden, and those that contributed to the progress of the fusion research program and brought results that exceeded our expectations

Of course, significant benefits have been obtained in almost all cooperative programs. One large principal example of a collaboration that involved the transfer of funds is the Doublet-III cooperative program, which we joined before the construction of JT-60 started. Instead of construction of a middle-scale tokamak device, which Japan had cancelled, Japan contributed 15 billion yen to the Doublet-III device and obtained a share of the machine-time to conduct experiments for our own programs. As a result of this, we acquired techniques to generate and control plasmas that could be applied directly to JT-60, the large tokamak device, and this made it possible for us to continue seeking achievements in experiments with advanced plasma.

On fusion technology, in the collaborative program on the Large Coil Task under the IEA, in which six coils were developed and manufactured by the participating nations. These coils were arranged in the figure of a torus and were provided for the experiment in time to gain data on the characteristics of large superconducting coils. The total cost of this experiment was estimated to be 30 billion yen, yet Japan joined the project and shared the results for only the cost of one coil, which was estimated to cost 2.4 billion yen. (United States carried out, as the host country, the production of 3 coils, and the construction and operation of the experimental facility)

To implement international cooperation, we have to have more accountability than when conducting a domestic program, and, therefore, the legal and other details of any decision must always be clear. This has played an important role developing personnel having the talent to be active in international scenes, and this has provided significant incentives, especially to young researchers.

3) Values aside from scientific and technical ones

Communication with foreign researchers enabled us to understand each other's views on future fusion reactors, which need acceptance by society, and thoughts on safety, with approaches on safety examinations and differences in ways to carry out a project. Communication was also valuable in that a worldwide network among researchers was built (the Internet) and is now in use.

In addition the research activities, communication helps develop a mutual understanding among people from other countries in that researchers (including their family members) become involved in the daily life and cultural activities with researchers and the general populace of the other party.

4) Total evaluation

International cooperation will bring some difficulties. Even striving for the same objectives of research, there are various ways to conduct the research. Especially in a new field of research and development, there is no textbook and it is difficult to agree on the way to conduct the research, because each researcher relies on his/her own ideas, which are new and original. Therefore, it takes much time and effort to agree on the details of a collaborative program. Additionally, among large projects, we can find many such cases where unexpected situations in a participating nation's budget arose and required the other parties to reapportion their budgets to resolve (compensate for) the problem.

However, in spite of such complications, it is highly appreciated that, in total, the beneficial results of each international cooperation in every field of fusion research have exceeded the expenditure and effort of each participant.

We have to note that in each international cooperation on fusion carried out until now, Japan had a facility similar to a facility of the other party and that environment for research was completed to understand and absorb the data provided from other party. In other words, those international collaborations were executed not to complement each other's potential, but to search for progress or make rapid advancements by joint research or supplemental research with partners having the potential to understand each other. In seeking progress in science, competition of ideas is the basis for obtaining successful results, and we must retain this essence. Achievement of continued beneficial results in the future will depend on how successful we are at harmonizing both competition and cooperation as the scale of the collaborations becomes larger.

4.5.2 ITER project and international cooperation

(1) Relation of Japan to the fusion research and development program

Methods to conduct the fusion research and development synthetically are specified in the Third Phase Basic Program of Fusion Research and Development, which was established by the Atomic Energy Commission in 1992. The relationships between these specifications and the specific issues discussed at this subcommittee are explained in Chapter 2 of this report. The results that have followed have enabled us to forecast the feasibility of the program, including development of an experimental reactor (ITER) as the core.

For progressive research and development with an experimental reactor as the core device, the knowledge acquired to the present can be used to the best advantage to successfully complete the program. But, at the same time, some flexibility is necessary to cope with timely optimization of the program during this, the execution of the Third Program. Such flexibility is required because the program includes unprecedented technologies in the fields of physics and other sciences.

Fusion research and development at this stage should consist of not only research concerning the experimental reactor as the core device but should also include critical subjects such as "complementary efforts in the areas of fusion plasma study that can not be covered by the experimental reactor and the advanced research and development to confirm and demonstrate new fusion plasma technologies before employing them into the core device in each phase, including the experimental reactor." Furthermore, studies for materials that can survive high-flux fast-neutron irradiation, studies for the system design of the future fusion power reactor, and safety studies must be conducted at this stage. These subjects must be coordinated well and executed toward the demonstration reactor (DEMO) as the goal of this stage, and must be conducted by timely reexamination and modification of the position of each subject to maintain the balance of the Third Phase Program.

Therefore, we are required to show our strong conviction and our commitment to successfully accomplish the Third Phase Program of Fusion Research and Development. Japan's initiative to host ITER would not show only our resolve to realize ITER, but also our intention to execute the advanced and supplemental plasma research to support the ITER and the development of materials that cannot be carried out in ITER. It will also be an incentive for fundamental research as the basis to support fusion research and development, and make it possible to achieve the whole fusion program in its entirety. In international cooperation, individual sharing of the collaborative responsibilities is very important, but leadership by one of the collaborating nations and the expression of its strong determination to carry out the project will provide the impetus for all to successfully complete the project.

(2) Relation between Practical Use of Talent and Vigor of Cooperative Research

In advanced research that depends on innovative ideas and concepts, the capability of individual researchers is critical for its success. In the large project, individual idea is assessed from various viewpoints and more researchers are involved to refine the idea to be a comprehensive program. Institutes and governments are expected to play their role in coordinating this process and in guiding the selection of superior ideas. A typical method should include embracing international cooperation.

Without considering financial matters, we might respond to the query, "Does Japan have enough talent and human resources to construct a fusion experimental reactor--can she accomplish the construction with the results obtained by domestic activities and international collaborations implemented to date and with the total technological potential of Japan?" We understand that domestic execution would make it easier for us to make decisions. But, to apply the results of other nations' activities and the international collaborations to the experimental reactor (ITER), direct participation of the researchers who were involved in those activities is essential. Furthermore, the process for agreement or decision at the international level seldom allows for a conclusion with uncertain ground since many personnel must review such a conclusion from various viewpoints and this requires very serious discussion. Such deliberation leads to a more reasonable and optimized choice. Especially for the development of the experimental reactor (ITER), because it is not the ultimate goal of fusion research and development, we have to make decisions on its total performance and the specifications of its components considering their relation to the technology of the future power reactor (Commercial Reactor). Thus, participation of researchers on the design of the commercial power reactor is necessary. In other words, international collaborations are implemented not only for apportionment of human and financial resources, but also for improvement of the quality of decisions to optimize the project with reviews from various clarifying viewpoints. This philosophy has universality that applies to both the technical and administrative aspects of the project.

However, the progress of science and technology depends on ideas created by each researcher, and the competition among researchers encourages them to create ideas. Therefore, even in such a large project as ITER, it is essential to keep competition at the level of individuals, institutions, and nations within it. For example, international competition in physics and engineering R&D and competition on site selection for the design activities was effective for the ITER Engineering Design Activities (ITER-EDA). In the next step of the international cooperation on ITER, its construction, it is essential to maintain the competition of industrial technology among nations in the manufacturing of core components and in the optimization of the device at construction. It may be difficult to construct many core devices due to both technical and financial reasons, but it is important to continue the ancillary or supplemental programs, such as the blanket system and fundamental research, and the basis of competition is essential to obtain advanced results. Accordingly, for the success of a robust project, it is very important to maintain a good balance between cooperation and competition within the entire Japanese program of fusion research and development.

(3) Share among the parties

One of the ways of sharing when involved in international collaboration activities is to decide each participant's share in the beginning of the project. Another way is to decide only principles in the beginning and decide on specific plans starting with the most important one. In both cases, the host nation must have a firm and strong resolve to provide leadership for, as well as to participate in, the whole fusion research and development program. The role of the host nation is very important in maintaining the stability of the project, especially an international cooperative project, because it is often affected by political or economic situations in the other partner nations. While the merits and demerits of being the host nation and the detailed manner of sharing are delineated in Chapter 2 of this report, a balance with the main objectives or the purposes of the project is indispensable when addressing specific problems or troubles.

As mentioned above, the role of personnel talent is important, even in large projects. In particular, with such a long-term project as fusion research and development, we must regard inflow of knowledge and intelligence from talented personnel more than the outflow of results by international cooperation. To acquire such knowledge and intelligence, it is important to make every effort, with internationalized judgement and wisdom, to prepare an attractive and stimulating atmosphere to encourage talented personnel to join this project. Doing so will open the entire fusion research and development program, including ITER, for the world to join and will provide full access to the results of this research.

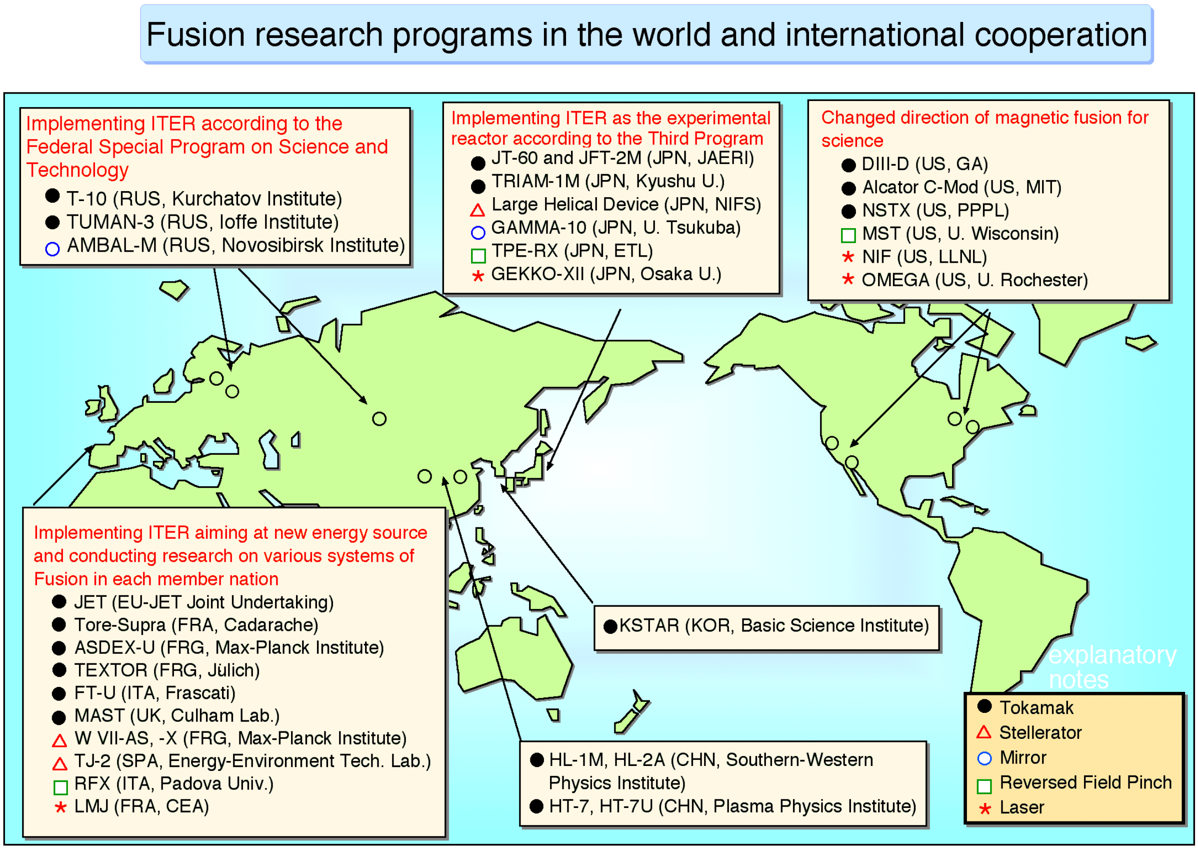

Fig. 4.5.1-1 Major research institutes and devices for fusion plasmas

International collaborations on fusion research and development

(only collaborations under inter-governmental agreement, in principle)

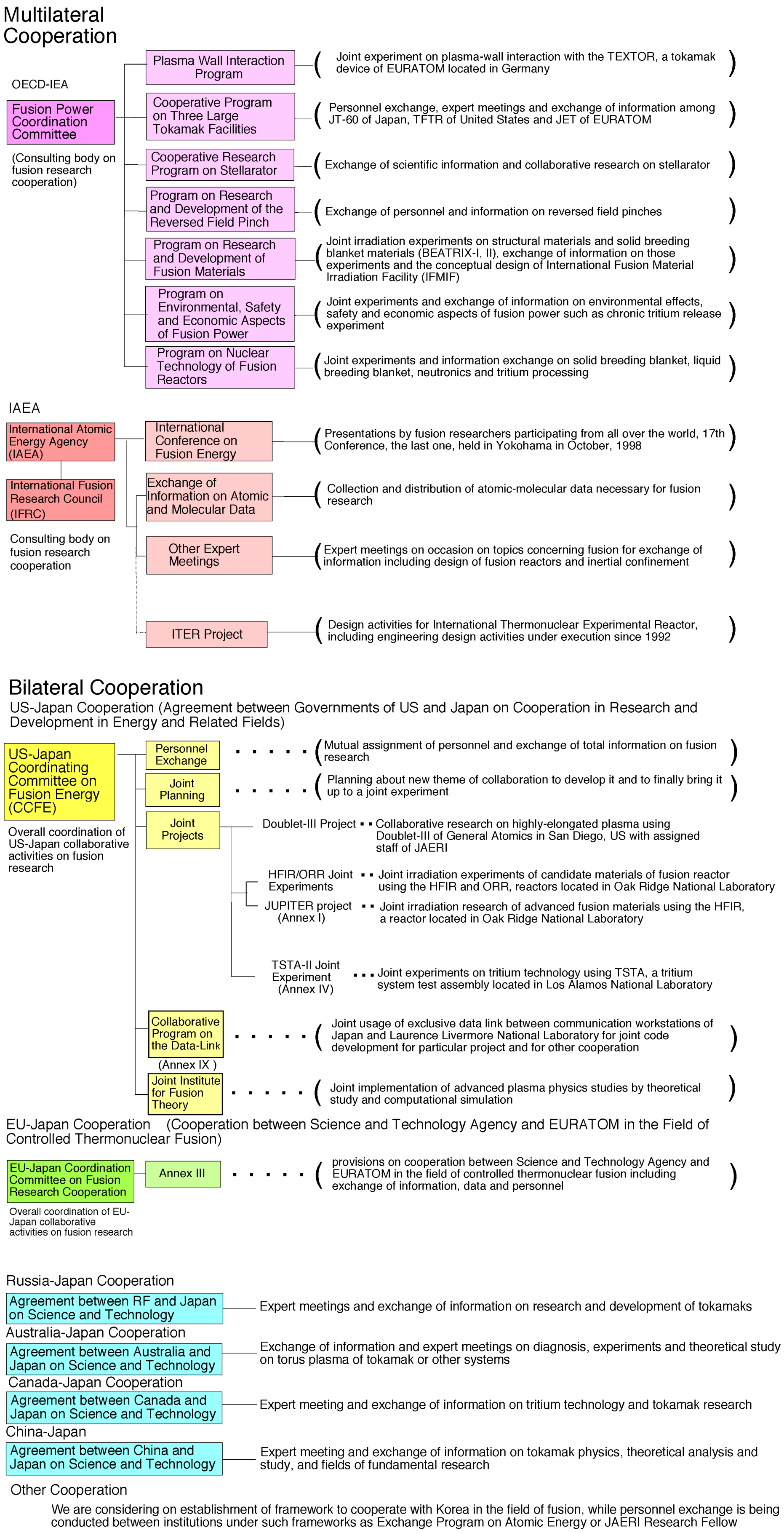

Fig. 4.5.1-2 Major international cooperation in the fusion research with Japanese participation

4.6 Summary of Extension of Fusion Program and Basic Supporting Research

This report concludes by discussing extension of the fusion program and basic supporting research as follows.

1) Fundamental Study for Fusion Research and Development

- "Plasma Physics" is the fundamental study that supports not only fusion but also advanced technology in various 21st century fields. To understand and control the conditions of plasmas is a critical issue to resolve problems in such fields as beam physics, accelerator physics, advanced photon science, and new substance production.
- In the ITER project, we are challenging new themes of research to scientifically clarify the various complex phenomena that appear in burning plasmas and to control the systems ruled by autonomy physics including self-organizing phenomena caused by interaction between the self-control function of plasmas and the fluctuations that arise in the process to confine plasmas in a limited space on earth.

2) Technological Spin-off from Fusion

- Fusion development requires the advanced technologies of various fields of engineering technology. Progress was made in related fields of engineering technology in the process to develop the advanced technologies necessary for ITER, and in turn fusion research contributes to the development of advanced technologies in other fields and to the progress of basic scientific research.

3) Status and Role of Fusion Research and Development in Universities

- University studies on physics and technology with tokamak and other plasma confinement systems, such as the helical, inertial, reversed field pinch, and spheromac systems, have contributed much and have elevated Japan to the position of world leader in advanced fusion reactor research. Universities have also fostered an understanding and academic systematization of plasma physics and control of plasmas. Moreover, their efforts may lead us to the development of a fusion reactor with a fusion plasma superior to what we now expect.
- Japanese universities can contribute to ITER and projects following by research of advanced confinement techniques and the education of students and junior researchers.
- To continue the education of young talent and to keep and improve the high level of their research staffs, it is very important for universities to make their fundamental research activities more active and attractive for graduate students.
- Fundamental research activities performed in universities are important for realization of an optimized configuration of a fusion reactor as well as for academic contributions. These activities must continue to be performed vigorously.

4) Role of Industry

- Japanese industry can contribute to the solution of the energy problem with their high-level manufacturing technologies and quality control. Joining with the activities of universities and in such projects as ITER, industry can uplift new enterprises, promote technology transfer from generation to generation, and improve manufacturing technology by incorporating advanced manufacturing processes.

5) Structure of Cooperation among Universities and Institutes

- Efforts to further strengthen the cooperation among universities, institutes, and industry are important to successfully achieve the ITER project and the fusion research activities that will follow ITER. These efforts will provide talented personnel for the main roles of those projects. This will be a very important influence because these projects shall extend over a long period of time.
- The importance of the above aspects has been stressed for some time and the effects of these efforts is now becoming apparent. We have to steadily continue these efforts.

6) International Cooperation

- International cooperation has allowed us to execute such activities as the chronic tritium release experiment and the D-T experiments, which we could not carry out inside Japan. Further, it has helped introduced internationalized talent and this has accelerated Japan's entry into the world community.
- As well as participating researchers an international collaboration, their families contribute to better mutual understanding through daily activities, including cultural activities.
- In any field related to fusion, it is totally understood that the benefits gained by international cooperation are much greater than the participation costs and efforts. Therefore, it is important to continue international cooperation for the progress of fusion research and development.

PART 3

Summary

Chapter 5 Summary

The Special committee on the ITER Project (Chairman: Hiroyuki Yoshikawa, Chairman of the Science Council of Japan) has identified six issues for decision regarding the hosting of ITER in its intermediate report, "Summary of Discussion and Remaining Subjects in the Committee," as follows,

(1) Survey of long-term energy demand and supply
(2) Feasibility study of alternative energy sources
(3) Technical feasibility of fusion energy
(4) Extension of the program and basic supporting researches
(5) Distribution of resource for researches
(6) International relations

The Subcommittee of Fusion Development Strategy, coordinated under the Fusion Council, investigated (3) and (4) in response to the decision of the Fusion Council (June 12, 1998).

For this report, the committee widely investigated the significance of fusion, the progress of fusion, the issues to be solved to realize fusion energy, etc., for the DT tokamak fusion system. This chapter is the last chapter of the report and contains summaries of the findings.

Part 1 Technical feasibility of fusion energy

It is necessary to define the characteristics of a target fusion reactor to discuss the technical feasibility of fusion energy. The importance and possible role of fusion energy to solve the energy issue will be described herein. In addition, the measures and pathway to realize fusion development are clarified. These are followed by the proposals on what we should do now to realize fusion energy.

5.1 Energy issue and the role of fusion power in the 21st century

Prospect for energy demands

It is very difficult to predict the energy demands of the 21st century. A significant increase in energy demand is expected due to a large increase in population in developing countries and the expected improvement in their living standards. However, the energy demands in developed countries may be reduced with the development of energy-saving technologies. It is predicted that development of new energy sources to satisfy large energy demands will be required in the 21st century based on the large increase in the energy demands that occurred at the end of the 20th century.

Criteria selection

An economic figure of merit for a power grid is thought to be an important factor for a power plant to be selected as an energy source from the present viewpoint of energy economy. However, it is possible that the selection of energy resources in the middle of the 21st century will be made based on comprehensive criteria related to sustained development capabilities, since the issue of environmental preservation becomes more important. Alternative comprehensive criteria may address the abundance and distribution of energy sources, safety and environmental effects, supply stability, energy security in view of the international context, and resilience against the proliferation of nuclear weapons. These elements are related to an economic figure of merit in a general sense and the criteria for energy selection would be a comprehensive one, different from those of today. The significance of the use of fusion energy is enhanced in such a situation.

5.2 Characteristics of fusion energy

Fuel resources

The fuel resources for fusion energy are deuterium and lithium and lithium is used to produce tritium. Deuterium and lithium are abundant in seawater (inexhaustible in fact), broadly distributed, and are stably affordable. International strain on energy resources, possible accidents during fuel transfer, and crisis management would be minimal for fusion energy due to these factors. Fusion energy will become, in practice, a domestic energy source, one that has been eagerly desired since Japan began to develop into a modern country.

Special materials

The advantage of fusion would be reduced if any specific material used in fusion reactors suffers from resource limitations. Therefore, required material resources are evaluated assuming that world electricity consumption is supplied by fusion energy. It has been shown that the world's material resources are sufficient to provide fusion energy for several centuries (resource base) to a few tens of thousands of years (ultimate potential material resource).

Role

A fusion energy power station would be a central energy source, like a fission power station, and would provide a stable base-load energy source. It would deliver its electricity immediately to any place in our country since the electric network is well developed. The need for central power stations as stable base-load energy sources will not change in an industrialized society; even if distributed power sources play significant roles in the future. Fusion energy will be suitable for distributed power systems, such as fuel cells, through the production of secondary fuel, such as hydrogen fuel.

Safety

Avoidance of nuclear accidents that accompany a chain reaction is a most important subject, one that concerns the safety of a fission reactor. In fusion, such nuclear accidents with loss of power control cannot take place in principle, and the biological hazard potential[1]of mobile radioactive material[2] for fusion is 1000 times less than that for a safely operated commercial fission reactor.

With the above advantages being retained, it is possible to design an adequately safe fusion power station with a radiation dose at its site boundary well below the regulated value, even if an unrealistic accident ever happened. If such a fusion power station is constructed with public acceptance near a city, based on these advantages, it becomes possible to substantially reduce power transmission costs and to utilize the available thermal energy for industrial purposes and residential heating of the local community. Such a situation is desirable from the viewpoint of energy savings and environmental protection.

Special attention to the safety issue should address the treatment and management of tritium and radioactive dust. For tritium treatment, safety assurance with multiple confinement barriers and closure of the fuel cycle can be incorporated into the fusion power system design. Radioactive waste, such as the expendable first wall, will be produced during and after the operation of a fusion power plant. However, implementing the “shallow land burial” scheme can significantly reduce the potential biological hazard poised by this waste, as no high-level waste is produced in fusion. Within a few hundred years, safety would be assured, and it would even be possible to recycle most materials involved with fusion energy. Many issues associated with radioactive waste transfer would be avoided if all radioactive materials were treated within the site.

Comparison with fission reactor

Both fusion and fission power reactors are energy sources with minimal global warming gas emission rates resulting from manufacturing, construction, and operation. Fusion power will be more costly than fission power because fusion power relies on high technology components, such as large superconducting magnets. On the other hand, fusion has many advantages, such as its inability to have a serious accident due to a nuclear chain reaction, its low biological hazard potential poised by mobile radioactive materials (1000 times less than that of a fission reactor), and its lack of high-level radioactive waste. It may also have other advantages, such as possible siting near cities and reduced chances for nuclear weapons proliferation. Overall judgement should consider these fusion power advantages for the selection of key future energy sources.

5.3 Stepwise approach toward the realization of fusion energy

Fusion research in Japan is promoted in a stepwise approach based on the "Basic Program" decided by the Atomic Energy Commission. In this stepwise research strategy, clear targets are defined for each step. Transition to the next step should be made when the appropriateness of the next step target and the scientific readiness to proceed to the next step is well assessed. This stepwise approach is being adopted to continuously develop the large-scale, expensive fusion system and to minimize the risk.

Second-phase basic program

The main objective of the second-phase basic program was to establish plasma production and confinement techniques in which fusion power is equal to the auxiliary heating power. This target has been realized in the tokamak device JT-60, which was built as the core device for the second-phase basic program.

Third-phase basic program

[1] Risk index as a necessary air volume to dilute its radioactivity to the allowable level when all radioactivity in the reactor is released as a gas in case of accident.

[2] Radioactive material mobilized by convection and diffusion of air in case of accidental release. Typical isotopes are tritium for fusion reactor and iodine-131 for fission reactor.

The third-phase basic program is the present fusion development program being pursued in our country. The main objective in the third-phase basic program is to establish control techniques for the burning plasma (induced by fusion reactions) and to form the technological basis necessary for the development of the fusion Demonstration Reactor (DEMO) to demonstrate the generation of electricity. The tokamak concept was selected for the core device in the third-phase basic program as well as that in the second-phase basic program, although other various confinement concepts have also been intensively investigated. The International Thermonuclear Experimental Reactor (ITER) was adopted as this core device in our country. Various technologies necessary to proceed to DEMO will be developed in ITER, but one major design guideline is the "single step to DEMO."

Advanced and supplemental research on tokamak devices as well as research on non-tokamak advanced concepts and development of fusion technologies, reactor materials, safety engineering, and fusion power system design are to be performed in the third-phase basic program, in parallel with burning plasma research conducted using ITER.

Fusion plasma research for the operation of ITER and to establish high-performance plasma confinement techniques for DEMO should be carried out as advanced and supplemental research in tokamak devices. Major research themes are the establishment of steady-state operation, suppression of disruptions, realization of low-temperature divertor plasma, improvement of plasma confinement, etc.

On the other hand, based on non-tokamak advanced concepts, other magnetic confinement systems, such as helical, reversed field pinch, compact torus, mirror, spherical torus, and inertial confinement systems are being studied. This research is aiming at the possibility of other attractive fusion power concepts, concepts that might have advantages over the tokamak concept. It is important to pursue research on these advanced concepts in parallel with ITER so that confinement schemes for DEMO can be selected from the various promising concepts. Thus, the R&D risk of fusion research can be reduced.

Engineering issues of various fusion technologies required for the development of the DEMO reactor are being solved through the ITER program. However, the neutron flux in ITER is insufficient for testing blanket structural materials to be used in the steady-state fusion DEMO reactor. Therefore, it is necessary to construct the 14-MeV neutron source and test heat-resistant, low-activation structural materials.

<u>Fusion DEMO phase</u>

The step after the third-phase basic program is the fusion Demonstration Reactor phase (DEMO), which has the principal objective of the engineering demonstration of electricity production by fusion power. Electricity production in a large-scale fusion power plant will be realized for the first time in this phase. The confinement concept for this phase will be decided based on the investigation of the tokamak type DEMO study and the performance of other confinement concepts at the transition from the third-phase basic program to this phase.

<u>Commercialization phase</u>

The step after the DEMO phase is the commercialization phase. If economic feasibility is demonstrated in this phase, fusion power will be qualified to penetrate the energy market as a commercially competitive option. The government will retain leadership through the DEMO phase, but the commercialization phase will be led by the private sector. Therefore, the prospects for commercialization should be firmly established by the DEMO phase.

5.4 Control of burning plasma and technologies addressed in ITER

<u>Control of burning plasma</u>

Fusion research progressed substantially in the last half century. Most research has been devoted to understanding plasma behavior. Plasma confined in a magnetic field is an autonomous open system in which the plasma itself determines its physical state while exchanging energy with its surroundings. Although the control of high-temperature plasma is very difficult due to this characteristic, plasma control techniques have been established and a knowledge base has been accumulated as a result of this long-term research effort. This knowledge base, however, was established in plasma that was not accompanied by fusion self-heating. Therefore, it is not certain whether the present knowledge is adequate and applicable to self-heated fusion plasma.

Self-heating power produced by the fusion reaction will be applied to the burning plasma itself, while only plasma heating from the external sources has been examined in experiments to date. It is difficult to predict burning plasma behavior with the present knowledge base since fusion self-heating simulation using external power is difficult. Therefore, without understanding this burning plasma behavior, it is difficult to clearly predict the technical feasibility of fusion energy. Nonetheless, fusion energy development can be achieved by advancement of existing technologies if the control of burning plasma becomes possible. Thus, the understanding and control of burning plasma is the last big challenge of fusion energy research.

Plasma is controlled by a magnetic field. The controllability of burning plasma can be proven by demonstrating that the plasma can be sustained stably for a period long enough for the magnetic field to control the whole plasma. For ITER, this period is 300-500 seconds.

The size of the experimental device and the strength of the magnetic field to satisfy these conditions have been defined using information accumulated in the non-burning plasma databases. The database applicability can be confirmed in the experimental reactor itself. Once confirmed, the design of the DEMO and the commercial fusion reactors becomes possible.

Technology development

The principal experimental objective of ITER is the production and control of burning plasmas. To achieve this objective, new technologies and facilities are necessary. Requisites are large super-conducting magnet technology to produce high magnetic fields, technology for an auxiliary heating facility to heat large-scale plasmas to 100 x 10^6 degrees centigrade, blanket technology to breed tritium (which is rare in nature), technology to safely handle tritium, radiation shielding technology for equipment and materials exposed to high-energy neutrons, radioactive material disposal technology, remote maintenance technology to assemble and disassemble large precision structures very accurately in intense radiation environments, as well as heat removal technology for plasma facing components exposed to the high heat flux, and handling technology of power, and particles from the high-temperature plasma. Furthermore, system technology to integrate these technologies and associated facilities as well as to assure a high level of safety and reliability is required. A major milestone for these technological developments, which are indispensable for the DEMO reactor, is the construction and application of ITER. Moreover, these technological developments are essential for both tokamak and other confinement concepts.

New developments for DEMO

Technologies required for DEMO should be developed in parallel with those needed for ITER. By confirming them in ITER, one major ITER design guideline, a "single step to DEMO," can be realized. Major issues of concern are discussed below.

(1) Development of steady-state operation scheme
The basic principle of steady-state operation in tokamaks has been proven at a number of research institutions in Japan and other countries. It is important to fully develop steady-state operation methods through the most productive use of existing tokamak devices and to apply their performances to ITER operation, especially to the burning plasma in ITER. At the same time, it is important to establish operational methods that avoid plasma disruptions, which preclude steady-state operation.

(2) Development of high-temperature blanket test modules
The blanket plays three important roles, neutron shielding, tritium breeding, and extraction of high-temperature thermal energy. The latter will produce steam for generation of electricity. To accomplish the technologies relevant to these roles, a high-temperature blanket is required. Developed in ITER, its design will be available for DEMO.

(3) Neutron irradiation test
Development of reduced activation materials that allow intense high-energy neutron irradiation and high-temperature operation is required to enhance safety and economics of fusion. Leading candidates for blanket structural materials to be used in DEMO and beyond have been identified. However, performance of these materials should be confirmed by neutron irradiation tests, as the material database has not been satisfactorily completed at present. Neutrons produced in ITER can be used for irradiation tests at low fluence and for component tests.

Item (2), (3), and much previously developed technology have common applications in tokamak and other confinement concepts.

5.5History and present status of the ITER project

History

The ITER project was initiated in 1988 as an international cooperative program among the United States, the EU, the Soviet Union (now the Russian Federation), and Japan. Many design activities have been successfully completed, and practical construction techniques have been investigated. Construction details of ITER were defined in its final design report (called ITER-FDR), published in 1998. A major difficulty was its high construction cost, estimated to be ~ 100 x 10^{10} yen. The design of a compact-sized ITER, with a construction cost about half that estimated in the ITER-FDR, has commenced. Three parties, the EU, the Russian Federation, and Japan, are designing the ITER after the international framework of collaboration was unexpectedly changed by the US withdrawal from the ITER project.

Impact of the US withdrawal

The US fusion policy to postpone the integration step for fusion research, i.e., the construction of ITER, and to alternatively proceed with the multi-machine intermediate step was debated both inside the US fusion community and among international experts. A report from the ITER Special Working Group (1998) concluded that the overall R&D cost and the required period of research will increase as a result of the US policy. However, the importance of burning plasma research has again been recognized by the US, and the SEAB (Secretary of Energy Advisory Board) recently recommended that re-entry into the ITER program, even with partial participation, should be considered if ITER construction starts.

Present status of the ITER project

The fusion power and burn time of the ITER was reduced by 1/3 to1/2 of the ITER-FDR design. Also, the design philosophy has been substantially modified to emphasize steady-state operation. Now in the design phase, the current ITER design satisfies the requirements of an experimental reactor as defined in the third-phase basic program. Except power generation technologies, research and development of burning plasma confinement and its long pulse control with deuterium and tritium fuels as well as the integrated demonstration of fusion technology are now feasible in ITER.

5.6 Actions related to the ITER project

Philosophy of international cooperation

Acquiring environmentally friendly energy sources to meet future energy demands is a particularly important issue of concern in Japan, which has been importing most of its energy resources from other countries. Specifically, the demand for a stable energy supply is more stringent than in the EU or Russian Federation. Based on past achievements, Japan has acquired international status as one of the leaders in fusion research. In addition, developed by capable scientists and engineers and industries, high technology is readily available in our country. Therefore, our country is well qualified to participate in the ITER program. Justified by the above reasons, Japan can either make, or is expected to make, significant contributions to the ITER project. Energy research is relevant to our national interests. However, the reason the ITER program is implemented as international collaboration and not as international competition is that various incorporeal values are found in ITER as coordinated under the international collaboration framework. The reduction of our share for construction costs is not the sole reason. Politically, ITER is important as a symbolic project that manifests the transition from a state of “confrontation” to that of “cooperation” in the international community. Scientifically, ITER is valuable as an important opportunity for our country to take the initiative for the development and advancement of burning plasma science, which can then benefit all humankind. This is similar to the US role in the space station project or the role of the EU in the LHC (Large Hadron Collider) project. Technically, ITER is valuable in promoting the development of advanced technologies through fusion relevant developments. The ITER program is also valuable from a social and cultural point of view in which international personnel exchanges through this cooperative work are involved.

Merits for hosting ITER

The merits of constructing ITER in our country are substantial. Hosting ITER in Japan will favorably help to maintain our leadership in fusion development. It is also important for Japanese industries to be involved in the development of various high technologies. Hosting ITER in Japan can show the international contribution of our country. Furthermore, the ITER infrastructure will remain in our country, and will be available for DEMO. This is an important opportunity to promote the stepwise approach of fusion research in our country.

Demerits and issues for hosting ITER

More national funds will be required for Japan to host ITER than if ITER is constructed abroad. Therefore, some concerns remain that hosting ITER in our country may affect the development of other fields of research or that the possible channeling of research funds for these other fields to ITER may hamper creative research in these other fields. The decision for hosting ITER should be made considering these points.

Fundamental considerations and issues to host ITER

The final decision to host ITER should be made with a national understanding and the agreement of the local community where ITER will be sited. Should serious siting obstacles to the ITER program arise after site selection in Japan, the reputation of our country as a reliable partner will be lost, and this will influence the fusion program for other parties. If Japan offers to host ITER, our country should take full responsibility for the smooth performance of the project and sincerely consider the needs of participants from the other parties.

It is also important to solicit the understanding of the local community, especially the issues of the necessity of tritium transfer from abroad and the continued surveillance of radioactive materials for a certain period, even after the completion of the project.

5.7 Advanced concept research and fusion technology

Research on non-tokamak concepts and its importance

The importance of non-tokamak concepts advanced research lies in their potential advantages over the tokamak, which are either not yet realized, or are difficult to realize in a tokamak. For example, steady-state operation, which has not yet been established in a tokamak, as well as the disruption phenomena that accompany the fast plasma current quench are not issues of serious concern in a helical system. It is recognized that it may be possible to realize more favorable concepts in a non-tokamak configuration, which necessitates non-tokamak research in parallel with that of the tokamak.

Contribution of non-tokamak concepts to ITER

Advanced technologies, such as the plasma heating and current drive, the plasma diagnostics, and fuel injection, all developed in non-tokamak research, are useful for the ITER program. Research with small- and medium-size devices in the universities are useful to explore new physics areas and perform plasma confinement "proof of principle" experiments. Universities also contribute much to improve the scientific capability of the less experienced research staffs in fusion relevant fields, which is of significant importance for ITER program promotion.

Important subjects of fusion technologies

Most component technologies of the fusion reactor will be realized through the ITER program. The blanket facing the plasma is a high-temperature component that performs important functions that include tritium breeding and neutron shielding in the adverse environment of high-energy neutron irradiation. The preparatory investigation for the DEMO blanket will be performed with a blanket module test using neutrons produced in ITER. However, the neutron fluence available in ITER is inadequate for testing blanket structural materials for the steady-state DEMO reactor. Therefore, development of high-temperature and low-activation structural materials is necessary in parallel with the ITER program. This development effort will employ the 14-MeV neutron source. The development of high-temperature and low-activation materials is important for safety assurance and also for the economic viability of fusion energy. With such materials, a compact fusion reactor may be achieved, and the storage period for activated materials used in the facility can be reduced. Realization of "long-lived" materials contributes to the reduction of maintenance and the extension of the reactor lifetime. Improvement of the thermal conversion efficiency by increasing the operating temperature will enhance the economics of fusion power. Development of high-strength field magnets is also to be emphasized to expedite the realization of the low-cost fusion reactor.

5.8 From ITER to commercial fusion power generation

Possible realization in the near future

A fusion reactor that generates electric power with repeated pulses, each having a duration of a few to 10 hours, is technically feasible using ITER technologies--provided the long burn is realized in ITER. From the present perspective, such a reactor tends to be large in size and not economically advantageous.

Fusion reactor extrapolated from ITER

Construction of the tokamak DEMO reactor having steady-state operation will become possible when the preparatory research on technologies required for the DEMO reactor, i.e., steady-state operation technologies and high-temperature blanket technologies, are developed in ITER. The construction and operation of the DEMO reactor will demonstrate the technical feasibility of fusion energy. The DEMO reactor is to be a prototype of a commercial fusion reactor and will complete the research and development phase of fusion reactor technology. In the third-phase basic program, the configuration of the DEMO confinement system will be decided taking the achievements of non-tokamak research into account—although the tokamak configuration is regarded as the fundamental arrangement. The burning plasma study and the fusion technologies developed in ITER will be useful even if a non-tokamak system is adopted for DEMO.

Cost reduction

Fusion must be economically competitive with other energy sources to enter the energy market. Reduction of the reactor size, the cost of maintenance, and the frequency of inspection and replacement, as well as the attainment of steady-state operation of the reactor and further improvements in plasma performance are vital to enhance the economic competitiveness of commercial fusion power reactors.

Size reduction

Development of low-activation materials and high-strength field magnets as well as the improvement in plasma confinement performance will be effective in reducing the size of the reactor. Proposed candidate low-activation materials have been selected and an irradiation test of these materials with the14-MeV neutron source is necessary.

A good prospect has been identified for the manufacture of higher strength field magnets than those of ITER. An engineering demonstration is such a high-strength field magnet is envisaged.

Maintenance, inspection, and replacement

Periodic replacement of the structural materials during the lifetime of a fusion reactor is necessary due to the irradiation damage caused by energetic neutrons. Effective approaches to reduce the cost of electricity are to extend the period between replacement of components by the extension of material lifetimes and to reduce the replacement times required. The former will be realized by the development of high-performance materials, and the latter will be accomplished by the improvement of maintenance procedures.

Steady-state operation

It is necessary to establish the steady-state operation mode for commercial fusion reactors, for the continuous operation of a power station is prerequisite to provide economically attractive and commercially competitive fusion power. If this mode is established in the burning plasma produced in ITER, it will contribute much to enhance the economic attractiveness of the commercial fusion reactor.

Improvement of fusion plasma performance

Reducing the cost of electricity produced using a magnetic confinement system will be effectively be achieved by confining the plasma at high pressure, namely, by realizing the confinement of the high-performance plasma. Research on the improvement of fusion plasma performance, in other words, confinement of high-performance plasma, should be continued even after the realization of fusion electric power generation. The research results obtained will contribute to improving economic attractiveness.

Time scale of realization of fusion power generation

From a technical point of view, power generation by the DEMO reactor would be possible around 2040--if developments in technology progress as they are doing at present and if achievements in ITER are incorporated into DEMO. It is expected that more than 10 years will be required after the DEMO reactor is operating satisfactorily for fusion power to be recognized as a commercially available option having economic competitiveness. An accurate prediction is difficult, as it depends on the progress of innovative technology, on the economic features of other energy sources, and on mid-century social conditions.

5.9 Conclusion of Part 1

The technical feasibility of fusion energy will be confirmed by demonstrating control of burning fusion plasma, by establishing the technical feasibility of an integrated fusion device, and by accomplishing safety and reliability in ITER. Furthermore, a high-performance fusion reactor will be realized by establishing steady-state operation. Most major technologies required for the DEMO reactor and beyond can be developed as an extension of ITER. Therefore, the prospects of fusion development for the DEMO reactor and beyond will become clearer during the ITER program, as compared to the present situation where clarification of physical phenomena receives more emphasis. In addition, it is possible that the construction cost of the DEMO reactor will be lower than that of ITER due to development of materials, technological innovations, and the progress of plasma physics. A similar possibility could apply to a commercial fusion power station that would follow DEMO.

Part 2 Extension of the Fusion Program and Basic Supporting Researches

Described below are extended scientific and organizational impacts produced by fusion research in other scientific fields, including the discovery of new fields of science, technological spin-off effects, organizational structures to promote fusion relevant research, and methods to educate and retain capable scientists. Following these discussions, the importance of the extension of the program and basic supporting researches for the realization of fusion energy will be clarified.

5.10 Significance of fusion research in universities and related organizations

Fusion plasma research in universities is extensive in the context that tokamak and non-tokamak advanced research as well as other basic plasma research to systematically understand plasma physics and acquire the scientific knowledge relevant to the control of high-temperature plasma are intensively performed. Such systematic knowledge bases are indispensable for promoting and accelerating the development of fusion reactors and identifying guidelines for making them compact and efficient. Newly acquired knowledge in plasma physics promotes interaction with other scientific fields of research and creates new frontiers in science. In addition, fusion plasma research has promoted many spin-off technologies, originated from plasma production techniques, various plasma heating technologies, fuel injection technology, plasma diagnostic techniques, and techniques to control high-voltage electric power. Many of these new technologies have applications in diverse industrial fields. Fusion technologies are comprised of various fields, namely, in-vessel material engineering, structural materials engineering, blanket engineering, tritium science, super-conducting magnet engineering, electromagnetic structural engineering, tritium biological effects science, thermal structural engineering, reactor design engineering, system safety engineering, neutronics engineering, and inertial confinement fusion reactor engineering.

Computer networks are being constructed among universities and related organizations to identify the topics of research, to prepare common samples for research, and to develop and expand applicable databases. Each university and institute is performing various fusion related basic research by fully utilizing their facilities and academic expertise. These research activities all support the basis of reactor technologies for the ITER project. They also contribute to the resolution of long-term issues relevant to fusion development and to the continuous education of personnel in various fields.

5.11 Expansion of fusion research organizations and education of personnel

Research organizations

In Japan, fusion research scientists are distributed among many universities, research institutions, and industries. These scientists are members of academic societies relevant to the field of their interest, such as The Japan Society of Plasma Science and Nuclear Fusion Research, the Atomic Energy Society of Japan, The Physical Society of Japan, The Institute of Electrical Engineers of Japan, The Japan Society of Applied Physics, The Japan Society of Mechanical Engineers, The Japan Institute of Metals, the Cryogenic Society of Japan, The Laser Society of Japan, the Japan Welding Society, The Vacuum Society of Japan, the Japan Radiation Research Society, and the Japan Society of Energy and Resources. The total number of members of these societies is in the thousands and the members are distributed over a broad range of ages.

Education of personnel

According to a survey, the number of graduate students participating in fusion research is steadily increasing. On the other hand, the growth rate of the total number of scientists in fusion research is diminishing somewhat. This is probably due to many research scientists in universities and companies who were originally involved in fusion science but changed their research field. This manifests the interdisciplinary aspect of fusion research. In other words, numerous capable scientists with a basic knowledge of fusion research are potentially available from industry. By preserving a certain number of scientists in industry who have been involved in fusion research, fusion research will be recognized by the societies and will be widely supported by them. However, if the present trend continues, many fusion researchers will be diverted to other fields. It then will be difficult to continue the research or perform development relevant to fusion in universities, and the succession of technological developments presently achieved in industry would hardly be possible in the future. Should the number of fusion research scientists continue increasing at the present rate in the future, the required human resources would be available for the ITER construction phase, taking the number of experienced fusion scientists now distributed to other fields into account. Nonetheless, to prepare for the upcoming experiment phase of ITER, we should continue to retain a number of younger fusion scientists and to improve their research skills. To accomplish this, providing basic research grants to universities and thus attracting graduate students is very effective. Since the ITER project is a long-term program, spanning a range of 30 years from the initiation of construction, continuous effort to maintain and increase the number of talented researchers is required. It is therefore necessary to formulate a national policy to train competent scientists and engineers in fusion research. This requires further reinforcement of fusion relevant research and technology development organizations in universities and industry. The great variety of attractive research in universities should encourage graduate students to participate in fusion research.

Promotion structure

Since ITER and following fusion research programs are long-term national programs, it is necessary to adopt a formal system that enables researchers in universities, research institutions, and industry to participate in practical activities to provide substantial contributions. A rudimentary investigation on possible approaches to realize this concept is about to start.

5.12 Academic aspects of fusion research

Plasma and fusion

Plasma is the fourth state of matter after solid, liquid, and gas. It is ionized gas, composed of nuclei and electrons. Initially, plasma was a research subject in space physics and electronic engineering. After fusion development started, basic research on plasmas progressed very rapidly and gave birth to new interdisciplinary fields, such as plasma physics and plasma science. Plasma physics is based on classical physics, such as dynamics, electromagnetics, fluid mechanics, statistical mechanics, thermodynamics, and relativity theory. The main research subjects involved therein are the many-body problem and the nonlinear problem of plasmas. Research on the complicated, but elementary processes of atoms and molecules in plasmas has also made great progress. Plasma science is a field that covers a broad range of plasma physics and deals with the various applications of plasmas, including fusion. This field interacts deeply with other diversified fields, such as electric engineering and electronics, nuclear engineering, materials science, space physics, and geophysics. While fusion research is directed primarily to the development of energy, it is also playing an important role in expanding the frontiers of science, as seen particularly in universities. To establish the control technology for reactor plasmas, it is necessary to thoroughly understand the behavior of plasma interacting with electric and magnetic fields. Plasma is an active medium with complexity, multiplicity, and non-linearity. To confine high-temperature plasmas smoothly with a magnetic field, it was necessary to identify and find solutions for various instabilities that were discovered one after another.

It has been vital to construct a new knowledge base by the interplay of experiments, theory, and computer simulation, and it by nature took a long time. Of course, cooperation between Japanese and international researchers and the exchange of information amongst the various research institutions were of great help in solving problems. With this effort and cooperation, we have elucidated much of the behavior of this strange medium "plasma" and have advanced to where we can begin the investigation of burning plasmas. Some important subjects of pure plasma physics are the nonlinear phenomena and non-equilibrium transport phenomena, such as the propagation of solitary waves, the generation of turbulent flow and chaos, and the self-organization phenomena. These are also frontier subjects of modern physics. The production of perfectly ionized heavy element ions contributes to atomic physics and accelerator physics.

5.13 Spin-off benefits of fusion technologies

Driving force of spin-off technologies

Since fusion development requires gathering knowledge from a myriad of advanced technologies, it is now making significant progress as a seed of these technologies. The fusion device is based on diverse research fields and fashioned from advanced technologies, such as physics, mechanical engineering, electric and electronic engineering, materials engineering, thermodynamics, heat transfer flow and thermal engineering, nuclear engineering, cryogenic engineering, electromagnetic dynamics, chemical engineering, and control engineering and instrumentation. Therefore, the development of this compound technology not only advances individual fusion technology but also raises the potential capability of all science and technology by mutual stimulation between different fields of science. The resultant spin-off benefits are seen in commercial technologies, such as the semiconductor industry and the large, precision machine-tool industry. Fusion research also contributes to the development of advanced technology and basic science of other fields, such as physics, space science, materials science, medicine, communications, and environmental science. These applied sciences include accelerator technology, superconductor technology, diagnosing techniques, plasma application technology, heatproof and heavy-irradiation-proof materials technology, impurity removal techniques, and computer simulation techniques.

Examples of spin-off technologies

Examples of spin-off technologies include the development of large superconducting coils for ITER, which reduced the cost by 75% of niobium/tin superconducting wire material necessary of the generation of the high-magnetic fields. This has allowed the high-magnetic field MRI used for medical diagnostics to become relatively commonplace. At the same time, the AC loss has been reduced by 80% of that for conventional superconductors, even at the strong magnetic field of 13 tesla. This makes it feasible to increase the stored energy in a superconducting power storage system by a factor of 5-7 when compared with a system designed with conventional technology and operating at 5-6 tesla. In addition, vacuum pumps for high thermal efficiency refrigerating machines, which operate below 4 K, have been developed and have been adopted at the Fermi Laboratory in the US and CERN in Europe. This also demonstrates the enormous contribution of fusion research to the frontiers of science. The technology of producing large positive-ion-beam currents, originally developed for the heating of fusion plasmas, has already pervaded into the technologies for products used in daily life, such the semiconductors used in the home electric appliances. In addition, the large negative-ion-beam current technology developed for ITER is expected to give birth to completely new research fields, such as the creation of previously unknown materials. The negative-ion beam, which has monochromatic energy, is also suitable for manufacture of intricate semiconductor devices. This allows the realization of low-cost, mass-produced single crystal silicon thin films for solar cells. Furthermore, high-power radio-frequency sources used for plasma heating are already applied to the manufacture of high-performance ceramics. Potential applications of these sources extend from solving environmental problems to the radar used in outer space. The integration of component technology for the fusion reactor also advances the systematic development of technologies addressing integration, such as system engineering, control engineering, and safety engineering. Additionally, an exploratory investigation related to the processing of radioactive waste by utilizing a fusion reactor itself as an intense neutron source is also being carried out and seems promising.

5.14 International cooperation in fusion research

Noticeable international cooperation in fusion research started around the end of the 1970s. The original intention was not to reduce development costs by sharing the technical information but rather to compete with each other on the advancement of plasma performance. In other words, it was meant to promote leadership by performing research a step ahead of collaborating parties and, based on information obtained under collaboration, to stimulate domestic research and development. Thus, international cooperation in fusion research began as a means of information exchange, and it then evolved into bilateral and multilateral competitive cooperation. However, it has matured to a multinational stage of collaboration, as represented by the ITER project, which aims at a multinational construction program.

International cooperation is very effective in conducting efficient research and development while minimizing technical development risks, because the scale of research and development is enlarged and vast amounts of human resources and funds are invested. For example, international cooperation enabled research on tritium emission and DT burning experiments, which have not been carried out in our country. This was effective for Japanese scientists to acquire the most advanced knowledge in the world. Further, the profound intermingling of individual cultures not only by the scientists but also by their families through cultural activities in daily life contributes to international amity. As a whole, achievements made through international collaboration are valued much more than the funds and labor expended on the program. For advancement of fusion research, it is important to continue the coordination of international cooperation.

The ideas of individual scientists are the source of advancements in science and technology. The creation of a new idea is enhanced by competition among researchers. Therefore, for the coordination of international cooperation, even for a large project such as ITER, it is important to maintain competition between individuals, research organizations, and nations to maintain progress and eventually attain project success. We have witnessed international competition in physics and engineering R&D that made the ITER EDA more reliable. The same logic applies to construction of ITER. It is important to maintain the competition within industries and between countries to optimize the equipment production process.

5.15 Conclusion of Part 2

Fusion research interacts with a variety of sciences through plasma science and fusion engineering, and contributes to the expansion and profound understanding of each field. Our country has a firm foundation in academic and industrial organizations that strongly promote fusion research--and a large number of scientific fields are significantly influenced by fusion research. To produce competent and dedicated scientists who support fusion research for a substantially long period of time and to permanently establish the basis of fusion research including ITER as a promising field of science, it is necessary to sustain continuous fusion research activities in universities and to advance fusion technologies in industry. Investigations of practical approaches for universities, national research institutions, and industry to contribute to the ITER project are about to start. Our country has devoted great international effort to contribute to the realization of humankind’s ultimate energy source. As a result of this process, we have significantly diversified our R&D projects. Now we have many facets of basic research and numerous technologies that surround fusion research, and all of these are quite energetic and prolific.

Appendix 1

Establishment of Subcommittee of The Fusion Council for Fusion Development Strategy

June 12, 1998
The Fusion Council

1. Objectives and Establishment

To consider overall development strategy toward realization of fusion energy, the Fusion Council establishes the Subcommittee for Fusion Development Strategy.

2. Investigation and Deliberation

The subcommittee investigates and deliberates matters concerned with;
(1) Technical feasibility of fusion energy
(2) Establishment of bases for fundamental research in various fields, education of personnel, and role and cooperative structure of universities and industry, supporting fusion reactor development over a long period of time
(3) Development strategy of fusion reactor

3. Period for Investigation and Deliberation

Period for investigation is one year and the subcommittee is dissolved at a point in time when the report, in which results of the investigation are compiled, is approved in the Fusion Council.

4. Member (draft)

(1) The Fusion Council nominates a chairman of the subcommittee.
(2) The member of subcommittee is listed in the attachment.
(3) If needed, people of experience or academic standing can participate by decision of the chairman.

5. Others

(1) The chairman has an obligation to report properly the status of the investigation and deliberation in subcommittee to the Fusion council.
(2) Office of Fusion Energy, Research and Technology Division, Atomic Energy Bureau, Science and Technology Agency provides support for the subcommittee.
(3) The other issues related with the subcommittee are decided by the chairman in consultation with the subcommittee.

(Reference)

Six Investigation Issues pointed out in
the report entitled 'Discussion points and further consideration in the Special Committee on ITER Project'
(extracts from the report)

(1) Survey of long-term demand and supply of energy sources.

Without being prejudiced in favor of a specific industrial field and life style based on specific values, consider possible situations as wide as possible, investigate the demands for each situation, and show the possibility of the supply for each situation.

(2) Feasibility study of alternative energy sources.

Show political options, which are not just anticipation, that strengthen the prospects of alternative energy sources and consider possible policies: investing in research, promoting industries and so on.

(3) Technical feasibility of fusion energy.

Study the feasibility of fusion energy as a safe and promising energy source from the viewpoint of technological potential, management ability and characteristics of the industrial structure in Japan. This work should be performed with active participation of the industry.

(4) Extension of the fusion program and basic supporting research.

Make a comprehensive plan toward the realization of fusion energy, including the required role and cooperation structure of universities and industry, concerning basic research in various fields such as advanced reactor and material development, education and personnel training, which will support the ITER project and fusion reactor development following ITER over a long period of time if Japan becomes the ITER host country.

(5) Distribution of resources for research.

There are extremely large numbers of basic research fields that require public expenditure. On the other hand, what determines the final distribution is nothing else but policy.

Identify principles for the distribution. Especially, trade-offs between 'expansion of front line'-type research and 'sustaining humankind' -type research are important, and these relations are not always contrary to each other. Create a philosophy for resource distribution that fits the present day when the importance of the latter will be increased while identifying clear fundamental relationships between them.

(6) International relations.

In sharing responsibility for international collaboration, there are various options depending on the circumstances of the project. Aim at establishing fundamental guidelines on sharing of responsibility in a practical project. The argument of Mega Science Forum in OECD/CSTP will be a reference for this work.

Attachment

Members of Subcommittee of Fusion Council for Fusion Development Strategy (draft)

Nobuyuki Inoue (Chairman)	Chairman of Fusion Council
Katsunori Abe	Professor, Department of Engineering, Tohoku University
Kunihiko Okano	Research Fellow, Komae Research Laboratory, Central Research Institute of Electric Power Industry
Yuichi Ogawa	Assistant Professor, Department of Engineering, University of Tokyo
Yuji Kanaki	Director, Department of Development, Japan Atomic Industrial Forum, INC
Mitsuru Kikuchi	General Manager, Plasma Analysis Division, Japan Atomic Energy Research Institute
Shigetada Kobayashi	The Representative of the Committee on Nuclear Fusion Research & Development Japan Electrical Manufacturers' Association (Senior Manager, Advanced Energy Development Department, Energy System Group, Toshiba Corporation)
Satoru Tanaka	Professor, Department of Engineering, University of Tokyo
Masami Fujiwara	Director, National Institute of Fusion Science
Shinzaburo Matsuda	Director, Department of Fusion Engineering Research, Japan Atomic Energy Research Institute
Kenzo Miya	Chairman of Planning and Promotion Subcommittee under Fusion Council

(11 members)

Members of Subcommittee of Fusion Council for Fusion Development Strategy
(April 2000)

Nobuyuki Inoue (Chairman)	Chairman of Fusion Council Professor, Institute of Advanced Energy, Kyoto University
Katsunori Abe	Professor, Graduate School of Engineering, Tohoku University
Kunihiko Okano	Research Fellow, Komae Research Laboratory, Nuclear Energy Systems Department, Central Research Institute of Electric Power Industry,
Yuichi Ogawa	Professor, High Temperature Plasma Center, University of Tokyo
Mitsuru Kikuchi	General Manager, Tokamak Program Division, Department of Fusion Plasma Research, Japan Atomic Energy Research Institute
Shigetada Kobayashi	Chairman of the Committee on Nuclear Fusion Research & Development, Nuclear industry Executive Committee Japan Electrical Manufacturers' Association (Senior Manager, Advanced Energy Design & Engineering Department, Power Systems & Services Company, Toshiba Corporation)
Satoru Tanaka	Professor, Department of Quantum Engineering and Systems Science, Graduate School of Engineering, University of Tokyo
Yoshiaki Hirotani	Manager, Department of Project Planning and Promotion, Japan Atomic Industrial Forum, Inc)
Masami Fujiwara	Director-General, National Institute of Fusion Science
Shinzaburo Matsuda	Director General, Naka Fusion Establishment, Japan Atomic Energy Research Institute
Kenzo Miya	Chairman of Planning and Promotion Subcommittee under Fusion Council (Professor, Graduate School of Engineering, University of Tokyo)

Date of Subcommittee

1st Meeting	June 29,	1998	13th Meeting	June 25,	1999
2nd Meeting	July 16,	1998	14th Meeting	July 30,	1999
3rd Meeting	July 30,	1998	15th Meeting	August 27,	1999
4th Meeting	August 25,	1998	16th Meeting	September 27,	1999
5th Meeting	October 1,	1998	17th Meeting	October 25,	1999
6th Meeting	January 14,	1999	18th Meeting	November 8,	1999
7th Meeting	February 8,	1999	19th Meeting	November 22,	1999
8th Meeting	March 1,	1999	20th Meeting	December 9,	1999
9th Meeting	March 18,	1999	21st Meeting	December 16,	1999
10th Meeting	April 5,	1999	22nd Meeting	January 6,	2000
11th Meeting	April 19,	1999	23rd Meeting	February 22,	2000
12th Meeting	June 3,	1999	24th Meeting	April 3,	2000
			25th Meeting	April 26,	2000