1. JT-60SA Mission and Program

1.1 Introduction

Realization of fusion energy requires long-term research and development. A schematic of fusion energy development is shown in Fig. 1.1-1. Fusion energy development is divided into 3 phases before commercialization. The large Tokamak phase achieved equivalent break-even plasmas in JET and JT-60 and significant DT Power productions in TFTR and JET. A programmatic objective of the ITER phase is demonstration of scientific and technical feasibility of fusion energy. A primary objective of the DEMO phase is to demonstrate power (electricity) production in a manner leading to commercialization of fusion energy.

The fast track approach to fusion energy is to shorten its development period for fusion energy utilization by adding appropriate programs (BA program) in parallel with ITER. Program elements are advanced tokamak/simulation studies and fusion technology/material development.

![Fig. 1.1-1 Schematic of fusion energy development](image1)

To specify program elements needed in parallel with ITER, we have to identify the concept of DEMO. Typical DEMO concepts of Japan and EU are shown in Fig.1.1-2. Although size spans widely, operation mode is unanimously “steady-state”. Ranges of the normalized beta are pretty close each other, $\beta_N=3.5$ to 4.3 for JA DEMO and 3.4 to 4.5 for EU DEMO.

![Fig. 1.1-2 Cross section and parameters of JA-EU DEMO studies](image2)

The neutron wall load of DEMO exceeds that of ITER ($P_{n, ave}^w=0.57\text{MW/m}^2$) by a factor of 3-6. The neutron fluence of DEMO (order of $10\text{MWa/m}^2$) is larger than that of ITER ($0.3\text{MWa/m}^2$).
Neutron damage of first wall is important for DEMO. There are good candidate materials for such high flux and fluence such as F82H in Japan and EUROFER in EU. These materials showed good characteristics for low energy fission neutron irradiation. It is necessary to confirm its properties in 14MeV neutron environment while the use of the ferritic steel in front of high performance plasmas should be checked in a reactor relevant plasma environment.

The blanket development is also important for DEMO. It is quite important to decide whether the blanket structure can have a stabilizing effect on RWM since minimization of the plasma-blanket coupling is required for structural soundness against disruption force even in DEMO. If the stabilizing shell should be located far from the plasma, stabilization of higher n RWM may not be practical.

The divertor PFC development is also an important element for DEMO to handle high heat and particle fluxes incident to its surface. It is believed that carbon-based materials may not be usable in the DEMO environment. If so, a metal-based divertor must be developed and tested in the reactor relevant tokamak configuration. Otherwise, a usable condition of carbon-based PFC must be identified and tested in the reactor relevant tokamak configuration.

Steady state and high beta are important features of DEMO concepts. However, the achievement of steady state operation with $Q > 5$ in ITER requires further physics R&D in other tokamaks. Sustainment of high beta above the no-wall limit has not been demonstrated in large tokamaks and not foreseen in the early phase of the ITER operation. Confirmation and expansion of the steady state operation, and exploration of the high beta operation for DEMO are important missions of the satellite tokamak.

As for configuration optimization of DEMO, the advanced tokamak configuration not foreseen in ITER could be explored for concept development of DEMO in the satellite tokamak.

ITER is an integrated test bed for DEMO relevant fusion plasma science and technology. But the application of various ideas to ITER is difficult. Especially, any risky idea not proven in a break-even class tokamak would be difficult.

In the Large Tokamak phase, there were three devices and various ideas were tested and compared each other. The role of Satellite Tokamak in providing confidence of any idea for application to ITER is quite important. Also ITER would require physics investigation in well-diagnosed devices if ITER encounters any difficulty to fulfill its missions. Such research is called ITER support program.

The Satellite Tokamak Program is a joint Program of Japan and EU under the BA Program using JT-60 facility modification shared with the JAEA’s program for national use. The scientific missions of satellite tokamak are described in Section 1.2.

### 1.2 Objectives and Mission of Satellite Tokamak Program

The objectives of the Satellite Tokamak Program are,

a) the participation in the upgrade of the JT-60 owned by JAEA to a superconducting tokamak JT-60SA; and
b) the participation in its exploitation to support the exploitation of ITER and the research towards DEMO by addressing key physics issues for ITER and DEMO.

The main missions in support of ITER are to optimize operation scenarios for ITER, to optimize ITER auxiliary systems, which come later in the construction of ITER, to train, in an international environment, scientists, engineers and technicians in view of integrated operation and scientific exploitation of ITER.

The main mission during ITER operation are, to support further development of operating scenarios and the understanding of physics issues, and to test possible modifications before their implementation on ITER.

The main missions in support of DEMO are to explore operational regimes and issues complementary to those being addressed in ITER. In particular these will include:

- steady state operation
- advanced plasma regimes (higher normalized $\beta$)
- control of power fluxes to walls.
1.3. JT-60SA Facility

1.3.1. Outline of JT-60SA Facility

JT-60SA is designed consistently with maximum utilization of existing JT-60 facilities such as plasma heating and current drive systems, power supplies, diagnostics and cooling system.

Figure 1.3.1-1 shows a bird's eye view of the JT-60SA tokamak with heating and current drive systems. A cross sectional view of the tokamak is shown in Fig. 1.3.1-2. Plasma with aspect ratio down to 2.6 can be formed to determine the optimum plasma shape for a cost effective DEMO reactor [1.3-1]. Major components of tokamak will be installed inside the spherical cryostat with a diameter of about 14 m for thermal shielding of superconducting magnets. The maximum plasma current is 5.5 MA for a full-bore (~127 m³) plasma with double-null or single-null divertor and 3.5 MA for an ITER-like shaped plasma with single null divertor as summarized in Table 1.3.1-1. The divertor geometry was optimized to produce both high triangularity double-null/single-null plasmas and ITER-like shaped single-null plasmas in one machine. A semi-closed vertical divertor with a flat dome was adopted to maintain high flexibility for plasma shaping. The divertor geometry will reflect further investigation.

Heating and current drive (H&CD) systems are upgraded from those reported in Refs. [1.3-2, 1.3-3] to enable power densities in excess of ITER and to allow independent control of heating, current and rotation profiles, which are essential for expanding advanced tokamak regime. The high power plasma heating results in a remarkable increase of DD neutron yield to about $2 \times 10^{19}$/shot. Significant efforts were made in the design of radiation shield for TF coils, vacuum vessel and cryostat structure. Borated acid water is introduced into the double walled space of vacuum vessel to reduce nuclear heating in TF coils. The space between the double walls of the cryostat was filled with boron doped concrete for bio-shielding of the torus hall. However, remote handling is considered to be absolutely necessary for maintenance and repair of in-vessel

![Fig. 1.3.1-1. Bird's eye view of JT-60SA with heating and current drive systems.](image)
devices such as divertor modules and first wall, because the expected dose rate at the vacuum vessel may exceed 1 mSv/hr after 10 years operation and three months of cooling.

The increased heating power has also affected the design of divertor targets and cooling system significantly.

Figure 1.3.1-3 shows the layout of the JT-60SA torus hall showing major components. Four large ports are devoted to remote handling devices. In addition, four beam towers of perpendicular positive ion based NBI (P-NBI), two beam tanks of tangential P-NBI, and a negative ion based NBI (N-NBI) located in the assembling hall will be reused. The ECH waveguides will be introduced from a neighboring room as illustrated. Current lead box, valve boxes and other major equipments are located near the cryostat. Plasma operation is planned mainly during the day time, because the regeneration of divertor cryo-panel is required every night in order to cope with high plasma density operation.

Each component of the JT-60SA facility is described briefly in the following subsections.
1.3.2. Superconducting Magnets

The superconducting magnets of JT-60SA consist of 18 toroidal field (TF) coils and 11 poloidal field (PF) coils, as shown in Fig. 1.3.1-2.

NbTi conductor is used for the TF coils because the maximum field strength is ~6.4 T, and it is expected to produce the vacuum toroidal field as indicated in Table 1.3.1-1. The choice of the final conductor design will be made on the basis of assuring superconducting stability, manufacturing feasibility, optimal compatibility with the power supply, and the cost.

PF coils consist of the Central Solenoid (CS) made of four identical coils (CS1, CS2, CS3, CS4) and seven equilibrium field (EF) coils (EF1, EF2, ..., EF7). NbTi conductor is used for all EF coils including the divertor coils (EF3 and EF4), because their maximum field strength at the conductor surface is ~6.1 T, under the assumption of uniform current distribution. Two kinds of NbTi conductor design are anticipated for the EF coils, because the maximum magnetic field strength is quite different between the large diameter outer ring EF coils (EF1, EF2, EF5, EF6, EF7) and smaller diameter ring divertor coils (EF3, EF4). For CS coils, Nb3Sn conductor is expected to be used because the maximum field strength is in the range of 9-10 T to ensure a flux swing capability of about 40 Wb, which is necessary to sustain the rated plasma current for 100 s.

All TF coils are planned to be cooled down and power tested in advance to shipment for the quality control. Similar tests are also planned for the CS units. All necessary data required by the Japanese High Pressure Regulation Law should be collected and kept.

1.3.3. Vacuum Vessel

The vacuum vessel is made of double-walled 316L stainless steel with low Co content of less

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Large Plasma (DN)</th>
<th>ITER Similar (SN)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plasma Current Ip (MA)</td>
<td>5.5</td>
<td>3.5</td>
</tr>
<tr>
<td>Toroidal Field Bt (T)</td>
<td>2.68</td>
<td>2.6</td>
</tr>
<tr>
<td>Major Radius (m)</td>
<td>3.06</td>
<td>3.15</td>
</tr>
<tr>
<td>Minor Radius (m)</td>
<td>1.15</td>
<td>1.02</td>
</tr>
<tr>
<td>Elongation, κ95</td>
<td>1.76</td>
<td>1.69</td>
</tr>
<tr>
<td>Triangularity, δ95</td>
<td>0.45</td>
<td>0.36</td>
</tr>
<tr>
<td>Aspect Ratio, A</td>
<td>2.66</td>
<td>3.1</td>
</tr>
<tr>
<td>Shape Parameter, S</td>
<td>5.5</td>
<td>4.0</td>
</tr>
<tr>
<td>Safety Factor q95</td>
<td>3.1</td>
<td>3.06</td>
</tr>
<tr>
<td>Flattop Duration</td>
<td>100 s (8 hours)</td>
<td></td>
</tr>
<tr>
<td>Heating &amp; CD power</td>
<td>41 MW x 100 s</td>
<td></td>
</tr>
<tr>
<td>N-NBI</td>
<td>34 MW</td>
<td></td>
</tr>
<tr>
<td>ECRH</td>
<td>7 MW</td>
<td></td>
</tr>
<tr>
<td>PFC wall load</td>
<td>15 MW/m²</td>
<td></td>
</tr>
<tr>
<td>Neutron (year)</td>
<td>4 x 10^{21}</td>
<td></td>
</tr>
</tbody>
</table>
than 0.05 wt% to minimize induced activation. One turn resistance of the vacuum vessel is \(\sim 13.5 \mu\Omega\) (L/R = \(-0.14\) s) including contribution from stabilizing plates. The double-wall vacuum vessel is filled with borated acid water to enhance the neutron shielding capability of the vacuum vessel.

The gravity support will be attached to every 20 or 40 degree section with spring plates at the bottom of the vacuum vessel. Connecting plates are introduced between the neighboring vertical ports and/or gravity supports to increase the stiffness against the twisting moment and seismic forces. An 80 K thermal shield will be placed between the vacuum vessel and the cold structures of superconducting magnets. The vacuum vessel can be baked at 200 °C by heated nitrogen gas. There is no doubt that plasma operation during vacuum vessel baking is preferable for wall conditioning, but it is not being planned because the neutron shielding capability would be lost during the vacuum vessel baking using nitrogen gas. In addition, it is not acceptable from the viewpoint of an excessive heat load to the cryogenic system. Also it must be taken into account that the temperature of in-vessel components such as divertor cassette should be around 40 °C.

A feasibility study is ongoing considering the stiffness of the vacuum vessel and gravity supports, and the performance of the silver coated 80 K thermal shield.

1.3.4. In-vessel Components Including Divertor Cassette Modules

In-vessel components include the divertor cassette, inboard first wall, stabilizing baffle plate, position control coils, and sector coils for RWM control.

The upper and lower divertor cassettes are asymmetric to allow formation of plasmas with ITER-like shape and high triangularity plasmas. A mono-block type CFC divertor armor is used for the outboard divertor armor to withstand the heat load of 15 MW/m², while flat-tile type divertor armor is used for the inboard armor because the expected heat load is about half of that on the outboard side. There is another design option in which both inboard and outboard divertor armors use the same mono-block type CFC armor, in order to reduce R&D items. It must be decided before call for tender based on the cost estimation.

CFC divertor armor will be connected to the divertor cassette in a way compatible with remote handling. The cryo-panels can be installed under the flatter dome and/or the outer baffle plate to ensure strong pumping. Placement of the cryo-panel into the vertical port like in ITER is another option. On the basis of the divertor performance simulation, the location of the cryo-panels will be finally decided considering the core and SOL plasma performances.

Two in-vessel field coils will be used for a fast plasma position control. In order to control vertical and horizontal field at the same time using two coil blocks, independent power supplies will be connected to each coil block. A large induced current is foreseen in the fast plasma position control coils, when plasma disrupts with a vertical motion such as VDE. To suppress the excessive current and electromagnetic forces, insert of external inductors in series to the coils and power supplies might be required. This would degrade the control response time so that an optimization between structural coil design and plasma feedback control system should be carried out.

The primary function of the stabilizing baffle plate is to enhance the ideal beta limit. It can be
baked up to 200 °C at least like the vacuum vessel.

The 18 sector coils for RWM control will be installed around the openings of stabilizing baffle plates to maximize the magnetic coupling with the plasma. Since each sector coil has its own current lead, the magnetic field mode of m/n = 3/1 or 3/2 can be excited according to the power supply connection.

1.3.5. Cryostat

The cryostat is a spherical vessel of almost 14 m diameter made of double-walled SS304 filled with boron doped concrete. The inner skin is the structure that supports the weight of all the ports and withstands the vacuum pressure. The thin outer metal skin has the advantage to decrease the activation level outside the cryostat. Boron effectively reduces thermal neutron flux and hence the activation of Ar in the torus hall air.

1.3.6. Remote Handling

Large annual neutron fluence of 4x10^{21} neutrons/year nearly prohibits human access inside the vacuum vessel after an extensive experimental campaign. Therefore, most in-vessel components need to be compatible with remote handling. The remote handling system is a vehicle type system adopted in ITER with a possibility to use its rail as a boom. Four large horizontal ports are used to extend and support a rail and bring in/out maintained components.

The large foreseen annual neutron fluence of 4x10^{21} neutrons/year nearly prohibits human access inside the vacuum vessel after an extensive experimental campaign. Therefore, most in-vessel components need to be compatible with remote handling. The presently planned remote handling system is a vehicle type system such as adopted in ITER for the shielding blanket, with the possibility to use its rail as a boom. Four large horizontal ports are used to extend and support a rail and bring in/out maintained components as shown in Fig. 1.3.1-3. The remote handling system will be normally in use for operator training at a separate place, and will be brought into the torus hall during maintenance period.

The expected functions of the remote handling system of JT-60SA are as follows: repair and exchange of divertor cassette, repair of first wall armor for the inboard side and the baffle plates, and small repair of beam limiters inside of NBI ports if possible. The weight of one divertor cassette is about 500 kg, while one first wall armor is several kg. Then two kinds of remote handling system are now being planned to cope with heavy and light loads.

1.3.7. Heating and Current Drive System

The H&CD system consists of positive-ion based NBI (P-NBI) with a beam energy of 85 keV, negative-ion based NBI (N-NBI) with a beam energy of 500 keV and two ECRF systems (110 GHz and 140 Ghz).

Eight perpendicular P-NBI units provide 16 MW of heating power with little current and rotation drives. Two co-tangential (4 MW) and two counter-tangential (4 MW) P-NBI provide flexibility of rotation and current control. A 500 keV co-tangential N-NBI (10 MW) is a unique feature of JT-60SA. This system provides efficient current drive and produces energetic particles for Alfven eigenmode studies. The N-NBI beam line is lowered to 0.6 m below the mid-plane for
off-axis current drive.

The design of upgraded NBI system is made based on the modification for 30 s operation in the present NBI system on JT-60U, where the long pulse operations of 30 s at 2 MW and 20 s at 3.2 MW have been achieved on the P-NBI and N-NBI units, respectively. The main modification is the upgrade of the power supply system. In addition, the negative ion source should be modified to improve the performance such as a voltage holding capability. Another key issue is to shield the stray magnetic field on the beamline from the JT-60SA tokamak, which will be about three times larger than JT-60U.

The ECRF system will be used for electron cyclotron heating and current drive, assistance to plasma start-up, and cleaning of the first wall of the vacuum vessel. Preparation of ECRF system with two frequencies enables effective heating and/or current drive for a range of toroidal field, $B_t = 1.5$-$2.7$ T. The 110 GHz system consists of four Gyrotrons to deliver 3 MW into the torus, while the 140 GHz system consists of five Gyrotrons to deliver 4 MW into the torus. Four antennas (two for 110 GHz and two for 140 GHz) will be installed through upper perpendicular ports.

The actual composition of the 110GHz system, made by Japan (JA), will be basically similar to the present JT-60U ECRF system (3 MW x 5 s at 110 G Hz with 4 gyrotrons). The components of the JT-60U ECRF system will be reused in the 110GHz system as much as possible. For the maximum pulse duration of 100 s, four gyrotrons and two main power supplies should be newly fabricated and two main power supplies should be upgraded. Many components of the transmission lines will be reused, though the cooling and evacuating systems will be upgraded for extended pulse duration. Launchers should be developed which can control the injection angle of RF beams both in the poloidal and toroidal directions.

Most of components of the 140GHz system will be fabricated newly except for the AC power lines. Its composition is the same as the 110GHz system. Five DC power supply systems and three gyrotron sets will be provided by EU while the two others by JA. As the transmission lines will be newly fabricated, wider waveguide components with the diameter of 63.5 mm will be used. So higher transmission efficiency of 80 % is expected and more stable high power transmission will be carried out at 1 MW levels. Launchers with functions similar to those of the 110GHz launcher have also to be developed.

1.3.8. Fuelling System

A gas puffing system and a pellet injection system are prepared for particle fuelling, in particular for high density operation ($\sim 1 \times 10^{20}$ m$^{-3}$) relevant to ITER operation regime. The gas puffing system consists of four injection lines capable of different gases (deuterium/hydrogen, helium and impurity gases). Multilayer piezoelectric actuator valves will be used. The expected fuelling rate is 90 Pa•m$^3$/s for deuterium per line. Three pellet injectors will be installed. The injector consists of a screw type pellet extruder and a centrifugal type accelerator. The injection speed is 100-1000 m/s and the injection duration is up to 100 s. The fuelling rate is $\sim 8 \times 10^{21}$ /s for 10 Hz injection per one injector. Injection both from the high-field side and for the low-field side will be prepared.
1.3.9. Diagnostics

Comprehensive diagnostics will be installed. Most of them will be available from the initial operation of JT-60SA. Majority of diagnostics presently used on JT-60U will be reused after necessary modifications. Port assignments for diagnostics have been determined tentatively.

1.3.10. Water Cooling System

The present primary water cooling system has a continuous heat removal capability of about 10 MW for in-vessel components. Since the maximum plasma heating power is 41 MW for 100 s, a large water tank of 360 m$^3$ and powerful pumps are introduced to deliver cooling water to in-vessel components including the divertor modules at the maximum flow rate of 4800 m$^3$/h at the pressure of 2.5 MPa for more than 100 s. It will be installed in the basement of JT-60 Torus Building.

A boron-water circulating system is also prepared for the boron water between the double wall of the vacuum vessel.

1.3.11. Cryogenic System

A cryogenic system is prepared to supply gas He (super-He, SHe) to superconducting magnets, and liquid He (LHe) to cryo-panels and pellet injectors. It consists of He liquefaction and refrigeration system, SHe circulation system, LN2 system, recovery and refining system, data acquisition and control system and vacuum pumping system. The requirement for refrigerator is tentatively specified to be 16 kW.

1.3.12. Supervisory Control System and Data Acquisition System

The existing supervisory control and data acquisition system in JT-60U will be modified for JT-60SA. The control system for JT-60SA will be optimized for real time profile control that is essential for sustainment of advanced tokamak plasmas for a long time. Major points of difference with respect to the existing system are: (a) Discharge duration time has to be extended to 200 s and longer, and (b) Superconducting poloidal and toroidal coils have to be well controlled. We should create and develop advanced real-time plasma control methods including controls of superconducting coil power supplies, a new scheme of data acquisition, a new concept of experimental database management, a long-pulse integrator for magnetic measurements, a new timing generation system, etc.

1.3.13. Power Supplies

New power supply systems will be designed and manufactured for the superconducting toroidal field (TF) coils and poloidal field coils. Major components for a coil power supply commonly consist of a transformer, thyristor-converters, a quench protection circuit, a crow-bar switch, instruments for measurements, and a controller. In addition, some power supplies have DC circuit interrupters.

Among additional heating devices P-NBI, N-NBI, and ECH, (41 MW, 100 s plasma heating), the P-NBI (60 MW) and ECH (30 MW) will be connected to the 275 kV commercial power grid, together with TF coil power supply. Discussion on higher harmonic disturbance, and power
factor in the operation of plasma discharge, has determined the specifications for the filters and shunt capacitors. The N-NBI (40 MW) and PF coil power supplies (30 MW) are connected to two existing motor generators (4 GJ, and 2.6 GJ)

References
1.4. Operating Scenarios

1.4.1 Daily Operation Scenario

The JT-60SA experimental time is planned to be 10 hours per day without meal break. The experiment will be conducted by an operation team in two shifts, including one hour of pre- or post-experimental inspection, respectively. Regeneration of cryo-panels for fuel pumping at the top and bottom divertor, wall conditioning by glow discharge cleaning and cooling of cryo-panels for the next day experiment are carried out overnight by an operation team for the cryogenic system operating on three shifts. The discharge with current flat-top duration of 100 s is planned to be repeated every ~30 minutes during the daily run time.

1.4.2 Annual Operation Scenario

Typical annual operation and maintenance periods are shown in Fig. 7-1, where five experimental cycles and commissioning operation are planned for the annual experimental run; here, each experimental cycle is defined as 3-weeks of experiments. Thus, the experimental period is about 7 months, including commissioning operation, short maintenance of facilities and diagnostics between experimental cycles. The maintenance period is planned in summer in order to save electricity.

Prior to the operational shut down for maintenance, the exhaust operation of tritium gas for about one month and warming up of the superconducting coils for about one month are planned. The annual maintenance period is typically about 2.5 months, where annual inspections of JT-60SA facilities are planned according to safety regulations for electricity, high pressure gas, cranes etc., maintenance and adjustment of diagnostic systems, and installations of new facilities or diagnostic systems. In addition, inspection or repair of in-vessel components is planned using the remote handling systems. After the maintenance period, preparation of the cryogenic system and cooling down time for superconducting coils takes about one and half months. After the completion of the cooling down of the superconducting coils, the JT-60SA operation restarts for commissioning and experiments.

![Fig. 1.4.2-1. Typical annual experiment and maintenance plan of JT-60SA](Fig_1.4.2-1.png)

1.4.3 Three-Year Operation Plan (tentative)

In the 8th year of the BA period, the JT-60SA will start commissioning and operating. The initial cooling and testing of superconducting coils are planned for about three months. Then, the first operation for about three months is planned including wall conditioning, control tests, NBI
commissioning and diagnostic adjustments etc. After these commissioning and adjustments, the superconducting coils are warmed up and in-vessel inspection is planned. After the in-vessel inspection, the coils are cooled down and the experimental run is continued for about two months until the end of fiscal year. In the 9th and 10th of the BA period, the nominal five cycle operation is planned for experiments as shown in Figure 7-1.

1.5 Overview of Naka Site

1.5.1 Outline of JAEA Naka Fusion Institute

JAEA Naka Fusion Institute, situated on grounds covering an area of approximately 130 ha, was established as the former JAERI Naka Fusion Research Establishment in 1985 and is one of the largest nuclear fusion research centers in the world. In an effort to realize a permanent energy source for human kind, JAEA Naka Fusion Institute is promoting the world's top class scientific and technical research on nuclear fusion using the large tokamak JT-60 and a variety of nuclear fusion experimental facilities.

1.5.2 Land and Site Layout

JT-60 is located in JAEA Naka Fusion Institute. In 1979, this land was handed over by Ibaraki Prefecture to the former JAERI as a site for nuclear fusion research with 60 ha of land required for construction of the planned JT-60. The present layout of the JT-60 site is shown below.

Fig.1.5.1-1 Layout of the JT-60 site in JAEA Naka Fusion Institute
1.5.3 Heat Removal and Water Supply

Public water supply and distribution facilities to the JT-60 site have already been constructed and water supply can be received from Naka Municipal Water Department. Water supply and distribution capacity is 320 m$^3$/day.

Industrial water supply and distribution facilities to the JT-60 site have already been constructed and water supply can be received from the Ibaraki Corporations Bureau Naka River industrial water supply system. The water supply and distribution capacity of this system is 5,000 m$^3$/day.

1.5.4 Wastewater

Miscellaneous and industrial wastewater from the JT-60 site, after undergoing the necessary treatment, can be discharged into the ocean via the exiting drainage pipeline (8,000 t/day, installed length approximately 11 km) installed in JAEA Naka Fusion Institute.

1.5.5 Electric Power Supply

Since the JT-60 site is located within the supply area of Tokyo Electric Power Co., Inc., existing infrastructure including transmission line conduits and power supply facilities, etc. can be utilized. At JAEA Naka Fusion Institute, transmission lines (275 kV system, 2 lines) that branch off (T-junction) from the transmission system starting at Tokai Nuclear Power Plant, Unit #2 are installed.

![Power supply facilities for JT-60](image-url)
1.5.6 Transportation of Large Heavy Objects

Since the JT-60 site is situated inland roughly 6 km away from the coast, in order to transport large and heavy objects to the site, it is necessary to have port facilities for landing and roads for overland transportation. Concerning these facilities, technical investigation has shown that existing infrastructure can be used almost as is.

For example, Hitachinaka International Port is one of Japan's core international ports. It has good access to Kitakanto Expressway and Joban Expressway and was opened in April 2000 as a high-grade state-of-the-art container port having the latest-scale container terminal capable of handling large container ships like the Over Panamax (50,000-60,000 D/W) and berths with depths of 14 m. It is also possible to carry out customs clearance and other necessary procedures at designated bond facilities at Hitachinaka International Port if necessary.

Concerning routes for transporting materials from Hitachinaka International Port to the JT-60 site, it may be necessary to remove some overhead obstacles (telephone lines, power lines, signs, streetlights, etc.) that are 10 m or less above the vehicle, make partial improvements to intersections, and carry out partial reinforcement and renovation of bridges.

1.5.7 Access

Naka site is located within the Mito-Hitachi urban area in Ibaraki Prefecture, which is the geological center of Japan. It is approximately 100 km north-east of Tokyo and is part of the Greater Tokyo metropolitan area. Tokyo is the capital and the central node of various transportation connections in Japan. It has Haneda Airport, which is the largest domestic airport, Tokyo station, which is the terminal station for Shinkansen and other major railways.

Naka site is connected to Tokyo by expressways and super-express trains. Tokyo can be reached from Naka site in just over an hour by super express trains departing every 30 minutes on the JR Joban Line, in approximately 1.5 hours by car via Joban Expressway from the Naka ramp, and in approximately 1.5 hours by express bus which runs every hour. As for international connections, the New Tokyo International Airport at Narita is within two hours of reach by train and also by direct bus service, which runs every hour during the daytime.

1.6 Possible Joint Scientific Program

The JT-60SA device described above is a powerful and flexible device, which incorporates various research elements to support and supplement ITER towards DEMO. This program could be a great opportunity for both JA and EU:

(1) To fulfill the first part of the Satellite Tokamak mission, namely to support ITER by developing an improved understanding of physics issues, optimizing operation scenarios, testing possible future modifications and training scientists, engineers and technicians.

(2) To address the second part of the Satellite Tokamak mission, namely to pursue an integrated exploration of steady state, high beta DEMO relevant scenarios with adequate power and particle control.
Given the fact that construction is expected to take 7 years, only 3 years would remain within the 10 years of the BA period for exploitation including integrated cold test and commissioning of heating and current drive systems. This period is too short to fulfill these important and ambitious missions.

An exploitation period of at least 10 years is necessary to fulfill the missions related to 100s pulses.