World Survey of Fusion Devices 2022

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International Atomic Energy Agency

WORLD SURVEY OF FUSION DEVICES
2022

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

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2022

INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

The IAEA fosters discussion on nuclear fusion research and development advancing extensive international dialogue to overcome highly technical challenges and make fusion energy a reality. The IAEA is at the forefront of worldwide efforts to make fusion energy production a reality, facilitating international coordination and sharing best practices in global projects.

In 2020 the IAEA released its first on-line database of fusion devices, the Fusion Device Information System (FusDIS). FusDIS contains information on public and private fusion devices with experimental and demonstration designs currently in operation, under construction or being planned, as well as the technical data of these devices.

This publication compiles and further elaborates on the information available on FusDIS, providing an up-to-date worldwide survey of experimental fusion devices, testing facilities and plant designs.

The IAEA gratefully acknowledges the International Fusion Research Council for contributing to this publication. The IAEA officer responsible for this publication was M. Barbarino of the Division of Physical and Chemical Sciences.

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1. INTRODUCTION

1.1.BACKGROUND

Nuclear fusion is the process by which two light atomic nuclei combine to form a single heavier one while releasing massive amounts of energy.

Fusion reactions take place in a state of matter called plasma – a hot, charged gas made of positive ions and free-moving electrons with unique properties distinct from solids, liquids or gases.

The sun, along with all other stars, is powered by this reaction. To fuse in our sun, nuclei need to collide with each other at extremely high temperatures, around ten million degrees Celsius. The high temperature provides them with enough energy to overcome their mutual electrical repulsion. Once the nuclei come within a very close range of each other, the attractive nuclear force between them will outweigh the electrical repulsion and allow them to fuse. For this to happen, the nuclei need to be confined within a small space to increase the chances of collision. In the sun, the extreme pressure produced by its immense gravity creates the conditions for fusion.

Ever since the theory of nuclear fusion was understood in the 1930s, scientists – and increasingly also engineers – have been on a quest to recreate and harness it. That is because if nuclear fusion can be replicated on earth at an industrial scale, it could provide virtually limitless clean, safe, and affordable energy to meet the world's energy demand.

Fusion generates four times more energy per kilogram of fuel than fission used in nuclear power plants, and nearly four million times more energy than burning oil or coal.

Most of the fusion reactor concepts under development will use a mixture of deuterium and tritium (or D-T) — hydrogen atoms that contain extra neutrons (Fig. 1). In theory, with just a few grams of these reactants, it is possible to produce a terajoule of energy, which is approximately the energy one person in a developed country needs over sixty years.

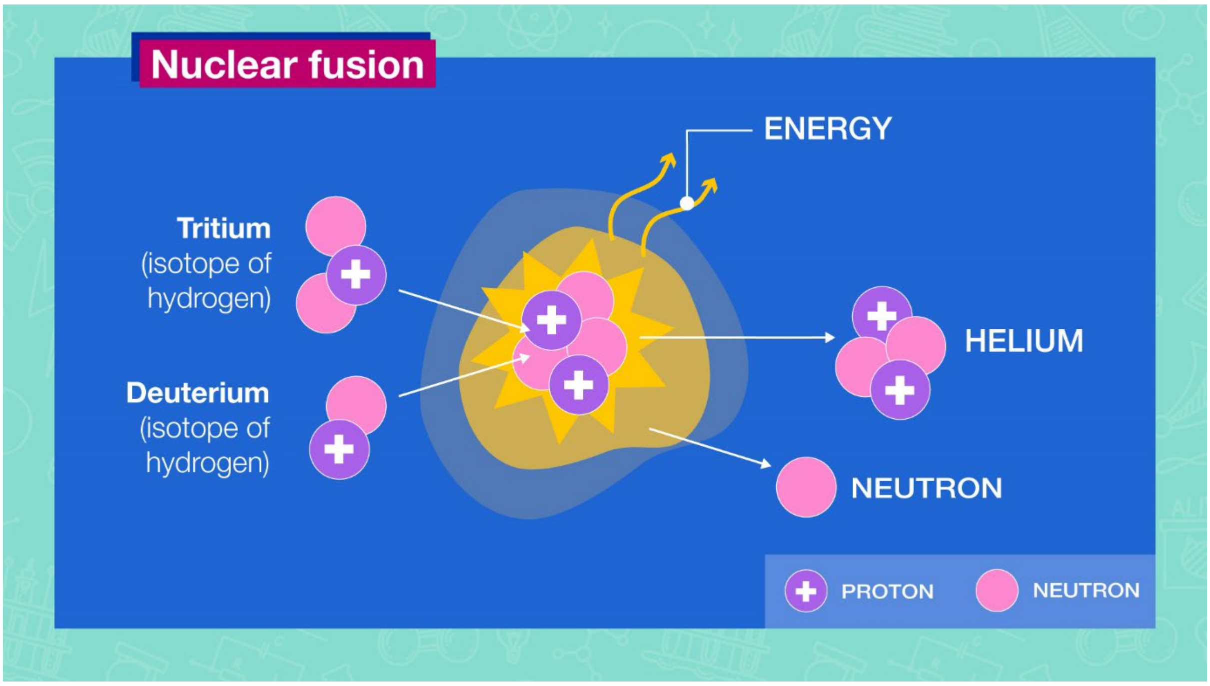

FIG.1. A mixture of deuterium and tritium – two Hydrogen isotopes – will be used to fuel future fusion power plants. Inside the reactor, deuterium and tritium nuclei collide and fuse, releasing Helium and neutrons.

Fusion fuel is plentiful and easily accessible: deuterium can be extracted inexpensively from seawater, and tritium can potentially be produced from the reaction of fusion generated neutrons with naturally abundant lithium. These fuel supplies would last for millions of years. Future fusion reactors are also intrinsically safe and are not expected to produce high activity or long-lived nuclear waste. Furthermore, as the fusion process is difficult to start and maintain, there is no risk of a runaway reaction and meltdown.

Importantly, nuclear fusion – just like fission – does not emit carbon dioxide or other greenhouse gases into the atmosphere, so it could be a long-term source of low-carbon electricity from the second half of this century onwards.

While the sun's massive gravitational force naturally induces fusion, without that force a temperature even higher than in the sun is needed for the reaction to take place. On Earth, we need temperatures of around 150 million degrees Celsius to make deuterium and tritium fuse, while regulating pressure and magnetic forces at the same time, for a stable confinement of the plasma and to maintain the fusion reaction long enough to produce more energy than what was required to start the reaction.

While conditions that are very close to those required in a fusion reactor are now routinely achieved in experiments, improved confinement properties and stability of the plasma are still needed to maintain the reaction and produce energy in a sustained manner. Scientists and engineers from all over the world continue to develop and test new materials and design new technologies to achieve net fusion energy.

Nuclear fusion and plasma physics research are carried out in more than 50 countries, and fusion reactions have been successfully produced in many experiments, albeit without so far generating more energy than what was required to start the process. Experts have come up with different designs and magnet-based machines in which fusion takes place, like stellarators and tokamaks (Fig. 2), but also approaches that rely on lasers, linear devices and advanced fuels.

How long it will take for fusion energy to be successfully rolled out will depend on mobilizing resources through global partnerships and collaboration, and on how fast the industry will be able to develop, validate and qualify emerging fusion technologies. Another important issue is to develop in parallel the necessary nuclear infrastructure, such as the requirements, standards and good practices, relevant to the realization of this future energy source.

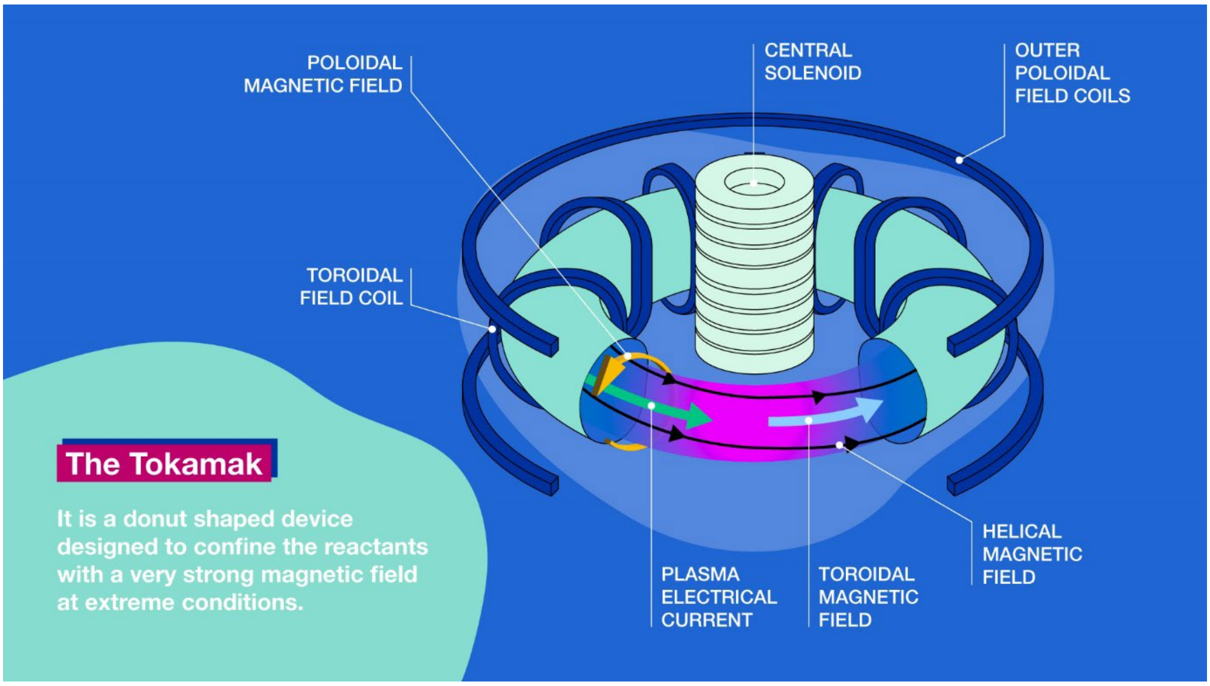

FIG.2. How a tokamak works: The electric field induced by a transformer drives a current (green horizontal arrow) through the plasma column. This generates a poloidal magnetic field that bends the plasma current into a circle (yellow vertical arrow). Bending the column into a circle prevents leakage and doing this inside a doughnut-shaped vessel creates a vacuum. The other magnetic field going around the length of the doughnut is referred to as toroidal (blue horizontal arrow). The combination of these two fields creates a three-dimensional curve, like a helix (shown in black), in which the plasma is highly confined. Twisting the magnets can also produce the helical shape without the need for a transformer – this kind of configuration is called a stellarator.

Following 10 years of component design, site preparation, and manufacturing across the world, the assembly of ITER in France, the world's largest international fusion facility, commenced in 2020. ITER (see p. 34) is an international project that aims to demonstrate the scientific and technological feasibility of fusion energy production and prove technology and concepts for future electricity-producing demonstration fusion power plants, called DEMOs (see pp. 143–153). ITER will start conducting its first experiments in the second half of the 2020s and full power experiments should commence in the second half of the 2030s.

DEMO timelines vary in different countries, but the consensus among experts is that an electricity producing fusion power plant could be built and operating by 2050. In parallel, numerous privately funded commercial enterprises are also making strides in developing concepts for fusion power plants, drawing on the know-how generated over years of publicly funded research and development, and proposing fusion power even sooner. Some of these private companies are pursuing concepts based on fusion reactions other than the D-T reaction. Traditionally, D-T reaction has been used to achieve fusion because they reach the highest reaction rate at a lower temperature than other fuels. However, tritium is radioactive and does not occur naturally in any significant quantities. Therefore, it has to be 'bred' in a nuclear reaction between the fusion-generated neutrons and lithium surrounding the reactor wall. The energy of these neutrons also presents significant challenges regarding the materials in the reactor vacuum vessel, since, when the neutrons collide with the reactor walls, its structures and components become radioactive. This necessitates considerations in radiation safety and waste disposal.

To bypass the challenges caused by the use of tritium, there are now experiments using alternative or advanced fusion fuels, like D-^{3}He or p-^{11}B. The list of the most favourable fusion reactions is given in Table 1. Boron-11 is non-radioactive and comprises around 80 percent of all boron found in nature, so it is readily available. However, the challenge with p-^{11}B fusion is that it would require the plasma to be a hundred times hotter than plasma containing deuterium and tritium.

TABLE 1. LIST OF THE MOST FAVOURABLE FUSION REACTIONS

Reactants	Products
D-T	^{4}He (3.5 MeV) + n (14.1 MeV)
D-D	T (1.01 MeV) + p (3.02 MeV) (50%)
	^{3}He (0.82 MeV) + n (2.45 MeV) (50%)
D-^{3}He	^{4}He (3.6 MeV) + p (14.7 MeV)
T-T	^{4}He + 2 n + 11.3 MeV
^{3}He-^{3}He	^{4}He + 2 p
^{3}He-T	^{4}He + p + n + 12.1 MeV (51%)
	^{4}He (4.8 MeV) + D (9.5 MeV) (43%)
	^{4}He (0.5 MeV) + n (1.9 MeV) + p (11.9 MeV) (6%)
D-^{6}Li	2 ^{4}He + 22.4 MeV
p-^{6}Li	^{4}He (1.7 MeV) + ^{3}He (2.3 MeV)
^{3}He-^{6}Li	2 ^{4}He + p + 16.9 MeV
p-^{11}B	3 ^{4}He + 8.7 MeV

Fusion research outputs have boomed over the last fifteen years since ITER was established, by looking the number of first authorship papers presented at the IAEA Fusion Energy Conference (FEC) between 2006 and 2021 (see Fig. 3). The United States of America (USA) retains the top position with 152 first authorship papers for 2021. Japan and China are in second and third place, respectively (see Fig. 4). The Princeton Plasma Physics Laboratory (USA) and the Max Planck Institute for Plasma Physics (Germany) are the leading organizations in this index, with a total of 42 first authorship papers in 2021 (see Fig. 5). The Institute for Plasma Research (India) and Southwestern Institute of Physics (China) are not far behind, with 37 and 35 first authorship papers in 2021, respectively.

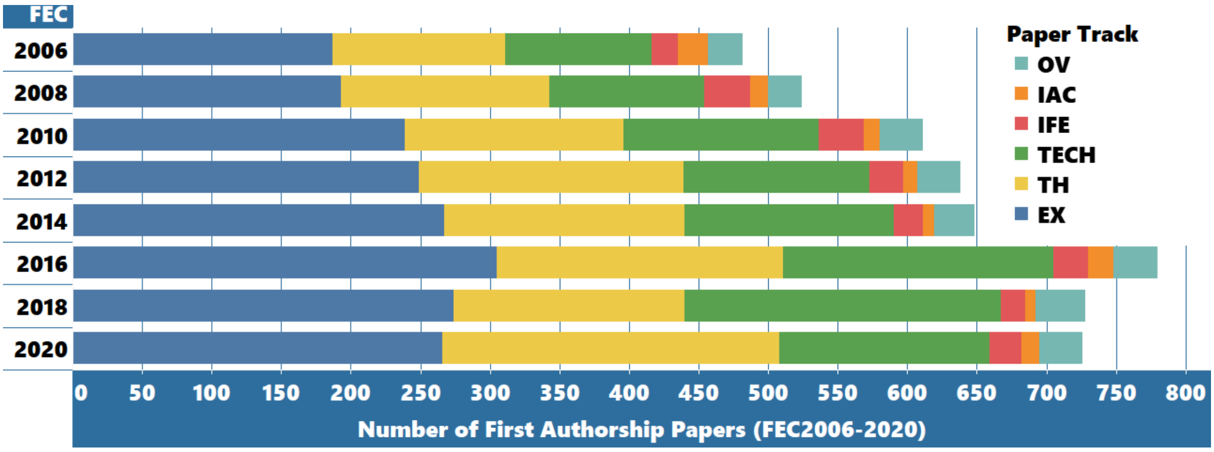

FIG.3. First authorship papers presented at the IAEA FEC between 2006 and 2021. Paper tracks are: EX - Magnetic Fusion Experiments; TH - Magnetic Fusion Theory and Modelling; TECH - Fusion Energy Technology; IFE - Inertial Fusion Energy; IAC - Innovative and Alternative Fusion Concepts; OV – Overview.

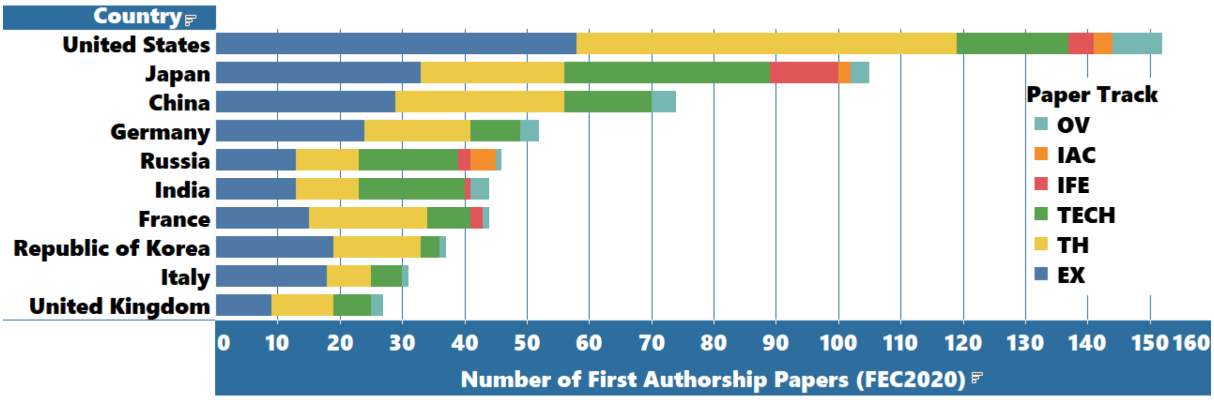

FIG.4. Top ten countries per number of first authorship papers submitted at the 28th IAEA Fusion Energy Conference (May 2021). Paper tracks are: EX - Magnetic Fusion Experiments; TH - Magnetic Fusion Theory and Modelling; TECH - Fusion Energy Technology; IFE - Inertial Fusion Energy; IAC - Innovative and Alternative Fusion Concepts; OV - Overview.

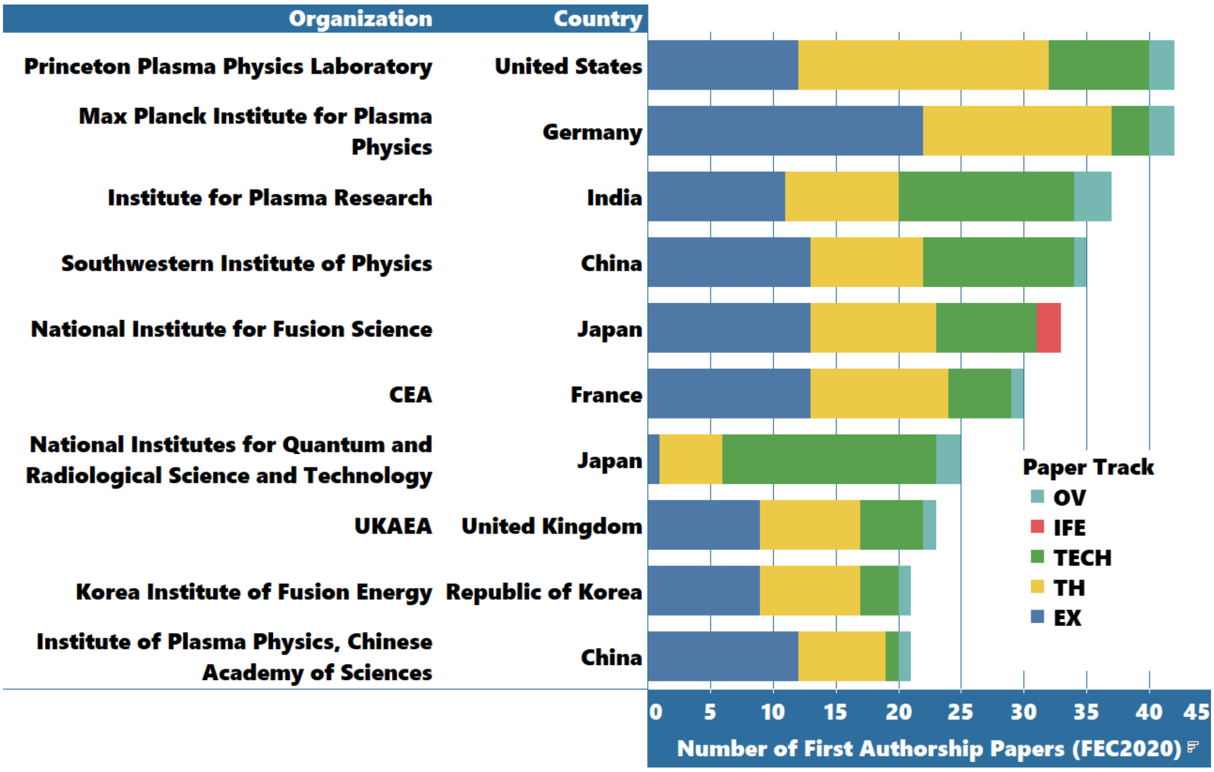

FIG.5. Top ten organizations per number of first authorship papers submitted at the 28th IAEA Fusion Energy Conference (May 2021). Paper tracks are: EX - Magnetic Fusion Experiments; TH - Magnetic Fusion Theory and Modelling; TECH - Fusion Energy Technology; IFE - Inertial Fusion Energy; IAC - Innovative and Alternative Fusion Concepts; OV - Overview.

A similar increasing pattern emerges when looking at private sector investments over the past two years. Over 30 private fusion companies can be found in Australia, Canada, China, France, Germany, Israel, Italy, Japan, United Kingdom, and USA. As of July 2022, private sector companies have declared that they have attracted around US$5 billion in total (Fig. 6).

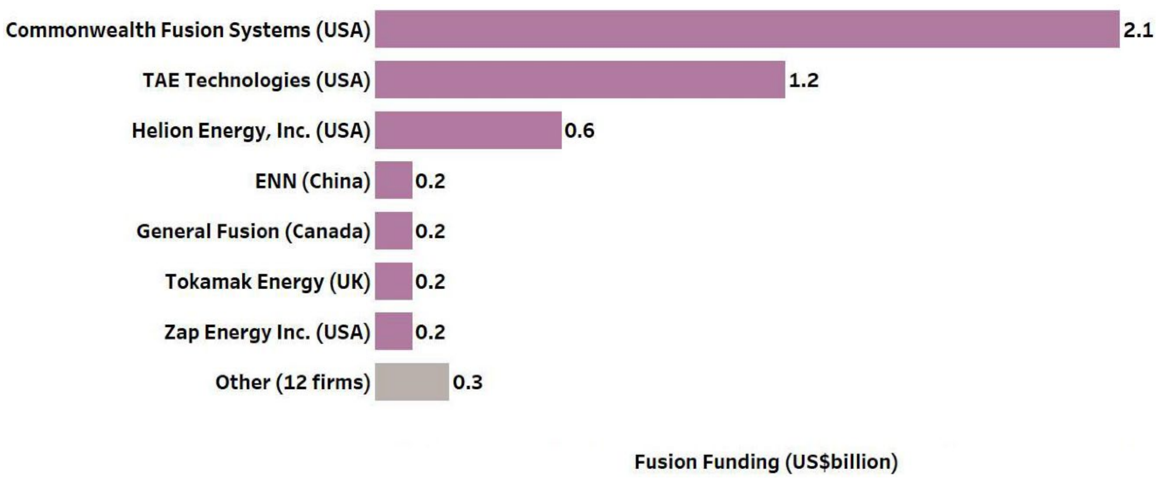

FIG.6. Private sector companies have disclosed around US$5 billion in fusion funding (more than $3 billion since June 2021). Readapted and updated from: The chase for fusion energy, Nature (2021); The global fusion industry in 2022, Fusion Industry Association (2022).

Reflecting the global interest rising around fusion energy R&D, this publication – based on the IAEA's Fusion Device Information System (FusDIS)[1] – provides information on all fusion devices public or private with experimental and demonstration designs, which are currently in operation, under construction or being planned, as well as technical data of these devices. An overview is given in Fig. 7.

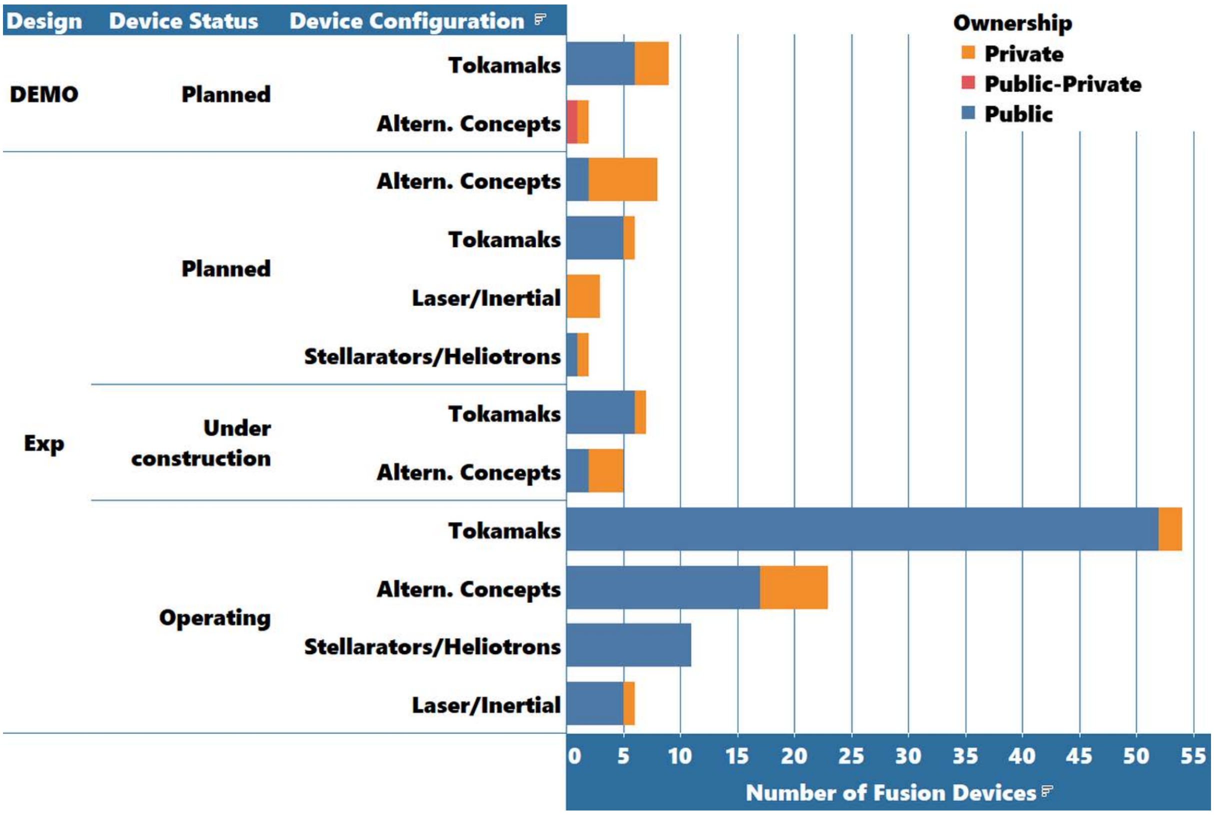

FIG.7. Over 130 experimental, public and private, fusion devices are operating, under construction or being planned, while a number of organizations are considering designs for demonstration fusion power plants.

1.2.OBJECTIVE

The objective of this publication is to provide a survey of public and private fusion devices worldwide with experimental and demonstration designs, which are currently in operation, under construction or being planned.

1.3. SCOPE

This publication focuses on the following device configuration categories:

- Tokamaks – both conventional and spherical type;
- Stellarators and heliotrons;
- Laser and inertial fusion; and

[1] Available at https://nucleus.iaea.org/sites/fusionportal/Pages/FusDIS.aspx

- Alternative concepts – this category includes the following types: dense plasma focus; field reversed configuration; inertial electrostatic fusion; levitated dipole; magnetic mirror machine; magnetized target fusion; pinch; reverse field pinch; simple magnetized torus; space propulsor; and spheromak.

For each device, the following information is provided:

- Country;
- Organization;
- Device name;
- Device type;
- Device status (operating; under construction; planned);
- Design (experimental; DEMO); and
- Ownership (public; private; public-private).

R&D in fusion science is also carried out at various experimental and testing facilities working in the areas of plasma physics, material science, nuclear engineering, manufacturing and robotics. These facilities have not been included in this publication.

1.4. STRUCTURE

This publication is divided into six parts:

— Section 1 (this section) gives a general background and describes the objective, scope and structure of this publication;
— Section 2 features public and private experimental tokamaks, which are in operation, under construction or being planned – see Table 2.
— Section 3 features public and private experimental stellarators and heliotrons, which are in operation or being planned – see Table 3.
— Section 4 features public and private experimental inertial and laser based-fusion devices, which are in operation or being planned – see Table 4.
— Section 5 features public and private experimental fusion devices based on alternative confinement concepts, which are in operation, under construction or being planned – see Table 5.
— Section 6 presents an overview of public and private fusion DEMO devices, which are being planned – see Table 6.

TABLE 2. LIST OF EXPERIMENTAL TOKAMAKS

Country	Organization	Name	Type	Status	Design	Ownership
Brazil	Federal University of Espírito Santo	NOVA-FURG	Conventional Tokamak	Operating	Experimental	Public
	National Institute for Space Research-INPE	ETE	Spherical Tokamak	Operating	Experimental	Public
	University of São Paulo	TCABR	Conventional Tokamak	Operating	Experimental	Public
Canada	University of Saskatchewan	STOR-M	Conventional Tokamak	Operating	Experimental	Public
China	Chinese Academy of Sciences	EAST	Conventional Tokamak	Operating	Experimental	Public
	ENN	EXL-50	Spherical Tokamak	Operating	Experimental	Private
	Southwestern Institute of Physics	HL-2A	Conventional Tokamak	Operating	Experimental	Public
		HL-2M	Conventional Tokamak	Operating	Experimental	Public
	Huazhong University of Science and Technology	J-TEXT	Conventional Tokamak	Operating	Experimental	Public
	Tsinghua University	SUNIST-1	Spherical Tokamak	Operating	Experimental	Public
Costa Rica	Instituto Tecnologico de Costa Rica	MEDUSA-CR	Spherical Tokamak	Operating	Experimental	Public
Czech Republic	Czech Technical University	GOLEM	Conventional Tokamak	Operating	Experimental	Public
	Institute of Plasma Physics of the Czech Academy of Sciences	COMPASS-U	Conventional Tokamak	Under construction	Experimental	Public
Denmark	Technical University of Denmark	NORTH	Spherical Tokamak	Operating	Experimental	Public
Egypt	Egyptian Atomic Energy Authority	EGYPTOR	Conventional Tokamak	Operating	Experimental	Public

TABLE 2. LIST OF EXPERIMENTAL TOKAMAKS CONTINUED

Country	Organization	Name	Type	Status	Design	Ownership
France	CEA	WEST	Conventional Tokamak	Operating	Experimental	Public
	ITER Organization	ITER	Conventional Tokamak	Under construction	Experimental	Public
Germany	Max Planck Institute for Plasma Physics	ASDEX UPGRADE	Conventional Tokamak	Operating	Experimental	Public
India	Institute for Plasma Research	ADITYA-U	Conventional Tokamak	Operating	Experimental	Public
		SST-1	Conventional Tokamak	Operating	Experimental	Public
		SSST	Spherical Tokamak	Under construction	Experimental	Public
Islamic Republic of Iran	Iran Atomic Energy Organization	ALVAND	Conventional Tokamak	Operating	Experimental	Public
		DAMAVAN D	Conventional Tokamak	Operating	Experimental	Public
	Islamic Azad University	IR-T1	Conventional Tokamak	Operating	Experimental	Public
Italy	ENEA	DTT	Conventional Tokamak	Planned	Experimental	Public
		FTU	Conventional Tokamak	Operating	Experimental	Public
Japan	Kyoto University	LATE	Spherical Tokamak	Operating	Experimental	Public
	Kyushu University	PLATO	Conventional Tokamak	Planned	Experimental	Public
		QUEST	Spherical Tokamak	Operating	Experimental	Public
	Nagoya University	HYBTOK-II	Conventional Tokamak	Operating	Experimental	Public
		TOKASTA R-2	Conventional Tokamak	Operating	Experimental	Public
	National Institutes for Quantum and Radiological Science and Technology	JT-60SA	Conventional Tokamak	Operating	Experimental	Public
	The University of Tokyo	TST-2	Spherical Tokamak	Operating	Experimental	Public
		UTST	Spherical Tokamak	Operating	Experimental	Public

TABLE 2. LIST OF EXPERIMENTAL TOKAMAKS CONTINUED

Country	Organization	Name	Type	Status	Design	Ownership
Japan	Tokyo Institute of Technology	PHiX	Conventional Tokamak	Operating	Experimental	Public
	University of Hyogo	HIST	Spherical Tokamak	Operating	Experimental	Public
Kazakhstan	Institute of Atomic Energy NNC RK	KTM	Spherical Tokamak	Operating	Experimental	Public
Libya	Tajoura Nuclear Research Centre	LIBTOR	Conventional Tokamak	Operating	Experimental	Public
Pakistan	Pakistan Atomic Energy Commission	GLAST-III	Spherical Tokamak	Operating	Experimental	Public
		MT-1	Spherical Tokamak	Operating	Experimental	Public
		MT-2	Spherical Tokamak	Under construction	Experimental	Public
		PST	Spherical Tokamak	Planned	Experimental	Public
Portugal	Instituto Superior Técnico	ISTTOK	Conventional Tokamak	Operating	Experimental	Public
Republic of Korea	Korea Institute of Fusion Energy	KSTAR	Conventional Tokamak	Operating	Experimental	Public
	Seoul National University	VEST	Spherical Tokamak	Operating	Experimental	Public
Russian Federation	Ioffe Institute	FT-2	Conventional Tokamak	Operating	Experimental	Public
		Globus-M2	Spherical Tokamak	Operating	Experimental	Public
		TUMAN-3M	Conventional Tokamak	Operating	Experimental	Public
	National Research Centre Kurchatov Institute	T-15MD	Conventional Tokamak	Under construction	Experimental	Public
	Saint Petersburg State University	GUTTA	Spherical Tokamak	Operating	Experimental	Public
	Troitsk Institute for Innovation and Fusion Research	T-11M	Conventional Tokamak	Operating	Experimental	Public

TABLE 2. LIST OF EXPERIMENTAL TOKAMAKS CONTINUED

Country	Organization	Name	Type	Status	Design	Ownership
Spain	University of Seville	SMART	Spherical Tokamak	Planned	Experimental	Public
Switzerland	Swiss Plasma Center	TCV	Conventional Tokamak	Operating	Experimental	Public
Thailand	Thailand Institute of Nuclear Technology	TT-1	Conventional Tokamak	Under construction	Experimental	Public
United Kingdom	EUROfusion	JET	Conventional Tokamak	Operating	Experimental	Public
	Tokamak Energy	ST40	Spherical Tokamak	Operating	Experimental	Private
	UKAEA	MAST-U	Spherical Tokamak	Operating	Experimental	Public
United States of America	Columbia University	HBT-EP	Conventional Tokamak	Operating	Experimental	Public
	Commonwealth Fusion Systems	SPARC	Conventional Tokamak	Under construction	Experimental	Private
	General Atomics	DIII-D	Conventional Tokamak	Operating	Experimental	Public
	Princeton Plasma Physics Laboratory	LTX-β	Spherical Tokamak	Operating	Experimental	Public
		NSTX-U	Spherical Tokamak	Operating	Experimental	Public
	University of Wisconsin-Madison	PEGAS US-III	Spherical Tokamak	Operating	Experimental	Public

TABLE 3. LIST OF EXPERIMENTAL STELLARATORS AND HELIOTRONS

Country	Organization	Name	Type	Status	Design	Ownership
China	Southwest Jiaotong University	CFQS	Stellarator	Planned	Experimental	Public
Costa Rica	Instituto Tecnologico De Costa Rica	CSR-1	Stellarator	Operating	Experimental	Public
France	Renaissance Fusion	RENAISSANCE FUSION	Stellarator	Planned	Experimental	Private
Germany	Max Plank Institute for Plasma Physics	WENDELSTEIN 7-X	Stellarator	Operating	Experimental	Public
	University of Stuttgart	TJ-K	Stellarator	Operating	Experimental	Public
Japan	National Institute for Fusion Science	LHD	Heliotron	Operating	Experimental	Public
	Kyoto University	HELIOTRON J	Heliotron	Operating	Experimental	Public
Spain	CIEMAT	TJ-II	Stellarator	Operating	Experimental	Public
Ukraine	Institute of Plasma Physics National Science Center	URAGAN-2M	Stellarator	Operating	Experimental	Public
		URAGAN-3M	Stellarator	Operating	Experimental	Public
United States of America	Auburn University	CTH	Torsatron	Operating	Experimental	Public
	University of Illinois	HIDRA	Stellarator/ Tokamak	Operating	Experimental	Public
	University of Wisconsin-Madison	HSX	Stellarator	Operating	Experimental	Public

TABLE 4. LIST OF EXPERIMENTAL INERTIAL AND LASER FUSION DEVICES

Country	Organization	Name	Type	Status	Design	Ownership
Australia	HB11 Energy	HB11	Laser Fusion	Planned	Experimental	Private
France	CEA	LMJ	Laser Fusion	Operating	Experimental	Public
Germany	Marvel Fusion	MARVEL FUSION	Laser Fusion	Planned	Experimental	Private
Japan	Osaka University	GEKKO XII	Laser Fusion	Operating	Experimental	Public
		LFEX	Laser Fusion	Operating	Experimental	Public
United Kingdom	First Light Fusion Ltd	FIRST LIGHT FUSION	Inertial Fusion	Operating	Experimental	Private
United States of America	Innoven Energy	INNOVEN ENERGY LLC	Laser Fusion	Planned	Experimental	Private
	Lawrence Livermore National Laboratory	NIF	Laser Fusion	Operating	Experimental	Public
	University of Rochester Laboratory for Laser Energetics	OMEGA	Laser Fusion	Operating	Experimental	Public

TABLE 5. LIST OF EXPERIMENTAL ALTERNATIVE FUSION DEVICES

Country	Organization	Name	Type	Status	Design	Ownership
Canada	General Fusion Inc	GENERAL FUSION	Magnetized Target Fusion	Under construction	Experimental	Private
China	University Of Science and Technology of China	KTX	Reversed Field Pinch	Operating	Experimental	Public
France	École Polytechnique	TORIX	Simple Magnetized Torus	Operating	Experimental	Public
Italy	Consorzio RFX	RFX	Reversed Field Pinch	Operating	Experimental	Public
Japan	Kyoto Institute of Technology	RELAX	Reversed Field Pinch	Operating	Experimental	Public
	Kyushu University	UH-CTI	Spheromak	Operating	Experimental	Public
	Nihon University	FAT-CM	Field Reversed Config.	Operating	Experimental	Private
	University of Tokyo	RT-1	Levitated Dipole	Operating	Experimental	Public
	University of Hyogo	UH-MCPG1	Spheromak	Operating	Experimental	Public
	University of Tsukuba	GAMMA 10/PDX	Magnetic Mirror Machine	Operating	Experimental	Public
		PILOT GAMMA PDX-SC	Magnetic Mirror Machine	Under construction	Experimental	Public
Russian Federation	Budker Institute of Nuclear Physics	CAT	Magnetic Mirror Machine	Under construction	Experimental	Public
		GDMT	Magnetic Mirror Machine	Planned	Experimental	Public
		GDMT CORE	Magnetic Mirror Machine	Planned	Experimental	Public
		GDT	Magnetic Mirror Machine	Operating	Experimental	Public
		GOL-NB	Magnetic Mirror Machine	Operating	Experimental	Public
		SMOLA	Magnetic Mirror Machine	Operating	Experimental	Public

TABLE 5. LIST OF EXPERIMENTAL ALTERNATIVE FUSION DEVICES CONTINUED

Country	Organization	Name	Type	Status	Design	Ownership
Sweden	KTH Royal Institute of Technology	EXTRAP T2R	Reversed Field Pinch	Operating	Experimental	Public
United States of America	Compact Fusion Systems	FUSION POWER CORE	Magnetized Target Fusion	Planned	Experimental	Private
	CTFusion	IDCD	Spheromak	Operating	Experimental	Private
	Helicity Space	HELICITY DRIVE	Space Propulsor	Planned	Experimental	Private
	Helion Energy	POLARIS	Field Reversed Config.	Planned	Experimental	Private
	Helion Energy Horne Technologies LLC	TRENTA	Field Reversed Config.	Operating	Experimental	Private
		HORNE HYBRID REACTOR	Inertial Electr. Fusion	Under construction	Experimental	Private
	Hyperjet Fusion Corporation	PJMIF	Magnetized Target Fusion	Planned	Experimental	Private
	Lawrenceville Plasma Physics	FOCUS FUSION	Dense Plasma Focus	Operating	Experimental	Private
	Lockheed Martin	CFR	Magnetic Mirror Machine	Operating	Experimental	Public
	Magneto-Inertial Fusion Technologies	MIFTI	Pinch	Planned	Experimental	Private
	Princeton Fusion Systems	PFRC	Field Reversed Config.	Planned	Experimental	Private
	Sandia National Laboratories	Z MACHINE	Pinch	Operating	Experimental	Public
	TAE Technologies	NORMAN (C2-W)	Field Reversed Config.	Operating	Experimental	Private
		COPERNICUS	Field Reversed Config.	Under construction	Experimental	Private
	University of Nevada	ZEBRA	Pinch	Operating	Experimental	Public
	University of Wisconsin-Madison	MST	Reversed Field Pinch	Operating	Experimental	Private
	Zap Energy Inc.	FUZE-Q	Pinch	Operating	Experimental	Private

TABLE 6. LIST OF DEMO DEVICES

Country	Organization	Name	Type	Status	Design	Ownership
China	Chinese Consortium	CFETR	Conventional Tokamak	Planned	DEMO	Public
European Union	EUROfusion	EU-DEMO	Conventional Tokamak	Planned	DEMO	Public
Japan	Japanese Consortium	J-DEMO	Conventional Tokamak	Planned	DEMO	Public
Republic of Korea	Korea Institute of Fusion Energy	K-DEMO	Conventional Tokamak	Planned	DEMO	Public
Russian Federation	Russian Consortium	DEMO-RF	Conventional Tokamak	Planned	DEMO	Public
United Kingdom	General Fusion Inc	FDP	Magnetized Target Fusion	Planned	DEMO	Public-Private
	Tokamak Energy	ST-E1	Spherical Tokamak	Planned	DEMO	Private
	UKAEA	STEP	Spherical Tokamak	Planned	DEMO	Public
United States of America	Commonwealth Fusion Systems	ARC	Conventional Tokamak	Planned	DEMO	Private
	General Atomics	GA-FPP	Conventional Tokamak	Planned	DEMO	Private
	TAE Technologies	DA VINCI	Field Reversed Config.	Planned	DEMO	Private

2. EXPERIMENTAL TOKAMAKS

2.1. NOVA-FURG (FEDERAL UNIVERSITY OF ESPÍRITO SANTO, BRAZIL)

2.1.1. Introduction

NOVA-FURG was formerly known as the NOVA-II tokamak and belonged to Kyoto University, Japan. It has been operating in Brazil since 1996. It is a small device in comparison with the other tokamaks around the world, but its size gives certain advantages, as for example it is cheaper to operate, and thus it makes possible to repeat an experiment many times for testing a theory.

2.1.2. Purpose

The purpose of NOVA-FURG tokamak is study plasma-wall interaction and optical diagnostic development.

2.1.3. Main features

NOVA-FURG is a small iron-cored machine operating with conducting shell stabilization [1]. Technical information is listed in Table 7 below.

TABLE 7. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.3 m
Minor radius, a	0.06 m
Plasma current, I_p	0.01 MA
Toroidal field, B_o	1 T
Pulse length	0.015 s
Magnetic field configuration	Circular limiter
Ownership	Public

2.2. ETE (NATIONAL INSTITUTE FOR SPACE RESEARCH, BRAZIL)

2.2.1. Introduction

The ETE spherical tokamak became operational at the end of 2000. It was fully constructed at Associated Plasma Laboratory of Brazil's National Institute for Space Research, Brazil.

2.2.2. Purpose

The main objectives of the project are to study the basic physics of low aspect ratio geometry plasmas with emphasis on the plasma edge as well as on plasma wall interaction. Development of new diagnostics and training in tokamak operation are also important objectives of ETE.

2.2.3. Main features

The ETE spherical tokamak is a small-to-medium size low aspect ratio machine. ETE was designed with a minimum set of coils for the poloidal and toroidal fields, minimizing stray magnetic fields and preserving good plasma accessibility. All coils are water cooled and were manufactured with standard pure cooper. The toroidal field system consists of 12 D-shaped single coils connected in series by two feed rings that compensate the stray magnetic field [2]. Technical information is listed in Table 8 below.

TABLE 8. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.3 m
Minor radius, a	0.2 m
Plasma current, I_p	0.045 MA
Toroidal field, B_o	0.4 T
Pulse length	0.006–0.012 s
Magnetic field configuration	Graphite limiter
Ownership	Public

2.3. TCABR (UNIVERSITY OF SÃO PAULO, BRAZIL)

2.3.1. Introduction

Originally TCABR was designed by Swiss Federal Institute of Technology Lausanne, Switzerland where it was operated from 1980 to 1992. Later it was rebuilt at the Plasma Physics Laboratory of the University of São Paulo, Brazil with new systems of discharge control and of Alfvén wave excitation and named TCABR. The first plasma was produced in 1999.

2.3.2. Purpose

The main purpose of TCABR is investigating plasma heating by Alfvén waves.

2.3.3. Main features

TCABR is a medium-sized, ohmically heated tokamak with a circular plasma cross-section that works with the hydrogen plasma. TCABR has the main standard diagnostics such as microwave interferometry, optical spectroscopy, soft and hard X ray detection, electron–cyclotron emission detector, bolometer, H_α-emission detector, Mirnov coils and a variety of magnetic and electrostatic probes [3]. Technical information is listed in Table 9 below.

TABLE 9. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.61 m
Minor radius, a	0.18 m
Plasma current, I_p	0.1 MA
Toroidal field, B_o	1.07 T
Pulse length	0.1 s
Magnetic field configuration	Poloidal graphite limiter Magnetic ergodic limiter
Ownership	Public

2.4. STOR-M (UNIVERSITY OF SASKATCHEWAN, CANADA)

2.4.1. Introduction

STOR-M is a research tokamak that was built at University of Saskatchewan, Canada. After Tokamak de Varennes was closed in 1997, STOR-M became the only Canadian magnetic fusion device.

2.4.2. Purpose

STOR-M aims to demonstrate the feasibility of quasi steady state tokamak reactors, study the compact torus injection (a method of fuelling the core of tokamak fusion reactors), as well as develop novel far infrared lasers-based diagnostics for ITER and research on plasma assisted material synthesis and processing.

2.4.3. Main features

The STOR-M tokamak is a small iron core research tokamak. For heating it uses method based on exploiting ohmic H-modes, turbulent heating. STOR-M is equipped with feedback control system for horizontal and vertical plasma positions, a driver for fast rising ohmic current, a circuit system for alternating current operation, compact torus injector, and diagnostics [4]. Technical information is listed in Table 10 below.

TABLE 10. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.46 m
Minor radius, a	0.125 m
Plasma current, I_p	30–60 MA
Toroidal field, B_o	0.5–1 T
Pulse length	0.002–0.005 s
Magnetic field configuration	Stainless steel limiter (circular)
Ownership	Public

2.5. EAST (CHINESE ACADEMY OF SCIENCES, CHINA)

2.5.1. Introduction

EAST is located at Institute of Plasma Physics Chinese Academy of Sciences (ASIPP), China. It followed the first Chinese superconducting tokamak HT-7, built by ASIPP in partnership with the Russian Federation in the early 1990s. The first plasma at EAST was obtained in 2006.

2.5.2. Purpose

EAST aims to contribute studies of plasma physics, provide a scientific basis for the design and construction of experimental reactors, including ITER, as EAST has shape and equilibrium similar to ITER.

2.5.3. Main features

EAST is a tokamak operating with superconducting magnets. The main distinguishing features of EAST are its non-circular cross-section, fully superconducting magnets and fully actively water-cooled plasma facing components, which will be beneficial to explore the advanced steady-state plasma operation modes [5]. Technical information is listed in Table 11 below.

TABLE 11. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	1.7 m
Minor radius, a	0.4 m
Plasma current, Ip	1 MA
Toroidal field, B_o	3.5 T
Pulse length	1-1000 s
Magnetic field configuration	Double-null divertor Pump limiter Single null divertor
Ownership	Public

2.6. EXL-50 (ENN, CHINA)

2.6.1. Introduction

EXL-50 is the Chinese first medium-sized experimental spherical tokamak. It was built by the ENN Group, a Chinese energy company. The first plasma discharge without solenoid was obtained in 2019. The EXL-50 is also part of the ENN Compact Fusion Project, funded by the private company ENN Group.

2.6.2. Purpose

The main purpose of EXL-50 is to simplify engineering requirements of a fusion device. One of the key experimental objectives of the EXL-50 is to test the efficiency of the electron cyclotron resonance heating and the current drive in the absence of the central solenoid magnet.

2.6.3. Main features

The EXL-50 device is a medium-sized experimental spherical tokamak with a cylindrical vacuum vessel and with fully non-inductive current drive. EXL-50 has six poloidal field coils which are located outside the vacuum vessel and the toroidal field coil conductors to obtain higher toroidal field discharges [6]. Technical information is listed in Table 12 below.

TABLE 12. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.58 m
Minor radius, a	0.41 m
Plasma current, I_p	0.5 MA
Toroidal field, B_o	0.45 T
Pulse length	5 s
Magnetic field configuration	Copper limiters coated with 0.3 mm tungsten
Ownership	Public

2.7. HL-2A (SOUTHWESTERN INSTITUTE OF PHYSICS, CHINA)

2.7.1. Introduction

Built in 2002, HL-2A was the first tokamak with a divertor in China. The tokamak is based on main components (magnet coils and plasma vessel) belonging to the former ASDEX tokamak. HL-2A is located at Southwestern Institute of Physics, China.

2.7.2. Purpose

The key mission of the HL-2A tokamak programme is to address critical physics and technology issues for ITER and next-step fusion devices. The research focuses on radio frequency wave heating, current drive, plasma confinement, turbulent transport, MHD instabilities, energetic particle physics, H-mode and ELM control.

2.7.3. Main features

HL-2A is a medium-sized tokamak with a lower single null divertor configuration. Recently a 2 MW lower hybrid current drive system with passive active multi-junction antenna has been developed for HL-2A [7]. Technical information is listed in Table 13 below.

TABLE 13. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	1.65 m
Minor radius, a	0.4 m
Plasma current, I_p	0.48 MA
Toroidal field, B_o	2.8 T
Pulse length	5 s
Magnetic field configuration	Limiter lower single null divertor
Ownership	Public

2.8. HL-2M (SOUTHWESTERN INSTITUTE OF PHYSICS, CHINA)

2.8.1. Introduction

HL-2M is located at Southwestern Institute of Physics, China. It is a new machine with the highest plasma performance in China. The first plasma was achieved in 2020.

2.8.2. Purpose

The key areas of the HL-2M research are high performance and high beta scenarios compatible with flexible advanced divertor configurations, tests and validation of high heat flux plasma-facing components, as well as investigation of advanced plasma physics with high performance.

2.8.3. Main features

The HL-2M device is a tokamak with copper conductor magnets. It is equipped with a more effective and flexible divertor and a new set of toroidal and poloidal field coils [8]. HL-2M features advanced divertor configurations (snowflake, tripod). Technical information is listed in Table 14 below.

TABLE 14. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	1.78 m
Minor radius, a	0.65 m
Plasma current, I_p	3 MA
Toroidal field, B_o	3 T
Pulse length	10 s
Magnetic field configuration	Divertor (snowflake, tripod, single null, double null)
Ownership	Public

2.9. J-TEXT (HUAZHONG UNIVERSITY OF SCIENCE AND TECHNOLOGY, CHINA)

2.9.1. Introduction

The J-TEXT, formerly TEXT/TEXT-U at University of Texas at Austin, USA, is located at Huazhong University of Science and Technology, China. First plasma was produced in 2007.

2.9.2. Purpose

The main purpose of J-TEXT is proving fundamental physics and control mechanisms of fusion plasma confinement and stability in support of ITER's successful operation and the design of CFETR (see p. 143).

2.9.3. Main features

The J-TEXT is a conventional tokamak with an iron core. The original limiter configuration (with three moveable poloidal rail limiter targets) was upgraded in 2016. J-TEXT configuration makes it possible to study 3-D effects and disruption mitigation in a tokamak thanks to the upgraded resonant magnetic perturbation system and the shattered pellet injection system [9]. Technical information is listed in Table 15 below.

TABLE 15. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	1.05 m
Minor radius, a	0.25–0.29 m
Plasma current, I_p	0.2 MA
Toroidal field, B_o	2 T
Pulse length	0.8 s
Magnetic field configuration	Titanium-carbide-coated graphite limiter Divertor
Ownership	Public

2.10. SUNIST-1 (TSINGHUA UNIVERSITY, CHINA)

2.10.1. Introduction

SUNIST-1 is the first spherical tokamak built in China in a partnership between National Nature Science Foundation, Tsinghua University and Institute of Physics, Chinese Academy of Sciences. First plasma was produced in 2002 and the machine was subsequently upgraded in 2008.

2.10.2. Purpose

The research objectives of SUNIST-1 are to increase the understanding of toroidal plasma physics with a low aspect ratio and to produce a maintainable target plasma with non-inductive start-up.

2.10.3. Main features

SUNIST-1 is a small-scale spherical tokamak. The central stack is the most important component of its design, consisting of a 1 mm thick 304 stainless steel column, surrounding the ohmic solenoid and toroidal field inner limbs. In the recent years, SUNIST-1 has been upgraded with newly developed plasma diagnostics and plasma actuators. In addition, a high current density plasma gun was installed [10]. Technical information is listed in Table 16 below.

TABLE 16. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.3 m
Minor radius, a	0.23 m
Plasma current, I_p	0.05–1 MA
Toroidal field, B_o	0.15 T
Pulse length	0.001 s
Magnetic field configuration	Poloidal limiters
Ownership	Public

2.11. MEDUSA-CR (INSTITUTO TECNOLOGICO DE COSTA RICA, COSTA RICA)

2.11.1. Introduction

MEDUSA-CR is a low aspect ratio spherical tokamak built by University of Wisconsin-Madison, USA and donated to Instituto Tecnológico de Costa Rica, Costa Rica.

2.11.2. Purpose

MEDUSA-CR is operated for educational and training activities in the field of plasma physics and tokamak physics.

2.11.3. Main features

The vacuum vessel of MEDUSA-CR is made of stainless steel, with the external Alfvén wave antennas and an ergodic limiter both placed externally to the vessel [11]. Technical information is listed in Table 17 below.

TABLE 17. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.14 m
Minor radius, a	0.1 m
Plasma current, I_p	0.04 MA
Toroidal field, B_o	0.5 T
Pulse length	0.003 s
Magnetic field configuration	Rail limiter
Ownership	Public

2.12. GOLEM (CZECH TECHNICAL UNIVERSITY, CZECH REPUBLIC)

2.12.1. Introduction

Originally built in the 1960s (as TM-1 tokamak) at Kurchatov Institute, Russian Federation – the GOLEM tokamak is located at Czech Technical University, Czech Republic. It is the oldest tokamak in operation.

2.12.2. Purpose

The GOLEM tokamak is used for educational and training activities in the field of tokamak physics, technology, diagnostics and operation.

2.12.3. Main features

GOLEM is a small tokamak with a circular cross-section. Remote participation and control using internet access are unique features of GOLEM, allowing for basic remote control in both online and offline mode [12]. Technical information is listed in Table 18 below.

TABLE 18. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.4 m
Minor radius, a	0.085 m
Plasma current, I_p	0.008 MA
Toroidal field, B_o	0.8 T
Pulse length	0.025 s
Magnetic field configuration	Limiter
Ownership	Public

2.13. COMPASS-U (INSTITUTE OF PLASMA PHYSICS, CZECH REPUBLIC)

2.13.1. Introduction

COMPASS-U is a high magnetic field tokamak being constructed at Institute of Plasma Physics, Czech Republic in cooperation between various European and international partners. Operation is expected to start in 2023.

2.13.2. Purpose

COMPASS-U research will support ITER's operation and help address key challenges in EU-DEMO design activities (see DEMOs section). The device will allow for operation with hot first wall and full recycling regime, and it is designed to study and test liquid metal technology in the divertor.

2.13.3. Main features

The main features are a closed divertor with high plasma and neutral density, high opacity, extreme power fluxes, high magnetic field, allowing access to advanced confinement modes [13]. Technical information is listed in Table 19 below.

TABLE 19. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Under construction
Major radius, R_o	0.894
Minor radius, a	0.27
Plasma current, I_p	2 MA
Toroidal field, B_o	5 T
Pulse length	1–3 s (up to 11 s in lower single null plasmas)
Magnetic field configuration	Lower single null, negative triangularity with limited plasma information (Phase 1-2) Double null (Phase 2-3) Snowflake, negative triangularity (Phase 3-4)
Ownership	Public

2.14. NORTH (TECHNICAL UNIVERSITY OF DENMARK, DENMARK)

2.14.1. Introduction

NORTH is a tokamak operating at Technical University of Denmark, Denmark, since 2019. This tokamak is a joint project between the Technical University of Denmark and private sector company Tokamak Energy, UK who constructed the device. The device is the first tokamak experiment at the Technical University of Denmark [14].

2.14.2. Purpose

NORTH is used for plasma physics research.

2.14.3. Main features

The NORTH tokamak is a small-scale spherical tokamak with two magnetrons (3 kW each) operating at 2.45 GHz being used for plasma heating. Technical information is listed in Table 20 below.

TABLE 20. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.25 m
Toroidal field, B_o	0.3 T
Ownership	Public

2.15. EGYPTOR (EGYPTIAN ATOMIC ENERGY AUTHORITY, EGYPT)

2.15.1. Introduction

EGYPTOR is a small tokamak originally built by Heinrich-Heine Universität Düsseldorf, Germany and later re-installed in Egypt.

2.15.2. Purpose

EGYPTOR research programme aims at testing new diagnostic and cleaning discharge techniques for developing better wall conditioning systems for large tokamaks.

2.15.3. Main features

EGYPTOR is a small-scale tokamak with stainless steel vacuum vessel, control and cleaning discharge systems. Diagnostics can be accessed thanks to six large lateral ports and twelve smaller windows located at top and bottom of the vessel [15]. Technical information is listed in Table 21 below.

TABLE 21. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.3 m
Minor radius, a	0.1 m
Plasma current, I_p	0.05 MA
Toroidal field, B_o	1.2 T
Pulse length	0.045 s
Magnetic field configuration	Rail limiter
Ownership	Public

2.16. ITER (ITER ORGANIZATION, FRANCE)

2.16.1. Introduction

ITER is an experimental fusion reactor under construction in France. The project is a joint international undertaking between China, the European Union with UK and Switzerland through other agreements, India, Japan, South Korea, Russian Federation, and the USA. As of May 2022, its construction is about 75% complete, with the successful testing of many key components such as vacuum vessel sectors, toroidal field coils, poloidal field coils, cryostat sections, and thermal shields. ITER will start conducting its first experiments in the second half of 2020 and full-power experiments are planned to commence in 2036.

2.16.2. Purpose

The purpose of ITER is to demonstrate the scientific and technological feasibility of fusion energy production. The principal scientific mission objectives of the ITER project are demonstrating a scientific energy gain[2] $Q_{sci}\geq 10$ for deuterium-tritium plasma burn durations of 300–500 s (inductive ELMy H-mode); and the development of long-pulse, non-inductive scenarios aiming at maintaining Q_{sci}~5 for periods of up to 3000 s [16].

2.16.3. Main features

Weighing 23 000 tonnes and standing at nearly 30 metres tall, ITER will sit at the heart of a 180-hectare site, together with auxiliary housing and equipment. The ITER tokamak will have a plasma volume of 830 m^3. Its magnet system will be made of 18 toroidal field magnets, 6 poloidal field coils, a thirteen meters tall central solenoid, 18 superconducting correction coils, 31 superconducting magnet feeders and 29 non-superconducting in-vessel coils. The divertor will be made up of 54 stainless-steel pieces known as cassettes, each weighing 10 tonnes. The components facing the plasma will be armoured with tungsten, a material that has both low tritium absorption and the highest melting temperature of any natural element. The ITER external heating systems will rely on 33 MW neutral beam injection, 20 MW ion cyclotron heating and 20 MW electron cyclotron heating. Technical information is listed in Table 22 below.

TABLE 22. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Under construction
Major radius, R_o	6.2 m
Minor radius, a	2 m
Plasma current, I_p	15 MA
Toroidal field, B_o	5.3 T
Pulse length	up to 3000 s
Magnetic field configuration	Divertor
Ownership	Public

[2] The scientific energy gain corresponds to the fusion energy released (heat produced) divided by the energy delivered to the fuel (heating given). This is different from the engineering energy gain (Q_{eng}), which corresponds to the ratio of grid power to recirculating electrical power. ITER is designed to achieve scientific energy gain. DEMO-type devices are designed to achieve net engineering gain (Q_{eng}>1).

2.17. WEST (CEA, FRANCE)

2.17.1. Introduction

Previously known as Tore Supra and renamed WEST following an upgrade in the configuration (from limiter to divertor), this tokamak has been operating since 1988. WEST first plasma was produced in 2016. WEST was the first tokamak with superconducting magnets and actively cooled plasma facing components.

2.17.2. Purpose

Key missions of WEST are the qualification of the high heat flux plasma facing components in integrating both technological and physics aspects in relevant heat and particle exhaust conditions (particularly for the tungsten monoblocks foreseen in ITER divertor), as well as the demonstration of integrated steady state operation, with a focus on power exhaust issues.

2.17.3. Main features

WEST is a superconducting tokamak equipped with two up-down symmetric divertors. Different divertor configurations can be studied, i.e., lower single null, upper single null and double null. The plasma facing components on WEST are made of tungsten and are actively cooled. Thanks to the radiofrequency heating and current drive systems (9 MW of ion cyclotron resonance heating and 6 MW of lower hybrid current drive), WEST can operate with long pulses up to 1000 s with high particle fluence [17]. Technical information is listed in Table 23 below.

TABLE 23. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	2.5 m
Minor radius, a	0.5 m
Plasma current, I_p	1 MA
Toroidal field, B_o	3.7 T
Pulse length	Up to 1000 s
Magnetic field configuration	Divertor lower single null, upper single null and double null
Ownership	Public

2.18. ASDEX UPGRADE (MAX PLANCK INSTITUTE FOR PLASMA PHYSICS, GERMANY)

2.18.1. Introduction

ASDEX Upgrade (AUG) is a newly built tokamak which is based on the experience of the ASDEX tokamak. AUG is operating since 1991.

2.18.2. Purpose

AUG programme aims to establish the scientific basis for the optimisation of the tokamak approach to fusion energy and prepare for ITER and DEMO.

2.18.3. Main features

AUG is a medium sized tokamak with stainless steel vacuum vessel inner wall clad with tiles made of or coated with tungsten metal. Sixteen large copper magnet coils wrapped around the donut-shaped plasma vessel form the confining magnetic field. The device is equipped with three different plasma heating methods: neutral particle injection (20 MW), high-frequency heating (6 MW), and microwave heating (8 MW) [18]. Technical information is listed in Table 24 below.

TABLE 24. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	1.65 m
Minor radius, a	0.5 m
Plasma current, I_p	1.4 MA
Toroidal field, B_o	3.2 T
Pulse length	10 s
Magnetic field configuration	D-shaped divertor
Ownership	Public

2.19. ADITYA-U (INSTITUTE FOR PLASMA RESEARCH, INDIA)

2.19.1. Introduction

ADITYA-U is the upgraded version of ADITYA tokamak. First plasma in ADITYA-U was obtained in 2016.

2.19.2. Purpose

The main purpose of ADITYA-U is to provide the physical and technological scientific base for future fusion installations with special emphasis on performing experiments on disruption and runaway electron mitigation.

2.19.3. Main features

ADITYA-U is a medium-sized tokamak. ADITYA-U is equipped with an open divertor configuration, with divertor plates without any baffle and with three set of divertor coils. Two sets of coils are placed at high field side and a set of coils is placed at the low field side to obtain the shaped plasmas in lower single null, upper single null and double null divertor configurations [19]. ADITYA-U is equipped with inductively driven macroparticle injector system. Technical information is listed in Table 25 below.

TABLE 25. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.75 m
Minor radius, a	0.2–0.25 m
Plasma current, I_p	0.1–0.25 MA
Toroidal field, B_o	1.5 T
Pulse length	0.25–0.4 s
Magnetic field configuration	Circular and Single/Double null divertor
Ownership	Public

2.20. SST-1 (INSTITUTE FOR PLASMA RESEARCH, INDIA)

2.20.1. Introduction

Operating since 2013, SST-1 is located at Institute for Plasma Research, India.

2.20.2. Purpose

The focus of SST-1 research plan is to study plasma current drive through lower hybrid, including the active feedback control and plasma-wall interactions in steady-state plasmas.

2.20.3. Main features

SST-1 is a medium-sized superconducting tokamak having superconducting toroidal field magnets operating in two-phase helium in cryo-stable conditions. Recent modifications in the external ohmic coils system have allowed repeatable and consistent ohmic plasmas with electron cyclotron assisted pre-ionization [20]. Technical information is listed in Table 26 below.

TABLE 26. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	1.1 m
Minor radius, a	0.2 m
Plasma current, I_p	0.1 MA
Toroidal field, B_o	1.5 T (max 3 T)
Pulse length	0.65 s
Magnetic field configuration	Circular limiter
Ownership	Public

2.21. SSST (INSTITUTE FOR PLASMA RESEARCH, INDIA)

2.21.1. Introduction

Currently under construction, SSST will be the first spherical tokamak of the Institute for Plasma Research, India.

2.21.2. Purpose

SSST research plan will focus on producing low aspect ratio plasmas and perform various basic plasma experiments to study low aspect ratio plasmas, ohmic start-up with electron cyclotron resonance heating and non-inductive start-up, among others. The device will also be used for training in the field of low aspect ratio plasmas and associated technologies.

2.21.3. Main features

SSST will be a low cost spherical tokamak All coils will be made of copper and naturally cooled. Toroidal field and poloidal field coils will be demountable and driven by capacitor bank based power supplies. The general configuration of all power supplies will comprise of fast and slow capacitor banks and electronic switches that discharge the energy stored in the banks on the magnetic coil. The vacuum vessel will be made of SS304 and will be a continuous structure without any toroidal breaks. Two magnetron (6 kW) based radiofrequency sources will be used for pre-ionization and for non-inductive current drive during advanced operation of the machine. The device will be equipped with several type of diagnostics such as Rogowski coils, Langmuir probes, bolometers, microwave and infrared diagnostics and spectroscopy, among others. Technical information is listed in Table 27 below.

TABLE 27. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Under construction
Major radius, R_o	0.28 m
Minor radius, a	0.16 m
Plasma current, I_p	0.028 MA
Toroidal field, B_o	0.15 T
Ownership	Public

2.22. ALVAND (IRAN ATOMIC ENERGY ORGANIZATION, ISLAMIC REPUBLIC OF IRAN)

2.22.1. Introduction

The Alvand tokamak is one of the three tokamaks operating in the Islamic Republic of Iran.

2.22.2. Purpose

Alvand is used to measure plasma information and study its physical properties.

2.22.3. Main features

Alvand tokamak is a small size tokamak with circular plasma cross section [21]. Technical information is listed in Table 28 below.

TABLE 28. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.456 m
Minor radius, a	0.126 m
Plasma current, I_p	0.035 MA
Toroidal field, B_o	0.8 T
Pulse length	0.009 s
Magnetic field configuration	Circular limiter
Ownership	Public

2.23. DAMAVAND (IRAN ATOMIC ENERGY ORGANIZATION, ISLAMIC REPUBLIC OF IRAN)

2.23.1. Introduction

Damavand tokamak was built from Russian tokamak TVD, reducing plasma elongation from 4 to 2.

2.23.2. Purpose

Damavand research is devoted to studies of plasma discharges with magnetic configuration similar to ITER tokamak.

2.23.3. Main features

Damavand is a small tokamak with an elongated plasma cross section and a poloidal divertor. Its passive coils inside the vacuum chamber provide the plasma formation at the centre of the torus, acting as a passive plasma current stabilizer [22]. Technical information is listed in Table 29 below.

TABLE 29. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.362 m
Minor radius, a	0.077 m
Plasma current, I_p	0.03 MA
Toroidal field, B_o	0.9 T
Pulse length	0.02 s
Ownership	Public

2.24. IR-T1 (ISLAMIC AZAD UNIVERSITY, ISLAMIC REPUBLIC OF IRAN)

2.24.1. Introduction

IR-T1 is based on a tokamak called HT-6B, which was originally built in China in 1984. Currently IR-T1 is located at Plasma Physics Research Center of Islamic Azad University in Tehran, Islamic Republic of Iran. First plasma was obtained in 1994.

2.24.2. Purpose

The main purpose of IR-T1 is studying plasma information under different experimental conditions.

2.24.3. Main features

IR-T1 is a small tokamak with large aspect ratio and with a circular cross section. The tokamak chamber is made of stainless steel and is surrounded by leaden walls. It consists of two poloidal stainless-steel limiters with a radial thickness of 2.5 cm [23]. Technical information is listed in Table 30 below.

TABLE 30. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.45 m
Minor radius, a	0.125 m
Plasma current, I_p	0.04 MA
Toroidal field, B_o	0.9 T
Pulse length	0.03 s
Magnetic field configuration	Ring limiter (tungsten)
Ownership	Public

2.25. DTT (ENEA, ITALY)

2.25.1. Introduction

The DTT is a tokamak being constructed at ENEA centre in Frascati, Italy. Operation is expected to start by 2026.

2.25.2. Purpose

The main purpose of DTT is to study strategies for the management of plasma exhaust in a reactor-grade tokamak plasma in support of ITER operation and DEMO design studies.

2.25.3. Main features

The DTT will be a superconducting tokamak. It will have a divertor configuration that can work in single and double null scenarios, as well as a number of advanced divertor configurations. The maximum performance of auxiliary heating power will be 45 MW [24]. Technical information is listed in Table 31 below.

TABLE 31. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Under construction
Major radius, R_o	2.19 m
Minor radius, a	0.7 m
Plasma current, I_p	5.5 MA
Toroidal field, B_o	6 T
Pulse length	100 s
Magnetic field configuration	Divertor
Ownership	Public

2.26. FTU (ENEA, ITALY)

2.26.1. Introduction

Operating since 1990, FTU was the first tokamak that performed experiments with a liquid lithium limiter.

2.26.2. Purpose

The main purpose of FTU tokamak is to study and optimize techniques of reduction of the impurities in plasma using different elements, i.e., liquid metal first wall materials [39]. Most of FTU experiments since 2018 are devoted to studies on liquid metal limiters, runaway electrons and MHD stability.

2.26.3. Main features

FTU is a medium-sized tokamak with a high toroidal magnetic field, a circular poloidal cross section and metallic first wall. The vacuum chamber is made of 2 mm thick stainless steel and is lined internally with 2 cm thick toroidal molybdenum tile limiter. The machine also features an external poloidal molybdenum limiter [25]. Technical information is listed in Table 32 below.

TABLE 32. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.935 m
Minor radius, a	0.3 m
Plasma current, I_p	0.04 MA
Toroidal field, B_o	8 T
Pulse length	1.5 s
Magnetic field configuration	Limiter
Ownership	Public

2.27. LATE (KYOTO UNIVERSITY, JAPAN)

2.27.1. Introduction

LATE is a fusion device located at Kyoto University, Japan.

2.27.2. Purpose

The purpose of the LATE device is to investigate and establish the physical bases on formation of low aspect ratio torus plasmas by electron cyclotron heating and electron cyclotron current drive solely.

2.27.3. Main features

LATE is a spherical tokamak without central solenoid. The non-inductive start-up is achieved by using electron cyclotron heating and electron cyclotron current drive [26]. General information is listed in Table 33 below.

TABLE 33. GENERAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Ownership	Public

2.28. PLATO (KYUSHU UNIVERSITY, JAPAN)

2.28.1. Introduction

PLATO is an experimental fusion device currently being developed and planned to be located at Kyushu University, Japan.

2.28.2. Purpose

Research on PLATO will focus on the physics of turbulent plasmas and their understanding.

2.28.3. Main features

PLATO will be the first device able to measure the entire cross-sections of turbulent plasma with a spatial resolution of microscale of Lamour radius. The device will allow to explore plasma cross-scale couplings, turbulence localization and principles of structural formation and function expression, by using tomography, heavy ion beam probe and microwaves diagnostics [27]. Technical information is listed in Table 34 below.

TABLE 34. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Planned
Major radius, R_o	0.7 m
Minor radius, a	0.25 m
Plasma current, I_p	0.075 MA
Toroidal field, B_o	0.3 T
Pulse length	0.2 s
Ownership	Public

2.29. QUEST (KYUSHU UNIVERSITY, JAPAN)

2.29.1. Introduction

QUEST is a device located at Kyushu University, Japan. It was built in 2008, becoming the largest spherical tokamak in Japan.

2.29.2. Purpose

The main purpose of QUEST research programme is to develop an integrated understanding of particle balance in the plasma core, scrape-off layer, and plasma facing walls.

2.29.3. Main features

QUEST is a medium-sized spherical tokamak with two plasma heating sources with frequency and power of 8.2 GHz, 50 kW and 28 GHz, 350 kW, respectively [28]. Technical information is listed in Table 35 below.

TABLE 35. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.64 m
Minor radius, a	0.4 m
Plasma current, I_p	0.3 MA
Toroidal field, B_o	0.25 T
Ownership	Public

2.30. HYBTOK-II (NAGOYA UNIVERSITY, JAPAN)

2.30.1. Introduction

Built in 1997, HYBTOK-II is a tokamak built by and located at Nagoya University, Japan.

2.30.2. Purpose

HYBTOK-II was built to study particle transport characteristics in the edge plasma.

2.30.3. Main features

HYBTOK-II is a small conventional tokamak with a set of magnetic field coils installed to perturbate the magnetic field confining the plasma from the outside [29]. Technical information is listed in Table 36 below.

TABLE 36. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.4 m
Minor radius, a	0.128 m
Plasma current, I_p	0.015 MA
Toroidal field, B_o	0.5 T
Pulse length	0.01 s
Magnetic field configuration	Ergodic divertor
Ownership	Public

2.31. TOKASTAR-2 (NAGOYA UNIVERSITY, JAPAN)

2.31.1. Introduction

TOKASTAR-2 is device with a variable configuration, which can operate either as a tokamak or a stellarator. TOKASTAR-2 is the successor of TOKASTAR and C-TOKASTAR devices and operational since 2009.

2.31.2. Purpose

The main purpose of TOKASTAR-2 is to investigate the effects of outer helical field application on tokamak plasmas.

2.31.3. Main features

On TOKASTAR-2, a set of magnetic coils can generate a variable configuration allowing to operate the machine either as a tokamak or a stellarator, independently [30]. Technical information is listed in Table 37 below.

TABLE 37. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.1 m
Minor radius, a	0.04 m
Plasma current, I_p	0.025 MA
Toroidal field, B_o	0.1 T
Ownership	Public

2.32. JT-60SA (NATIONAL INSTITUTES FOR QUANTUM AND RADIOLOGICAL SCIENCE AND TECHNOLOGY, JAPAN)

2.32.1. Introduction

JT-60SA is the upgrade version of the JT-60U tokamak and is located at Naka Fusion Institute, Japan. Its commissioning started in 2020 and it was interrupted due to insufficient voltage insulation capability in one of the magnetic coils. Improvement for isolation capability is on-going and integrated commissioning is expected to restart early in 2023. JT-60SA is a joint project between Japan and the European Union.

2.32.2. Purpose

The main purpose of the JT-60SA project is to support fusion R&D by addressing key physics and technology issues relevant for ITER operation and DEMO design studies.

2.32.3. Main features

JT-60SA is a fully superconducting tokamak capable of confining high temperature deuterium plasmas. It can operate with both single and double null divertor configurations and with a wide range of plasma shapes and aspect ratios [31]. The total heating power is of 41 MW. Technical information is listed in Table 38 below.

TABLE 38. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating (under repair until 2023)
Major radius, R_o	2.96 m
Minor radius, a	1.18 m
Plasma current, I_p	5.5 MA
Toroidal field, B_o	2.25 T
Pulse length	100 s
Magnetic field configuration	Single and double null divertor
Ownership	Public

2.33. TST-2 (THE UNIVERSITY OF TOKYO, JAPAN)

2.33.1. Introduction

Operating since 1999, TST-2 is located at University of Tokyo, Japan. TST-2 is the upgraded version of TST-M.

2.33.2. Purpose

The main purpose of TST-2 is to study helicity injection and turbulence-induced transport.

2.33.3. Main features

TST-2 is a medium-sized spherical tokamak. Recently manufactured, the vacuum vessel is continuous in the toroidal direction and consists of a 1.4 m diameter, 6 mm thick stainless-steel cylinder and top and bottom domes [32]. Technical information is listed in Table 39 below.

TABLE 39. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.36 m
Minor radius, a	0.23 m
Plasma current, I_p	0.2 MA
Toroidal field, B_o	0.3 T
Pulse length	0.05 s
Ownership	Public

2.34. UTST (THE UNIVERSITY OF TOKYO, JAPAN)

2.34.1. Introduction

UTST is a spherical tokamak located at University of Tokyo, Japan. Its construction started in 2004 and became operational in 2007.

2.34.2. Purpose

The main objectives of UTST are to study central solenoid-free start-up scheme and to obtain high-beta spherical tokamak equilibria by using the plasma merging technique.

2.34.3. Main features

UTST is a medium-sized spherical tokamak without poloidal field coils [33]. Technical information is listed in Table 40 below.

TABLE 40.TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.35 m
Minor radius, a	0.2 m
Plasma current, I_p	0.1 MA
Toroidal field, B_o	0.25 T
Pulse length	0.01 s
Ownership	Public

2.35. PHIX (TOKYO INSTITUTE OF TECHNOLOGY, JAPAN)

2.35.1. Introduction

PHiX is a tokamak located at Tokyo Institute of Technology, Japan, operational since 2014.

2.35.2. Purpose

The main purpose of PHiX is to study both longitudinal cross-section and vertical position stability of fusion plasma [34].

2.35.3. Main features

PHiX is a small-sized tokamak. Technical information is listed in Table 41 below.

TABLE 41. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.33 m
Minor radius, a	0.09 m
Plasma current, I_p	0.005 MA
Toroidal field, B_o	0.3 T
Pulse length	0.02 s
Ownership	Public

2.36. HIST (UNIVERSITY OF HYOGO, JAPAN)

2.36.1. Introduction

HIST is a spherical tokamak located at University of Hyogo, Japan.

2.36.2. Purpose

The main purpose of HIST is to study the helicity injection physics on the spherical tokamak-line.

2.36.3. Main features

HIST is a spherical tokamak with stainless-steel vacuum vessel, 9 mm thick, 1.5 m in diameter and 3 m long [35]. Technical information is listed in Table 42 below.

TABLE 42. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.3 m
Minor radius, a	0.24 m
Plasma current, I_p	0.1 MA
Toroidal field, B_o	0.2 T
Ownership	Public

2.37. KTM (INSTITUTE OF ATOMIC ENERGY OF NATIONAL NUCLEAR CENTER OF THE REPUBLIC OF KAZAKHSTAN, KAZAKHSTAN)

2.37.1. Introduction

The KTM tokamak is operating since 2017 at Institute of Atomic Energy of National Nuclear Center of the Republic of Kazakhstan.

2.37.2. Purpose

The main purpose of the KTM tokamak is to study and test materials and design solutions for protecting the fusion reactor first wall, to develop methods to reduce the heat loads on divertor plates and divertor units as well as various methods of heat and energy removal, including ways to quickly pump out the divertor volume, and develop methods for preventing out-of-order failure of intra-chamber elements.

2.37.3. Main features

KTM is a spherical tokamak with aspect ratio equal to 2, plasma elongation κ_{95}=1.7 and radiofrequency heating power of 5–7 MW. Its divertor design, which consists of plasma-facing plates mounted on a rotary table, is an asset for materials testing campaigns, allowing swiftly replacements of the divertor plates [36]. Technical information is listed in Table 43 below.

TABLE 43. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.9 m
Minor radius, a	0.45 m
Plasma current, I_p	0.75 MA
Toroidal field, B_o	1 T
Pulse length	5 s
Magnetic field configuration	Single null divertor
Ownership	Public

2.38. LIBTOR (TAJOURA NUCLEAR RESEARCH CENTRE, LIBYA)

2.38.1. Introduction

The LIBTOR tokamak is located at Tajoura Nuclear Research Centre, Libya.

2.38.2. Purpose

The main purpose of LIBTOR research programme is to study plasma confinement and transport models.

2.38.3. Main features

LIBTOR is a small-sized conventional tokamak. Technical information is listed in Table 44 below.

TABLE 44. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.53 m
Minor radius, a	0.115 m
Plasma current, I_p	0.12 MA
Toroidal field, B_o	4 T
Magnetic field configuration	Rail limiter
Ownership	Public

2.39. GLAST-III (PAKISTAN ATOMIC ENERGY COMMISSION, PAKISTAN)

2.39.1. Introduction

GLAST-III, located in Pakistan, Islamabad, is the upgrade from GLAST-I and GLAST-II tokamaks. These devices were developed by Pakistan Atomic Energy Commission, National Tokamak Fusion Programme of Pakistan.

2.39.2. Purpose

The main purpose of GLAST-III is to study tokamak plasma in a small dielectric vacuum vessel with electron cyclotron resonance-assisted tokamak start-up.

2.39.3. Main features

GLAST-III is a small-sized spherical tokamak with a vacuum vessel made of Pyrex glass. Technical information is listed in Table 45 below.

TABLE 45. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.2 m
Minor radius, a	0.1 m
Plasma current, I_p	0.002 MA
Toroidal field, B_o	0.0875 T
Pulse length	0.0012 s
Ownership	Public

2.40. MT-1 (PAKISTAN ATOMIC ENERGY COMMISSION, PAKISTAN)

2.40.1. Introduction

MT-1 is operating since 2018 by Pakistan Atomic Energy Commission, National Tokamak Fusion Programme of Pakistan.

2.40.2. Purpose

MT-1 is used for training purposes.

2.40.3. Main features

MT-1 is a small-sized metallic spherical tokamak, with pre-ionization source power up to 3 kW. Technical information is listed in Table 46 below.

TABLE 46. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.25 m
Minor radius, a	0.1 m
Plasma current, I_p	0.015 MA
Toroidal field, B_o	1.2 T
Pulse length	0.008 s
Magnetic field configuration	Limiter
Ownership	Public

2.41. MT-2 (PAKISTAN ATOMIC ENERGY COMMISSION, PAKISTAN)

2.41.1. Introduction

MT-2 is the upgraded and elongated version of MT-1 and is currently under construction (coils system is being manufactured). It is being developed by Pakistan Atomic Energy Commission, National Tokamak Fusion Programme of Pakistan.

2.41.2. Purpose

MT-2 will be used for training purposes.

2.41.3. Main features

MT-2 vacuum vessel can achieve a vacuum up to ~10^{-7}mbars. General information is listed in Table 47 below.

TABLE 47. GENERAL INFORMATION

Device type	Spherical Tokamak
Status	Under construction
Ownership	Public

2.42. PST (PAKISTAN ATOMIC ENERGY COMMISSION, PAKISTAN)

2.42.1. Introduction

PST tokamak is being developed by Pakistan Atomic Energy Commission, National Tokamak Fusion Programme of Pakistan. It is currently in the engineering design phase.

2.42.2. Purpose

The objectives of PST are to explore plasma information for steady-state operation of spherical tokamaks with aspect ratio equal to 2, as well as research and develop high temperature superconducting coils for tokamak and a liquid lithium divertor system. PST research programme will also focus on capacity building activities. An extensive training program will be launched for young scientists and engineers, both nationally and internationally.

2.42.3. Main features

PST will be a medium-sized spherical tokamak planned to have electron Bernstein wave heating and neutral beam injection systems. A pulsating discharge cleaning system will be used to test diagnostics [37]. Technical information is listed in Table 48 below.

TABLE 48. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Planned
Major radius, R_o	0.5 m
Minor radius, a	0.25 m
Plasma current, I_p	0.31 MA
Toroidal field, B_o	0.5 T
Ownership	Public

2.43. ISTTOK (INSTITUTO SUPERIOR TÉCNICO, PORTUGAL)

2.43.1. Introduction

ISTTOK is a device for nuclear fusion research and training located at Instituto Superior Técnico, Portugal.

2.43.2. Purpose

The main objectives of ISTTOK are to: (i) study plasma turbulence; (ii) operation and control on alternating plasma current regimes; (iii) test liquid metal limiter concepts; (iv) develop and upgrade plasma-relevant diagnostics for nuclear fusion; and (v) serve as an academic facility to train master and PhD students.

2.43.3. Main features

ISTTOK is a large aspect ratio circular cross-section iron core tokamak. ISTTOK is highly suitable for edge physics studies due to its flexibility, low operation costs and short time scale for diagnostics implementation. Moreover, ISTTOK was the first tokamak equipped with a switchable insulated-gate bipolar transistor primary circuit with a considerable larger capacitor bank (3.8 F), allowing multiple alternating current discharges [38]. Technical information is listed in Table 49 below.

TABLE 49. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.46 m
Minor radius, a	0.085 m
Plasma current, I_p	0.007 MA
Toroidal field, B_o	0.8 T
Ownership	Public

2.44. KSTAR (KOREA INSTITUTE OF FUSION ENERGY, REPUBLIC OF KOREA)

2.44.1. Introduction

KSTAR is located at Korean Institute of Fusion Energy, Republic of Korea. Operating since 2008, this device produced over 20000 plasma experimental shots.

2.44.2. Purpose

The main objectives of KSTAR research programme are to research and develop on steady-state superconducting tokamak physics and establishing a scientific and technological base for commercial fusion power plants.

2.44.3. Main features

KSTAR is a medium-sized tokamak with superconducting magnets made of Nb3Sn, active cooled in-vessel components and long-pulse non-inductive heating and current drive. The device can ensure high performance operational capability thanks to a passive stabilizer, in-vessel control coils and strong plasma shaping features [39]. Technical information is listed in Table 50 below.

TABLE 50. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	1.8 m
Minor radius, a	0.5 m
Plasma current, I_p	2 MA
Toroidal field, B_o	3.5 T
Pulse length	300 s
Ownership	Public

2.45. VEST (SEOUL NATIONAL UNIVERSITY, REPUBLIC OF KOREA)

2.45.1. Introduction

Designed and built by Seoul National University, Republic of Korea, VEST is a tokamak operating since 2013.

2.45.2. Purpose

The main objectives of VEST are to study spherical tokamak plasmas, wave heating and high beta operations, as well as divertor concepts, including Super-X divertor configuration.

2.45.3. Main features

VEST is a small-sized spherical tokamak. This device can operate in high performance with high beta thanks to key features [40]. Technical information is listed in Table 51 below.

TABLE 51. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.45 m
Minor radius, a	0.33 m
Plasma current, I_p	0.1 MA
Toroidal field, B_o	0.1 T
Magnetic field configuration	Tungsten and graphite inboard limiters
Ownership	Public

2.46. FT-2 (IOFFE INSTITUTE, RUSSIAN FEDERATION)

2.46.1. Introduction

Operating since the 1908s, FT-2 is located at Ioffe Institute, Russian Federation.

2.46.2. Purpose

FT-2 experimental research programme focuses on lower hybrid current drive studies.

2.46.3. Main features

FT-2 is a high aspect ratio conventional tokamak with high toroidal field [41]. Technical information is listed in Table 52 below.

TABLE 52. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.55 m
Minor radius, a	0.087 m
Plasma current, I_p	0.04 MA
Toroidal field, B_o	2.3 T
Magnetic field configuration	Circular limiter
Ownership	Public

2.47. GLOBUS-M2 (IOFFE INSTITUTE, RUSSIAN FEDERATION)

2.47.1. Introduction

Built at Ioffe Institute, Russian Federation, Globus-M2 is the upgraded version of Globus-M, and it operates since 2018.

2.47.2. Purpose

The main purpose of the Globus-M2 programme is to establish the scientific and technological basis for compact fusion reactor systems, compact fusion neutron sources and fusion-fission hybrid systems.

2.47.3. Main features

Globus-M2 is a spherical tokamak, which can operate in both single and double null divertor configurations. Its electromagnetic system can withstand higher currents and mechanical loads [42]. Technical information is listed in Table 53 below.

TABLE 53. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.36 m
Minor radius, a	0.24 m
Plasma current, I_p	0.5 MA
Toroidal field, B_o	1 T
Magnetic field configuration	Single and double null divertor

2.48. TUMAN-3M (IOFFE INSTITUTE, RUSSIAN FEDERATION)

2.48.1. Introduction

TUMAN-3M is operating at Ioffe Institute, Russian Federation, since 1984.

2.48.2. Purpose

The main purpose of TUMAN-3M is to study plasma confinement and reaction mechanism.

2.48.3. Main features

TUMAN-3M is a tokamak with circular cross-section and without a divertor. The vessel and circular limiters are made of Inconel, while the sector limiter is made of molybdenum [43]. Technical information is listed in Table 54 below.

TABLE 54. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.53 m
Minor radius, a	0.22 m
Plasma current, I_p	0.18 MA
Toroidal field, B_o	1.2 T
Pulse length	0.080 s
Magnetic field configuration	Circular limiter
Ownership	Public

2.49. T-15MD (NATIONAL RESEARCH CENTRE KURCHATOV INSTITUTE, RUSSIAN FEDERATION)

2.49.1. Introduction

T-15MD is the upgraded version of T-15 tokamak, which operated during 1988–1995 at Kurchatov Institute, Russian Federation. T-15MD constriction started in 2011 and finished in 2020. The start of operation was celebrated in 2021.

2.49.2. Purpose

T-15MD programme will contribute to ITER's and future fusion power plants' operation.

2.49.3. Main features

T-15MD is a medium-sized superconducting tokamak, which will be able to achieve high plasma temperature and high plasma density, thanks to three auxiliary plasma heating systems (neutral beam injection, electron cyclotron resonance – i.e., seven gyrotrons and ion cyclotron resonance heating) and current drive systems (low hybrid heating and current drive with pulse duration up to 30 s) [44]. Technical information is listed in Table 55 below.

TABLE 55. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	2.43 m
Minor radius, a	0.42 m
Plasma current, I_p	1 MA
Toroidal field, B_o	3.5 T
Pulse length	30 s
Magnetic field configuration	Divertor
Ownership	Public

2.50. GUTTA (SAINT PETERSBURG STATE UNIVERSITY, RUSSIAN FEDERATION)

2.50.1. Introduction

GUTTA is located at St. Petersburg State University, Russian Federation, and is operating since 2004.

2.50.2. Purpose

The main objectives of GUTTA are to develop and improve mathematical models applicable to large tokamaks, study the electron cyclotron resonance heating assisted breakdown and non-solenoid plasma formation in low aspect ratio tokamak, as well as develop of diagnostics and train students [45].

2.50.3. Main features

GUTTA is a small spherical tokamak. Technical information is listed in Table 56 below.

TABLE 56. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.16 m
Minor radius, a	0.08 m
Plasma current, I_p	0.18 MA
Toroidal field, B_o	1.5 T
Ownership	Public

2.51. T-11M (TROITSK INSTITUTE FOR INNOVATION AND FUSION RESEARCH, RUSSIAN FEDERATION)

2.51.1. Introduction

Built in 1985, T-11M is located at Troitsk Institute for Innovation and Fusion Research, Russian Federation.

2.51.2. Purpose

The main purpose of T-11M is to conduct research on lithium plasma facing components protection and lithium injection into scrape-off-layer plasma.

2.51.3. Main features

T-11M is a small-sized conventional tokamak with a lithium capillary-pore system limiter and a lithium collecting system in the scrape-off-layer [46]. Technical information is listed in Table 57 below.

TABLE 57. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.7 m
Minor radius, a	0.2 m
Plasma current, I_p	0.1 MA
Toroidal field, B_o	1 T
Pulse length	0.25 s
Magnetic field configuration	Capillary-pore system limiter
Ownership	Public

2.52. SMART (UNIVERSITY OF SEVILLE, SPAIN)

2.52.1. Introduction

SMART is being designed at University of Seville, Spain.

2.52.2. Purpose

SMART will serve for educational and training purposes, to study plasma confinement and stability, plasma facing materials and develop novel technical equipment essential for fusion research.

2.52.3. Main features

SMART will be a spherical tokamak, which can be operated in three phases depending on various plasma information [47]. Technical information is listed in Table 58 below.

TABLE 58. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Planned
Major radius, R_o	0.4 m
Minor radius, a	0.25 m
Plasma current, I_p	Phase 1 – 0.03 MA Phase 2 – 0.1 MA Phase 3 – 0.5 MA
Toroidal field, B_o	Phase 1 – 0.1 T Phase 2 – 0.3 T Phase 3 – 1 T
Pulse length	Phase 1 – 0.002 s Phase 2 – 0.1 s Phase 3 – 0.5 s
Magnetic field configuration	Divertor
Ownership	Public

2.53. TCV (SWISS PLASMA CENTER, SWITZERLAND)

2.53.1. Introduction

Operating since 1992, TCV is located at Swiss Federal Institute of Technology Lausanne, Swiss Plasma Center, Switzerland.

2.53.2. Purpose

The main purpose of TCV is to study plasma configurations and shapes and research magnetic confinement plasma physics.

2.53.3. Main features

TCV is a medium-sized tokamak with a highly elongated, rectangular vacuum vessel and sixteen poloidal field coils for plasma formation, equally divided into two stacks located on both sides of the vessel. An ohmic heating coil drives an inductive current into the plasma [48]. TCV plasma auxiliary heating system consists of neutral beam injector (2 MW), electron cyclotron resonance heating (3 MW) and electron cyclotron current drive (1.5 MW). Technical information is listed in Table 59 below.

TABLE 59. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.89 m
Minor radius, a	0.25 m
Plasma current, I_p	1.2 MA
Toroidal field, B_o	1.54 T
Pulse length	2.6 s in ohmic, 4 s with electron cyclotron current drive
Ownership	Public

2.54. TT-1 (THAILAND INSTITUTE OF NUCLEAR TECHNOLOGY, THAILAND)

2.54.1. Introduction

TT-1, known as HT-6M until decommissioned in 2002, is a tokamak donated to Thailand Institute of Nuclear Technology, Thailand by the Institute of Plasma Physics of the Chinese Academy of Sciences, China. Plans call for TT-1 to be commissioned by 2023.

2.54.2. Purpose

TT-1 will become the first Thai tokamak and the first fusion device in Southeast Asia. It will help to develop capacity building and development programmes in the region.

2.54.3. Main features

TT-1 is a small-sized tokamak under construction. In the recommissioning phase, supporting system of TT-1 (power supply system for the magnet, plasma control system, vacuum system, data acquisition system) will be redesigned [49]. Technical information is listed in Table 60 below.

TABLE 60. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Under construction
Major radius, R_o	0.65 m
Minor radius, a	0.2 m
Plasma current, I_p	0.1 MA
Toroidal field, B_o	1.52 T
Ownership	Public

2.55. JET (EUROFUSION, UNITED KINGDOM)

2.55.1. Introduction

Operating since 1983, JET is the world's largest tokamak in operation. It is located at Culham Centre for Fusion Energy, UK. Its scientific programme is run by EUROfusion.

2.55.2. Purpose

The main purpose of JET is to test plasma physics, systems and materials for ITER, and to study plasma behaviour in conditions and dimensions close to those of a fusion power plant.

2.55.3. Main features

JET is a large tokamak with a divertor installed at the bottom of the vacuum vessel and an ITER-like first wall made of beryllium and tungsten. JET plasma auxiliary heating system consists of neutral beam injector (34 MW), ion cyclotron resonance heating (10 MW) and lower hybrid current drive (7 MW). JET is the only operating tokamak capable of operating with tritium fuel [50]. Technical information is listed in Table 61 below.

TABLE 61. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	2.96 m
Minor radius, a	1.25 m
Plasma current, I_p	5 MA
Toroidal field, B_o	3.4 T
Pulse length	60 s
Magnetic field configuration	Tungsten divertor
Ownership	Public

2.56. ST40 (TOKAMAK ENERGY, UNITED KINGDOM)

2.56.1. Introduction

ST-40 is a fusion device constructed by UK private sector company Tokamak Energy. The company was founded in 2009.

2.56.2. Purpose

The main purpose of ST-40 is to prove the feasibility of commercial fusion energy with compact spherical tokamaks.

2.56.3. Main features

ST-40 is a compact spherical tokamak with high temperature superconducting magnets made of REBCO, manufactured in 0.1 mm thick tapes. ST-40 toroidal field coils are made of copper [51]. Technical information is listed in Table 62 below.

TABLE 62. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.5 m
Minor radius, a	0.3 m
Plasma current, I_p	2 MA
Toroidal field, B_o	3 T
Pulse length	1 s
Magnetic field configuration	Divertor
Ownership	Private

2.57. MAST-U (UKAEA, UNITED KINGDOM)

2.57.1. Introduction

MAST-U is the upgrade of MAST, which operated during 2000–2013. MAST-U's first plasma was produced in 2020.

2.57.2. Purpose

The main purpose of MAST-U is to advance plasma exhaust research and support the development of compact fusion power plants.

2.57.3. Main features

MAST-U is a low aspect ratio tokamak able to operate with a large variety of different divertor configurations, and first one to operate with Super-X divertor configuration [52]. Technical information is listed in Table 63 below.

TABLE 63. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.85 m
Minor radius, a	0.65 m
Plasma current, I_p	2 MA
Toroidal field, B_o	0.52 T
Pulse length	5 s
Magnetic field configuration	Super-X divertor
Ownership	Public

2.58. HBT-EP (COLUMBIA UNIVERSITY, UNITED STATES OF AMERICA)

2.58.1. Introduction

Operating since 1993, HIBT-EP is located at Columbia University, USA.

2.58.2. Purpose

The main purpose of HBT-EP is to study high-beta tokamak plasma performance.

2.58.3. Main features

HIBT-EP is a conventional tokamak with adjustable walls and a vacuum chamber made with several quartz cylindrical breaks [53]. Technical information is listed in Table 64 below.

TABLE 64. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	0.94 m
Minor radius, a	0.2 m
Plasma current, I_p	0.32 MA
Toroidal field, B_o	0.3 T
Magnetic field configuration	Circular limiter
Ownership	Public

2.59. SPARC (COMMONWEALTH FUSION SYSTEMS, UNITED STATES OF AMERICA)

2.59.1. Introduction

SPARC is a joint project of US private sector company Commonwealth Fusion Systems (CFS) and MIT's Plasma Science and Fusion Center, USA. SPARC is planned to generate a net scientific energy gain[2] (Q_{sci}>1). CFS have disclosed around US$2.1 billion in fusion funding, which is almost as much as all the other (roughly 30) private sector fusion companies (see Fig. 6 on p.6). SPARC is presently under construction by CFS near Boston, USA.

2.59.2. Purpose

SPARC aims to operate with deuterium and tritium fuel and become the first fusion device that can produce a net scientific energy gain[2].

2.59.3. Main features

SPARC will be a compact high-field tokamak that will operate with, deuterium and tritium fuel. It will feature tungsten first walls and high temperature superconducting magnets. In addition to moderate ohmic heating, SPARC will rely on ion cyclotron resonance heating (25 MW) [54]. Technical information is listed in Table 65 below.

TABLE 65. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Under construction
Major radius, R_o	1.85 m
Minor radius, a	0.57 m
Plasma current, I_p	8.7 MA
Toroidal field, B_o	12.2 T
Ownership	Private

2.60. DIII-D (GENERAL ATOMICS, UNITED STATES OF AMERICA)

2.60.1. Introduction

DIII-D is owned by the US Department of Energy and is located at and managed on their behalf by General Atomics in San Diego with a large team of collaborating institutions.

2.60.2. Purpose

The main purpose of DIII-D is to develop the plasma solutions and scientific basis to project them for future fusion reactors. In particular it targets the development of a high performance fusion core and a compatible power handling solutions to enable a compact fusion power plant, as well as a range of associated plasma interacting fusion technologies. It also seeks to develop the operational approach and modelling tools to help ITER maximize performance and reach its goals rapidly.

2.60.3. Main features

DIII-D's key feature is its very high degree of flexibility to explore wide range of plasma configurations and develop solutions for future reactors. This couples with a highly extensive diagnostic set in order to identify and resolve models of the underlying physics needed to project to solutions developed. These flexibilities include plasma shapes from highly positive to negative triangularity, single or double null operation, flexible open and closed divertors, particle control through cryo-pumping, wall conditioning, gas and inboard or outboard pellet injection, 3 arrays of 3-D magnetic perturbation coils, and independent control of heating, current drive, torque and profile breadth [55]. Technical information is listed in Table 66 below.

TABLE 66. TECHNICAL INFORMATION

Device type	Conventional Tokamak
Status	Operating
Major radius, R_o	1.7 m
Minor radius, a	0.6 m
Plasma current, I_p	2 MA
Toroidal field, B_o	2.17 T
Pulse length	10 s
Magnetic field configuration	Divertor, single or double null
Ownership	Public

2.61. LTX-B (PRINCETON PLASMA PHYSICS LABORATORY, UNITED STATES OF AMERICA)

2.61.1. Introduction

Operating since 2020, LTX-β is located at Princeton Plasma Physics Laboratory, USA.

2.61.2. Purpose

The purpose of LTX-β is to research and develop plasma facing components based on liquid lithium.

2.61.3. Main features

LTX-β is a spherical tokamak with low aspect ratio, with a lithium coated shell made of 5 mm stainless steel bonded to 1 cm thick copper, without use of any low Z materials [56]. Technical information is listed in Table 67 below.

TABLE 67. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.4 m
Minor radius, a	0.26 m
Plasma current, I_p	0.175 MA
Toroidal field, B_o	3.7 T
Ownership	Public

2.62. NSTX-U (PRINCETON PLASMA PHYSICS LABORATORY, UNITED STATES OF AMERICA)

2.62.1. Introduction

NSTX-U is the upgraded version of NSTX. NSTX-U started operating in 2016 but was paused to make repairs. It is expected to restart in 2022.

2.62.2. Purpose

The objectives of NSTX-U are: (i) to study plasma control scenarios in a high-performance spherical tokamak; (ii) to investigate non-inductive plasma smart-up; and (iii) to study heat exhaust solutions.

2.62.3. Main features

NSTX-U is a spherical tokamak with plasma facing components made of carbon-based materials. Its vacuum vessel is made of stainless steel. The plasma heating and current drive systems include the high harmonic fast wave, coaxial helicity injection current drive, the electron cyclotron resonance and neutral beam injection systems [57]. Technical information is listed in Table 68 below.

TABLE 68. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.934 m
Minor radius, a	0.68 m
Plasma current, I_p	2 MA
Toroidal field, B_o	1 T
Pulse length	5 s
Magnetic field configuration	Divertor
Ownership	Public

2.63. PEGASUS-III (UNIVERSITY OF WISCONSIN-MADISON, UNITED STATES OF AMERICA)

2.63.1. Introduction

Pegasus is located at University of Wisconsin-Madison, USA.

2.63.2. Purpose

The research purpose of Pegasus is to study non-inductive plasma start up and operational scenarios.

2.63.3. Main features

Pegasus is a spherical tokamak with stainless steel vacuum vessel [58]. Technical information is listed in Table 69 below.

TABLE 69. TECHNICAL INFORMATION

Device type	Spherical Tokamak
Status	Operating
Major radius, R_o	0.45 m
Minor radius, a	0.37 m
Plasma current, I_p	1 MA
Toroidal field, B_o	0.58 T
Ownership	Public

3. EXPERIMENTAL STELLARATORS/HELIOTRONS

3.1. CFQS (SOUTHWEST JIAOTONG UNIVERSITY, CHINA)

3.1.1. Introduction

CFQS is a stellarator being constructed as a cooperative project between the National Institute of Fusion Science, Japan and the Southwest Jiaotong University, China. It will be the first ever quasi-axisymmetric stellarator.

3.1.2. Purpose

The purpose of CFQS is to research and develop the stellarator concept line.

3.1.3. Main features

CFQS will be a quasi-axisymmetric stellarator. It will feature four poloidal field coils and twelve toroidal field coils. Its vacuum vessel will be made of SUS316L with thickness of 6 mm [59]. Technical information is listed in Table 70 below.

TABLE 70. TECHNICAL INFORMATION

Device type	Stellarator
Status	Planned
Major radius, R_o	1 m
Minor radius, a	0.25 m
Toroidal field, B_o	1 T
Ownership	Public

3.2. SCR-1 (INSTITUTO TECNOLOGICO DE COSTA RICA, COSTA RICA)

3.2.1. Introduction

Operating since 2016, SCR-1 is a stellarator designed and built by Costa Rica Institute of Technology, Costa Rica, and the first fusion device in Latin America.

3.2.2. Purpose

The purpose of SCR-1 is to support education and training in plasma physics and fusion research in Costa Rica and in the Latin America region.

3.2.3. Main features

SCR-1 is the smallest stellarator in the world. It features twelve magnetic coils [60]. Technical information is listed in Table 71 below.

TABLE 71. TECHNICAL INFORMATION

Device type	Stellarator
Status	Operating
Major radius, R_o	0.238 m
Minor radius, a	0.042 m
Plasma current, I_p	0.04 MA
Toroidal field, B_o	0.0878 T
Ownership	Public

3.3. RENAISSANCE FUSION (RENAISSANCE FUSION, FRANCE)

3.3.1. Introduction

Renaissance fusion is a French private sector company working on stellarators research and development and developing a stellarator concept [61].

3.3.2. Purpose

The purpose of Renaissance fusion is to put fusion electricity in the grid using the stellarator concept.

3.3.3. Main features

Renaissance Fusion's stellarator design is intended to feature high temperature superconducting magnets and operate with flowing liquid metal walls for the plasma facing components. Technical information is listed in Table 72 below.

TABLE 72. GENERAL INFORMATION

Device type	Stellarator
Status	Planned
Toroidal field, B_0	10 T
Ownership	Private

3.4. WENDELSTEIN 7-X (MAX PLANK INSTITUTE FOR PLASMA PHYSICS, GERMANY)

3.4.1. Introduction

Wendelstein 7-X is located at Max Planck Institute in Greifswald, Germany. Operating since 2015, it is the largest stellarator device in the world.

3.4.2. Purpose

The main purpose of Wendelstein 7-X is to prove the possibility of using the stellarator concept for power plant designs, by operating up to 30 minutes with a heating power up to 10 MW [62].

3.4.3. Main features

Wendelstein 7-X is a large stellarator type fusion device with modular superconducting coils. The vessel needs to accurately repeat the swirling shape of the plasma and it is made of stainless steel segments 17 mm thick, accurately bent and welded to each other. The magnetic system of Wendelstein 7-X includes 20 planar and 50 non-planar superconducting magnetic coils. Since 2022 all plasma facing components are water-cooled. Technical information is listed in Table 73 below.

TABLE 73. TECHNICAL INFORMATION

Device type	Stellarator
Status	Operating
Major radius, R_o	5.5 m
Minor radius, a	0.53 m
Toroidal field, B_o	3 T
Ownership	Public

3.5. TJ-K (UNIVERSITY OF STUTTGART, GERMANY)

3.5.1. Introduction

TJ-K is a stellarator which was built by CIEMAT, Madrid and it is located at the University of Stuttgart, Germany, since 2005.

3.5.2. Purpose

The main purpose of TJ-K is to carry out low temperature plasmas research.

3.5.3. Main features

TJ-K consists of two vertical field coils and one helical coil, which is wrapped six times around the vacuum vessel. These three coils are generating toroidally closed magnetic flux surfaces. TJ-K operates with low temperature plasmas [63]. Technical information is listed in Table 74 below.

TABLE 74. TECHNICAL INFORMATION

Device type	Stellarator
Status	Operating
Major radius, R_o	0.6 m
Minor radius, a	0.1 m
Toroidal field, B_o	0.07 T
Ownership	Public

3.6. HELIOTRON J (KYOTO UNIVERSITY, JAPAN)

3.6.1. Introduction

Heliotron J was designed and manufactured by Kyoto University, Japan, in 2000.

3.6.2. Purpose

The main objectives of Heliotron J are to study the concept of the non-symmetric quasi-isodynamic approach to the heliotron line of stellarators optimization with a helical axis, as well as to investigate the physics basis for the heliotron helical axis.

3.6.3. Main features

Heliotron J is a stellarator with a three-dimensional magnetic axis, so called helical magnetic axis. The magnetic coil system includes a continuous helical coil, two types of toroidal coils and three pairs of vertical coils. The heating system consists of electron cyclotron resonance heating (70 GHz, 0.4 MW), 2 neutral beam injectors (30 kV, 0.7MW) and 2 ion cyclotron resonance heating (16–24 MHz, 0.4 MW) [64]. Technical information is listed in Table 75 below.

TABLE 75. TECHNICAL INFORMATION

Device type	Heliotron
Status	Operating
Major radius, R_o	1.2 m
Minor radius, a	0.17 m
Toroidal field, B_o	1.5 T
Ownership	Public

3.7. LHD (NATIONAL INSTITUTE FOR FUSION SCIENCE, JAPAN)

3.7.1. Introduction

Operating since 1998, the LHD fusion device is one of the largest stellarators in the world. LHD is located at the National Institute for Fusion Science, Japan.

3.7.2. Purpose

The main purpose of LHD is to contribute to research on the development of helical-type fusion devices.

3.7.3. Main features

LHD is a large heliotron type device. It operates using superconducting coils, various plasma heating systems and multiple diagnostics [65]. Technical information is listed in Table 76 below.

TABLE 76. TECHNICAL INFORMATION

Device type	Heliotron
Status	Operating
Major radius, R_o	3.9 m
Minor radius, a	0.65 m
Toroidal field, B_o	4 T
Ownership	Public

3.8. TJ-II (CIEMAT, SPAIN)

3.8.1. Introduction

Operating since 1997, TJ-II is a joint project between the National Fusion Laboratory, Spain and Oak Ridge National Laboratory, USA. It was also partly financed by EURATOM. After Wendelstein 7-X, TJ-II is the second largest stellarator in Europe.

3.8.2. Purpose

The main purpose of TJ-II is to study magnetic configuration influence on heat and particle transport.

3.8.3. Main features

TJ-II is a flexible, medium-sized stellarator. It operates with 32 toroidal field coils, one circular coil, one helical coil and a set of vertical field coils. Power heating is provided by means of microwave heating (800 kW, 53 GHz) and neutral beam injector (1.6 MW, 30 keV) [66]. Technical information is listed in Table 77 below.

TABLE 77. TECHNICAL INFORMATION

Device type	Stellarator
Status	Operating
Major radius, R_o	1.5 m
Minor radius, a	0.22 m
Toroidal field, B_o	0.95 T
Ownership	Public

3.9. URAGAN-2M (INSTITUTE OF PLASMA PHYSICS NATIONAL SCIENCE CENTER, UKRAINE)

3.9.1. Introduction

Operating since 2006, Uragan-2M is a located at Institute of Plasma Physics National Science Center in Kharkiv Institute of Physics and Technology, Ukraine.

3.9.2. Purpose

The main purpose of Uragan-2M is to conduct studies on the influence of helical magnetic field inhomogeneities in plasma confinement.

3.9.3. Main features

Uragan-2M is a stellarator. Its configuration with reduced helical magnetic can moderate sheers and magnetic wells. Uragan-2M features with 16 toroidal field coils and 2 helical coils. The large number of magnetic windings makes it possible to vary the information of the magnetic configuration over a wide range. Technical information is listed in Table 78 below.

TABLE 78. TECHNICAL INFORMATION

Device type	Stellarator
Status	Operating
Major radius, R_o	1.7 m
Minor radius, a	0.22 m
Toroidal field, B_o	2.4 T
Ownership	Public

3.10. URAGAN-3M (INSTITUTE OF PLASMA PHYSICS NATIONAL SCIENCE CENTER, UKRAINE)

3.10.1. Introduction

Uragan-3M is located at Institute of Plasma Physics National Science Center in Kharkiv Institute of Physics and Technology, Ukraine.

3.10.2. Purpose

The main purpose of Uragan-3M is to conduct research on radiofrequency plasma production and heating, stellarator plasma physics and divertor study in the stellarator-line research.

3.10.3. Main features

Uragan-3M is a medium-sized stellarator. Its magnetic system is located inside the vacuum chamber, allowing for helical divertor configuration. Technical information is listed in Table 79 below.

TABLE 79. TECHNICAL INFORMATION

Device type	Stellarator
Status	Operating
Major radius, R_o	1 m
Minor radius, a	0.12 m
Toroidal field, B_o	0.7 T
Ownership	Public

3.11. CTH (AUBURN UNIVERSITY, UNITED STATES OF AMERICA)

3.11.1. Introduction

Operating since 2005, CTH is located at Auburn University, USA.

3.11.2. Purpose

The main purpose of CTH is to investigate magnetohydrodynamics instabilities and disruptions in conductive stellarator plasmas.

3.11.3. Main features

CTH is a stellarator of torsatron type. Its magnetic system consists of external helical, vertical, and toroidal field coils. In addition, CTH can also operate with a set of ohmic coils typical of pulsed tokamak designs. The vacuum vessel is made of Inconel alloy 625 and has a circular torus shape [67]. Technical information is listed in Table 80 below.

TABLE 80. TECHNICAL INFORMATION

Device type	Torsatron
Status	Operating
Major radius, R_o	0.75 m
Minor radius, a	0.29 m
Plasma current, I_p	0.08 MA
Toroidal field, B_o	0.7 T
Ownership	Public

3.12. HIDRA (UNIVERSITY OF ILLINOIS, UNITED STATES OF AMERICA)

3.12.1. Introduction

Formerly known as WEGA, HIDRA is located at University of Illinois, USA. It operates since 2016.

3.12.2. Purpose

The main purpose of HIDRA is to study plasma material interactions and develop the technology essential for innovative plasma facing components (e.g., using liquid lithium technologies).

3.12.3. Main features

HIDRA is a medium-sized device which can operate either as a stellarator or tokamak. Its vacuum vessel has a circular shape. Plasma in HIDRA is generated using magnetron heating (26 kW, 2.45 GHz). Technical information is listed in Table 81 below.

TABLE 81. TECHNICAL INFORMATION

Device type	Stellarator/Tokamak
Status	Operating
Major radius, R_o	0.72 m
Minor radius, a	0.19 m
Toroidal field, B_o	0.5 T
Ownership	Public

3.13. HSX (UNIVERSITY OF WISCONSIN-MADISON, UNITED STATES OF AMERICA)

3.13.1. Introduction

Operating since 1999, HSX is located at University of Wisconsin-Madison, USA. HSX has a unique structure of the magnetic field, called the Quasi-Helically Symmetric (QHS).

3.13.2. Purpose

The main purpose of HSX is to conduct research in investigation of transport, turbulence and confinement in a QHS magnetic field.

3.13.3. Main features

The QHS configuration of HSX is achieved thanks to 48 magnetic coils and a set of 12 auxiliary coils [100]. HSX vacuum vessel is made of stainless steel. Plasma is generated using gyrotron system (200 kW, 28 GHz) [68]. Technical information is listed in Table 82 below.

TABLE 82. TECHNICAL INFORMATION

Device type	Stellarator
Status	Operating
Major radius, R_o	1.2 m
Minor radius, a	0.15 m
Toroidal field, B_o	1.25 T
Ownership	Public

4. EXPERIMENTAL INERTIAL/LASER FUSION DEVICES

4.1. HB11 (HB11 ENERGY, AUSTRALIA)

4.1.1. Introduction

HB11 is a laser fusion device being developed by HB11 Energy, an Australian private sector company launched in 2019 [61].

4.1.2. Purpose

The purpose is to show the economic advantage of p-^{11}B fusion reaction for energy production.

4.1.3. Main features

The laser system will be composed of a nanosecond pulse laser and a picosecond pulse laser. The device will produce energy via p-^{11}B fusion reactions. General information is listed in Table 83 below.

TABLE 83. GENERAL INFORMATION

Device type	Laser Fusion
Status	Planned
Ownership	Private

4.2. LMJ (CEA, FRANCE)

4.2.1. Introduction

LMJ is located at CEA, France. It was commissioned at the end of 2014. Since 2017, it has been merged with the high-power PETAL laser.

4.2.2. Purpose

The main purpose of LMJ is to support high energy density physics studies and inertial confinement fusion experiments.

4.2.3. Main features

LMJ is designed with multiple lines of a three-frequency neodymium glass laser beam capable of irradiating targets at a wavelength of 351 nm [69]. General information is listed in Table 84 below.

TABLE 84. GENERAL INFORMATION

Device type	Laser Fusion
Status	Operating
Ownership	Public

4.3. MARVEL FUSION (MARVEL FUSION, GERMANY)

4.3.1. Introduction

Established in 2018, Marvel Fusion is a German private sector company planning to develop laser fusion device technology [61].

4.3.2. Purpose

The main purpose of Marvel Fusion is to show the feasibility of commercial laser fusion via p-^{11}B reactions.

4.3.3. Main features

Marvel Fusion concept is based on ultra-short pulses, high intensity lasers and manufactured nanostructured fuel pellets triggering p-^{11}B fusion reactions. General information is listed in Table 85 below.

TABLE 85. GENERAL INFORMATION

Device type	Laser Fusion
Status	Planned
Ownership	Private

4.4. GEKKO XII (OSAKA UNIVERSITY, JAPAN)

4.4.1. Introduction

Built in 1983 and located at Osaka University, Japan, the GEKKO XII is a large-scale laser facility capable of producing high temperature plasma (hundred million degrees Celsius) and compressing matter to density greater than 600 times its solid density.

4.4.2. Purpose

The main purpose of GEKKO XII is to support high energy density physics studies and inertial confinement fusion experiments.

4.4.3. Main features

GEKKO XII is a large scale powerful 12-beam glass laser. Its output energy is 20 kJ, and its peak power is 40 TW. The GEKKO XII laser operates with two vacuum chambers. In chamber, the target can be irradiated by 12 laser beams with spherical symmetry, while in another chamber, the 12 beams are combined into a single laser beam which allows to irradiate the target with high-power from a single direction [70]. General information is listed in Table 86 below.

TABLE 86. GENERAL INFORMATION

Device type	Laser Fusion
Status	Operating
Ownership	Public

4.5. LFEX (OSAKA UNIVERSITY, JAPAN)

4.5.1. Introduction

LFEX is a laser fusion facility at Osaka University, Japan.

4.5.2. Purpose

The main purpose of LFEX is to conduct fusion research and study relativistic plasma interactions and laser-induced nuclear physics.

4.5.3. Main features

LFEX is an ultra-high intensity laser facility designed to produce an output energy of 10 kJ [71]. General information is listed in Table 87 below.

TABLE 87. GENERAL INFORMATION

Device type	Laser Fusion
Status	Operating
Ownership	Public

4.6. FIRST LIGHT (FIRST LIGHT FUSION LTD, UNITED KINGDOM)

4.6.1. Introduction

First Light Fusion Ltd is a UK private sector company developing inertial fusion technology [61].

4.6.2. Purpose

First Light Fusion purpose is to achieve energy generation by inertial confinement fusion using shockwaves.

4.6.3. Main features

In First Light Fusion's device, high temperatures and compressions required for fusion are achieved using a gas gun that launches a projectile into a vacuum chamber at speeds of more than 6.5 km/s. General information is listed in Table 88 below.

TABLE 88. GENERAL INFORMATION

Device type	Inertial Fusion
Status	Operating
Ownership	Private

4.7. INNOVEN ENERGY LLC (INNOVEN ENERGY, UNITED STATES OF AMERICA)

4.7.1. Introduction

Innoven Energy is a US private sector company founded in 2010 and working on laser fusion R&D.

4.7.2. Purpose

The purpose of Innoven Energy is to show the possibility of producing controlled and safe inertial fusion energy.

4.7.3. Main features

The Innoven Energy's device is planned to be based on a novel laser architecture that allows the nanosecond time compression of laser pulses with extremely high optical quality at low cost. General information is listed in Table 89 below.

TABLE 89. GENERAL INFORMATION

Device type	Laser Fusion
Status	Planned
Ownership	Private

4.8. NIF (LAWRENCE LIVERMORE NATIONAL LABORATORY, UNITED STATES OF AMERICA)

4.8.1. Introduction

Operating since 2009 and located at Lawrence Livermore National Laboratory, USA, NIF is the largest laser facility in the world.

4.8.2. Purpose

The main purpose of NIF's lasers includes laser fusion R&D, high energy density science, energy security, and building future generations of scientists.

4.8.3. Main features

NIF is the largest and highest energy laser system in the world. NIF consists of 192 high energy, finely focused laser beams that converge at the centre of the target chamber [72]. General information is listed in Table 90 below.

TABLE 90. GENERAL INFORMATION

Device type	Laser Fusion
Status	Operating
Ownership	Public

4.9. OMEGA (UNIVERSITY OF ROCHESTER LABORATORY FOR LASER ENERGETICS, UNITED STATES OF AMERICA)

4.9.1. Introduction

Operating since 1995, the OMEGA laser is located at University of Rochester, USA.

4.9.2. Purpose

The main purpose of OMEGA is to advance high energy density science and conduct research on the interaction of a laser with a substance of ultrahigh intensity.

4.9.3. Main features

OMEGA consists of 60 laser beams capable of focusing up to 30 kJ of energy on a target smaller than 1 mm in diameter in about one billionth of a second [73]. General information is listed in Table 91 below.

TABLE 91. GENERAL INFORMATION

Device type	Laser Fusion
Status	Operating
Ownership	Public

5. EXPERIMENTAL ALTERNATIVE DEVICE CONCEPTS

5.1. GENERAL FUSION (GENERAL FUSION INC, CANADA)

5.1.1. Introduction

Established in 2002, General Fusion is a Canadian private company developing an experimental fusion device based on magnetized target fusion [61].

5.1.2. Purpose

The main purpose of General Fusion is to research and develop magnetized target fusion.

5.1.3. Main features

General Fusion developing magnetized target fusion based on the use of liquid metal wall in combination with lithium for tritium breeding. General information is listed in Table 92 below.

TABLE 92. GENERAL INFORMATION

Device type	Magnetized Target Fusion
Status	Under construction
Ownership	Private

5.2. KTX (UNIVERSITY OF SCIENCE AND TECHNOLOGY OF CHINA, CHINA)

5.2.1. Introduction

KTX is a reversed field pinch device located at University of Science and Technology of China.

5.2.2. Purpose

The main objective of KTS is to study resistive wall instabilities.

5.2.3. Main features

KTX is a middle-sized fusion device of reversed field pinch type. It consists of the vacuum vessel, the conducting shell, ohmic heating, plasma equilibrium, toroidal field and active feedback coils and the supporting structures [74]. Technical information is listed in Table 93 below.

TABLE 93. TECHNICAL INFORMATION

Device type	Reversed Field Pinch
Status	Operating
Major radius, R_o	1.4 m
Minor radius, a	0.4 m
Plasma current, I_p	1 MA
Ownership	Public

5.3. TORIX (ÉCOLE POLYTECHNIQUE, FRANCE)

5.3.1. Introduction

ToriX is located at Ecole Polytechnique, France.

5.3.2. Purpose

The purpose of ToriX is to support training and education in fusion research and plasma physics.

5.3.3. Main features

ToriX is a small-sized device of simple magnetized torus type. Technical information is listed in Table 94 below.

TABLE 94. TECHNICAL INFORMATION

Device type	Simple Magnetized Torus
Status	Operating
Major radius, R_o	0.6 m
Minor radius, a	0.05 m
Pulse length	0.1 s
Ownership	Public

5.4. RFX (CONSORZIO RFX, ITALY)

5.4.1. Introduction

Opiating since 1992, RFX is the largest reversed field pinch device in the world.

5.4.2. Purpose

The main objectives of RFX are to research and develop the reversed field pinch configuration and support ITER research plan.

5.4.3. Main features

RFX is a reversed field pinch device featuring ohmic plasma heating [75]. Technical information is listed in Table 95 below.

TABLE 95. TECHNICAL INFORMATION

Device type	Reversed Field Pinch
Status	Operating
Major radius, R_o	2 m
Minor radius, a	0.45 m
Plasma current, I_p	2 MA
Ownership	Public

5.5. RELAX (KYOTO INSTITUTE OF TECHNOLOGY, JAPAN)

5.5.1. Introduction

RELAX is a reversed field pinch fusion device located at the Kyoto Institute of Technology, Japan.

5.5.2. Purpose

The main purpose of RELAX is to research and develop the reversed field pinch in the low aspect ratio regime.

5.5.3. Main features

RELAX is a low aspect ratio reversed field pinch device [76]. Technical information is listed in Table 96 below.

TABLE 96. TECHNICAL INFORMATION

Device type	Reversed Field Pinch
Status	Operating
Major radius, R_o	0.5 m
Minor radius, a	0.25 m
Ownership	Public

5.6. UH-CTI (KYUSHU UNIVERSITY, JAPAN)

5.6.1. Introduction

Operating since 2005, UH-CTI is located at Kyusyu University, Japan. Since 2012, UH-CTI is installed on QUEST tokamak (see p. 47).

5.6.2. Purpose

The main purpose of UH-CTI is to study advanced fuelling in spherical tokamak plasmas.

5.6.3. Main features

UH-CTI is a compact torus used a particles injector and installed on QUEST tokamak. UH-CTI is mounted perpendicular to the magnetic axis on the midplane of QUEST [77]. General information is listed in Table 97 below.

TABLE 97. GENERAL INFORMATION

Device type	Spheromak
Status	Operating
Ownership	Public

5.7. FAT-CM (NIHON UNIVERSITY, JAPAN)

5.7.1. Introduction

FAT-CM is located at Nihon University, Japan.

5.7.2. Purpose

The purpose of FAT-CM is to investigate the collisional-merging formation process in field reversed configuration devices at super Alfvén velocity.

5.7.3. Main features

FAT-CM consists of the central confinement chamber and two field reversed theta-pinch formation sections. The formation tubes are made of transparent quartz and the confinement chamber is made of stainless steel. Initial field reversed configurations are formed with D_2 gas puffing [78]. General information is listed in Table 98 below.

TABLE 98. GENERAL INFORMATION

Device type	Field Reversed Configuration
Status	Operating
Ownership	Private

5.8. RT-1 (THE UNIVERSITY OF TOKYO, JAPAN)

5.8.1. Introduction

Operating since 2006, RT-1 is located at University of Tokyo, Japan.

5.8.2. Purpose

The main purpose of RT-1 is to demonstrate very high-beta (~1) plasma confinement.

5.8.3. Main features

RT-1 is a levitated dipole device equipped with a superconducting ring magnet that generates a dipole magnetic field with strength ranging 0.01-0.3 T. The superconducting ring is levitated in the middle of the chamber by a feedback-controlled magnet placed on the top of the device [79]. General information is listed in Table 99 below.

TABLE 99. GENERAL INFORMATION

Device type	Levitated Dipole
Status	Operating
Ownership	Public

5.9. UH-MCPG1 (UNIVERSITY OF HYOGO, JAPAN)

5.9.1. Introduction

UH-MCPG1 is located at University of Hyogo, Japan.

5.9.2. Purpose

The purpose of UH-MCPG1 is to carry out research relevant to fusion and plasma physics.

5.9.3. Main features

UH-MCPG1 is a spheromak. General information is listed in Table 100 below.

TABLE 100. GENERAL INFORMATION

Device type	Spheromak
Status	Operating
Ownership	Public

5.10. GAMMA 10/PDX (UNIVERSITY OF TSUKUBA, JAPAN)

5.10.1. Introduction

Operating since 1983, GAMMA 10/PDX is located at University of Tsukuba, Japan.

5.10.2. Purpose

The main purpose of GAMMA 10/PDX is to study divertor physics, using high heat-flux and particle flux plasma.

5.10.3. Main features

GAMMA 10/PDX is a magnetic mirror machine and the largest tandem mirror in the world. Its plasma heating system consists of radiofrequency wave and neutral beam injection [80]. General information is listed in Table 101 below.

TABLE 101. GENERAL INFORMATION

Device type	Magnetic Mirror Machine
Status	Operating
Ownership	Public

5.11. PILOT GAMMA PDX-SC (UNIVERSITY OF TSUKUBA, JAPAN)

5.11.1. Introduction

Pilot GAMMA PDX-SC is being constructed at University of Tsukuba, Japan. The installation of two superconducting coils was completed in 2021.

5.11.2. Purpose

The purpose of Pilot GAMMA PDX-SC is to serve as a pilot device for building a database of results, contributing to the construction of a future divertor plasma generator.

5.11.3. Main features

Pilot GAMMA PDX-SC will be a magnetic mirror machine with central plasma diameter ranging 0.4–1.1 m, mirror diameter of 0.1 m, plasma diameter of 0.2 m, magnetic field strength on the central axis ranging 0.05–0.1 T, and mirror end magnetic field strength of 1.5 T [80]. General information is listed in Table 102 below.

TABLE 102. GENERAL INFORMATION

Device type	Magnetic Mirror Machine
Status	Under construction
Ownership	Public

5.12. CAT (BUDKER INSTITUTE OF NUCLEAR PHYSICS, RUSSIAN FEDERATION)

5.12.1. Introduction

CAT is being constructed at Budker Institute of Nuclear Physics, Russian Federation.

5.12.2. Purpose

The main purpose of CAT is to study production and sustainment of high pressure plasmas.

5.12.3. Main features

CAT is an axisymmetric magnetic mirror machine. CAT will consist of a central chamber, a plasma gun and a plasma ejection chamber with a target for absorbing plasma flowing from a mirror trap. It will produce fast ions by injecting neutral beams (3.5 MW, 15 keV, 5 ms) into a compact axisymmetric mirror trap [81]. General information is listed in Table 103 below.

TABLE 103. GENERAL INFORMATION

Device type	Magnetic Mirror Machine
Status	Under construction
Ownership	Public

5.13. GDMT (BUDKER INSTITUTE OF NUCLEAR PHYSICS, RUSSIAN FEDERATION)

5.13.1. Introduction

GDMT is being designed and planned at Budker Institute of Nuclear Physics, Russian Federation. Its construction is expected to be completed by 2024.

5.13.2. Purpose

The main purpose of GDMT will be to evaluate and prove the feasibility of the gas dynamic mirror concept for different practical fusion applications.

5.13.3. Main features

GDMT is being designed as an axisymmetric magnetic mirror trap. It will produce pulses of 5 s duration [82]. General information is listed in Table 104 below.

TABLE 104. GENERAL INFORMATION

Device type	Magnetic Mirror Machine
Status	Planned
Ownership	Public

5.14. GDMT CORE (BUDKER INSTITUTE OF NUCLEAR PHYSICS, RUSSIAN FEDERATION)

5.14.1. Introduction

GDMT CORE is being designed and planned at Budker Institute of Nuclear Physics, Russian Federation. It will be the upgraded and final version of GDMT (see previous page).

5.14.2. Purpose

The main purpose of GDMT CORE will be to evaluate and prove the feasibility of the gas dynamic mirror concept for different practical fusion applications.

5.14.3. Main features

GDMT CORE is being designed as an axisymmetric magnetic mirror trap. GDMT CORE will be 70 m in length and produce pulses of 1000 s duration. The superconducting magnet system of GDMT CORE will consist of a central solenoid, compact mirror coils with high magnetic field and tail sections to suppress plasma flux [82]. General information is listed in Table 105 below.

TABLE 105. GENERAL INFORMATION

Device type	Magnetic Mirror Machine
Status	Planned
Ownership	Public

5.15. GDT (BUDKER INSTITUTE OF NUCLEAR PHYSICS, RUSSIAN FEDERATION)

5.15.1. Introduction

Built in 1986, GDT operates at Budker Institute of Nuclear Physics, Russian Federation.

5.15.2. Purpose

The main purpose of GDT is to study magnetic mirror machine physics.

5.15.3. Main features

GDT is a magnetic mirror machine. It is a long axial-symmetric mirror system (with mirror ratio ranging 12.5–100) that confines fast ions (produced with neutral beam injection system) and a collisional target plasma. General information is listed in Table 106 below.

TABLE 106. GENERAL INFORMATION

Device type	Magnetic Mirror Machine
Status	Operating
Ownership	Public

5.16. GOL-NB (BUDKER INSTITUTE OF NUCLEAR PHYSICS, RUSSIAN FEDERATION)

5.16.1. Introduction

GOL-NB is located at the Budker Institute of Nuclear Physics, Russian Federation. It serves for testing the main components of the larger GDMT project (see pp. 119–120).

5.16.2. Purpose

The main purpose of the GOL-NB device is to support GDMT's design and construction.

5.16.3. Main features

GOL-NB is an operational magnetic mirror machine. Its magnetic system consists of a central trap for plasma confinement and two multiply mirrors for energy and particle confinement. GOL-NB can work as a classical gas dynamic trap (with short mirrors), or a trap with long collisional mirrors, as well as with multiple mirrors system. The plasma heating system consists of two neutral beam injectors (25 keV, 0.75 MW) [83]. General information is listed in Table 107 below.

TABLE 107. GENERAL INFORMATION

Device type	Magnetic Mirror Machine
Status	Operating
Ownership	Public

5.17. SMOLA (BUDKER INSTITUTE OF NUCLEAR PHYSICS, RUSSIAN FEDERATION)

5.17.1. Introduction

SMOLA is a magnetic mirror machine located at Budker Institute of Nuclear Physics, Russian Federation.

5.17.2. Purpose

The main purpose of SMOLA is to explore plasma flow suppression in a helical magnetic field.

5.17.3. Main features

SMOLA is a magnetic mirror machine. It consists of a plasma tank, a solenoid with independent windings for uniform and helical field components, a set of bias electrodes and an exit tank with plasma receiver [84]. General information is listed in Table 108 below.

TABLE 108. GENERAL INFORMATION

Device type	Magnetic Mirror Machine
Status	Operating
Ownership	Public

5.18. EXTRAP T2R (KTH ROYAL INSTITUTE OF TECHNOLOGY, SWEDEN)

5.18.1. Introduction

Operating since 1994, EXTRAP T2R is located at Royal Institute of Technology in Stockholm, Sweden.

5.18.2. Purpose

The main objective of EXTRAP T2R is to study the stability of resistive wall modes.

5.18.3. Main features

EXTRAP T2R is a medium-sized fusion device of reversed field pinch type. Its ring-shaped plasma chamber features various access ports for insertion of material samples and probes. General information is listed in Table 109 below.

TABLE 109. GENERAL INFORMATION

Device type	Reversed Field Pinch
Status	Operating
Ownership	Public

5.19. TORPEX (SWISS PLASMA CENTER, SWITZERLAND)

5.19.1. Introduction

TORPEX is located at Center for Plasma Physics Research, Switzerland.

5.19.2. Purpose

The main purpose of TORPEX is to carry out research in plasma physics [85].

5.19.3. Main features

TORPEX is a simple magnetized torus featuring 28 toroidal magnetic coils, 4 vertical coils and a central coil stack for high toroidal loop voltage operations. Technical information is listed in Table 110 below.

TABLE 110. TECHNICAL INFORMATION

Device type	Simple Magnetized Torus
Status	Operating
Major radius, R_o	1 m
Minor radius, a	0.2 m
Plasma current, I_p	0.001 MA
Ownership	Public

5.20. FUSION POWER CORE (COMPACT FUSION SYSTEMS, UNITED STATES OF AMERICA)

5.20.1. Introduction

Fusion Power Core is a device being planned by US private company Compact Fusion Systems [61].

5.20.2. Purpose

The main purpose of Fusion Power Core is to research and develop magnetized target fusion.

5.20.3. Main features

Fusion Power Core is being designed as a magnetized target fusion device. General information is listed in Table 111 below.

TABLE 111. GENERAL INFORMATION

Device type	Magnetized Target Fusion
Status	Planned
Ownership	Private

5.21. IDCD (CTFUSION, UNITED STATES OF AMERICA)

5.21.1. Introduction

IDCD is a device developed and operated by US private company CTFusion [61].

5.21.2. Purpose

The purpose of IDCD purpose is to demonstrate reactor-relevant plasma operation.

5.21.3. Main features

IDCD is based on an approach called dynomak, in which plasmas are formed and maintained by non-axisymmetric, fully inductive magnetic helicity injectors. General information is listed in Table 112 below.

TABLE 112. GENERAL INFORMATION

Device type	Spheromak
Status	Operating
Ownership	Private

5.22. HELICITY DRIVE (HELICITYSPACE, UNITED STATES OF AMERICA)

5.22.1. Introduction

Helicity Drive is a device being planned by US private company Helicity Space [61].

5.22.2. Purpose

The main purpose of Helicity Drive is to make space missions faster and more efficient based on fusion-driven power and propulsion systems.

5.22.3. Main features

Helicity Drive is being designed as a device for both space propulsion and power. Its systems are based on plasma magnetic reconnection and magnetic compression with passive coils. General information is listed in Table 113 below.

TABLE 113. GENERAL INFORMATION

Device type	Space Propulsor
Status	Planned
Ownership	Private

5.23. POLARIS (HELION ENERGY, UNITED STATES OF AMERICA)

5.23.1. Introduction

Polaris is a device being planned by US company Helion Energy. Its construction is expected to start in 2023 [61]. Helion Energy is designing a pulsed fusion energy generator based on self-organized, high beta plasmas in a field reversed configuration. The high beta of the plasma would enable direct conversion of the fusion energy held in the charged particles in the plasma into electricity, which is efficient as compared to thermal cycle conversion of neutron energy. Such a design motivates the use of alternate fuels like D-^{3}He (see Table 1, p. 4), where most of the fusion energy is released in charged particles as opposed to the D-T case, in which most of the fusion energy is released in neutrons.

5.23.2. Purpose

The purpose of Polaris is to achieve ^{3}He production via D-D fusion reactions (see Table 1, p. 4).

5.23.3. Main features

Polaris is being designed as a field reversed configuration device. General information is listed in Table 114 below.

TABLE 114. GENERAL INFORMATION

Device type	Field Reversed Configuration
Status	Planned
Ownership	Private

5.24. TRENTA (HELION ENERGY, UNITED STATES OF AMERICA)

5.24.1. Introduction

Trenta is a device developed and operated by US private company Helion Energy. Its construction was completed in 2020 [61].

5.24.2. Purpose

The purpose of Trenta is to support Helion Energy's research and development programme, with the aim of demonstrating fusion conditions and achieving ^{3}He production via D-D fusion reactions (see Table 1, p. 4).

5.24.3. Main features

Trenta is a field reversed configuration device. Technical information is listed in Table 115 below.

TABLE 115. TECHNICAL INFORMATION

Device type	Field Reversed Configuration
Status	Operating
Magnetic field, B	>10 T
Plasma lifetime	0.001 s
Ownership	Private

5.25. HORNE HYBRID REACTOR (HORNE TECHNOLOGIES LLC, UNITED STATES OF AMERICA)

5.25.1. Introduction

Horne Hybrid Reactor is a device designed and being constructed by US private company Horne Technologies LLC [61].

5.25.2. Purpose

The main purpose of Horne Hybrid Reactor is to optimize plasma confinement and energy balance in this class of inertial electrostatic fusion devices.

5.25.3. Main features

Horne Hybrid Reactor is an inertial electrostatic fusion device under construction. It will use REBCO superconducting magnets for producing a high-beta magnetic configuration. The device will rely on inertial electrostatic confinement for plasma heating. General information is listed in Table 116 below.

TABLE 116. GENERAL INFORMATION

Device type	Inertial Electrostatic Fusion
Status	Under construction
Ownership	Private

5.26. PJMIF (HYPERJET FUSION CORPORATION, UNITED STATES OF AMERICA)

5.26.1. Introduction

PJMIF is a device being planned by US private company HyperJet Fusion Corporation, whose focus is designing and building plasma guns [61].

5.26.2. Purpose

The purpose of PJMIF is to study the feasibility of driving pulsed fusion energy reaction via imploding plasma liners.

5.26.3. Main features

PJMIF is being designed as a magnetized target fusion device in which plasma is formed at the centre of a spherical vacuum vessel. Fusion combustion is achieved by compressing the plasma to a diameter of 1 cm, thanks to the ejection of plasma jets with hypersonic velocities from the periphery of the vacuum vessel. The jets merge forming a plasma liner, which continues to narrow towards the centre. The device will feature a liquid wall to absorb the fusion neutrons from D-T reactions (see Table 1, p. 4), breeding tritium and eventually serving as a coolant in a heat exchanger. General information is listed in Table 117 below.

TABLE 117. GENERAL INFORMATION

Device type	Magnetized Target Fusion
Status	Planned
Ownership	Private

5.27. FOCUS FUSION (LAWRENCEVILLE PLASMA PHYSICS, INC. DBA LPPFUSION, UNITED STATES OF AMERICA)

5.27.1. Introduction

Focus Fusion is a device located at Lawrenceville Plasma Physics, Inc., USA [61].

5.27.2. Purpose

The purpose of Focus Fusion is to pave the way for net fusion energy from p-^{11}B reactions (see Table 1, p. 4).

5.27.3. Main features

Focus Fusion is a dense plasma focus designed to work with p-^{11}B fuel. It consists of two cylindrical metal electrodes, nested inside each other, and placed inside the vacuum chamber. General information is listed in Table 118 below.

TABLE 118. GENERAL INFORMATION

Device type	Dense Plasma Focus
Status	Operating
Ownership	Private

5.28. CFR (LOCKHEED MARTIN, UNITED STATES OF AMERICA)

5.28.1. Introduction

CFR is a device designed and operated by Lockheed Martin. The project started in 2010.

5.28.2. Purpose

The purpose of CFR is to demonstrate high beta plasma operation in a compact magnetic mirror machine.

5.28.3. Main features

CFR is a compact magnetic mirror machine able to achieve high beta plasma by combining cusp confinement and magnetic mirrors. CFR features superconducting magnets. General information is listed in Table 119 below.

TABLE 119. GENERAL INFORMATION

Device type	Magnetic Mirror Machine
Status	Operating
Ownership	Public

5.29. MIFTI (MAGNETO-INERTIAL FUSION TECHNOLOGIES, INC., UNITED STATES OF AMERICA)

5.29.1. Introduction

MIFTI is a device being planned by US private company Magneto-Inertial Fusion Technologies, Inc [61].

5.29.2. Purpose

The purpose of MIFTI is to research and develop the pinch line for fusion energy production.

5.29.3. Main features

MIFTI is being designed as a pinch device. General information is listed in Table 120 below.

TABLE 120. GENERAL INFORMATION

Device type	Pinch
Status	Planned
Ownership	Private

5.30. PFRC (PRINCETON FUSION SYSTEMS, UNITED STATES OF AMERICA)

5.30.1. Introduction

PFRC is a device being planned by US private company Princeton Satellite Systems [61].

5.30.2. Purpose

The purpose of PFRC is to demonstrate the feasibility of the field reversed configuration as a portable and modular fusion power plant.

5.30.3. Main features

PFRC is a fusion microreactor concept based on the field reversed configuration. It is expected to be portable (sized to fit on a truck) and produce power ranging 1–10 MW. The concept is expected to be applicable for space applications as a thrust-controlled rocket, or as direct fusion drive. General information is listed in Table 121 below.

TABLE 121. GENERAL INFORMATION

Device type	Field Reversed Configuration
Status	Planned
Ownership	Private

5.31. Z MACHINE (SANDIA NATIONAL LABORATORIES, UNITED STATES OF AMERICA)

5.31.1. Introduction

Z machine is located at Sandia National Laboratories, USA. It is a part of a research program launched in 1960. Z machine is the most powerful laboratory radiation source in the world.

5.31.2. Purpose

The purpose of Z machine is to support the development of magnetized liner inertial fusion.

5.31.3. Main features

Z machine is a pinch fusion device. It uses capacitors to activate powerful electrical pulses. These hit a small target made of hundreds tungsten wires placed in a hohlraum (a small metal container) located in the centre of the machine. The deposited energy creates a strong magnetic field that pushes the exploded particles inside to collide. The hohlraum walls get heated up to 1.8 million degrees Celsius by the radiation resulting from collisional processes [86]. General information is listed in Table 122 below.

TABLE 122. GENERAL INFORMATION

Device type	Pinch
Status	Operating
Ownership	Public

5.32. NORMAN (C-2W) (TAE TECHNOLOGIES, UNITED STATES OF AMERICA)

5.32.1. Introduction

Operating since 2017, Norman is the fifth device developed and operated by US private company TAE Technologies [61].

5.32.2. Purpose

The purpose of Norman is to show plasma ramp-up by neutral beam injection and current drive in a reversed field configuration, as well as to improve edge and divertor plasma performance, obtaining plasma temperature up to 3 keV.

5.32.3. Main features

Norman is a field reversed configuration device. Plasma heating and current drive is achieved via neutral beam injection. Norman is the largest theta-pinch collisional-merging system in the world. It consists of a central confinement section surrounded by 2 internal divertors; 2 sections for field reversed theta-pinch formation; and 2 outer divertors. Liquid nitrogen is used for the cooling system, improving pumping performance inside the divertors [87]. General information is listed in Table 123 below.

TABLE 123. GENERAL INFORMATION

Device type	Field Reversed Configuration
Status	Operating
Ownership	Private

5.33. COPERNICUS (TAE TECHNOLOGIES, UNITED STATES OF AMERICA)

5.33.1. Introduction

Copernicus is the next step device being constructed by US private company TAE Technologies [61].

5.33.2. Purpose

The purpose of Copernicus is to demonstrate net scientific energy gain[2] in a reversed field configuration by 2025, simulating the D-T fuel cycle performance while using only hydrogen fuel.

5.33.3. Main features

Copernicus will be a field reversed configuration device operating with hydrogen plasma. It is expected to become operational by 2024. General information is listed in Table 124 below.

TABLE 124. GENERAL INFORMATION

Device type	Field Reversed Configuration
Status	Under construction
Ownership	Private

5.34. ZEBRA (UNIVERSITY OF NEVADA, UNITED STATES OF AMERICA)

5.34.1. Introduction

Operating since 2000, Zebra is located at University of Nevada, USA.

5.34.2. Purpose

The purpose of Zebra is to carry out research as well as contribute to training students in the field of high energy density science.

5.34.3. Main features

Zebra is a pulsed power generator [88]. General information is listed in Table 125 below.

TABLE 125. GENERAL INFORMATION

Device type	Pinch
Status	Operating
Ownership	Public

5.35. MST (UNIVERSITY OF WISCONSIN-MADISON, UNITED STATES OF AMERICA)

5.35.1. Introduction

MST is located at University of Wisconsin-Madison, USA.

5.35.2. Purpose

The main purpose of MST is to advance plasma physics research.

5.35.3. Main features

MST is a reversed field pinch, which can also operate with tokamak geometries. MST features an aluminium shell that, depending on the chosen configuration, can serve either as vacuum vessel or as equilibrium magnet [89]. General information is listed in Table 126 below.

TABLE 126. GENERAL INFORMATION

Device type	Reversed Field Pinch
Status	Operating
Ownership	Private

5.36. FUZE-Q (ZAP ENERGY INC., UNITED STATES OF AMERICA)

5.36.1. Introduction

Operating since 2018, FuZE-Q is a device designed and operated by US private company Zap Energy Inc. [61].

5.36.2. Purpose

The purpose of FuZE-Q is to reach scientific energy gain[2] Q_{sci}=1.

5.36.3. Main features

FuZE-Q is a pinch. An electric current generates sheared flows. These produce a magnetic field that confines and compresses the plasma. The Z-pinch plasma is heated and compressed by flowing an extremely large current (~10^6 amps) through the plasma. General information is listed in Table 127 below.

TABLE 127. GENERAL INFORMATION

Device type	Pinch
Status	Operating
Ownership	Private

6. DEMO DEVICES

6.1. CFETR (CHINESE CONSORTIUM, CHINA)

6.1.1. Introduction

CFETR is a DEMO concept based on conventional tokamak design being developed in China by a Chinese Consortium. CFETR is the next device in the roadmap for the realization of fusion energy in China. The conceptual design was completed in 2015. Construction of the CFETR is expected to be completed by 2040.

6.1.2. Purpose

CFETR is expected to bridge the gaps between ITER and a fusion power plant, as well as to demonstrate net engineering gain[2] (Q_{eng}>1).

6.1.3. Main features

CFETR R&D plan is expected to consist of two phases. During the first phase, the efforts will focus on achieving steady-state operation and tritium self-sufficiency with fusion power up to 200 MW. This phase will address issues relevant to burning plasma physics R&D in order to demonstrate steady-state advanced operation. The second phase will focus on validating DEMO-relevant issues with fusion power above 1 GW [90]. Some of CFETR conceptual features are: major radius of 7.2 m, minor radius of 2.2 m, plasma elongation κ_{95}=2, plasma current of 14 MA, magnetic field on axis of 6.5 T, normalized beta β_N=2.3, a predicted scientific energy gain[2] Q_{sci}=30. General information is listed in Table 128 below.

TABLE 128. GENERAL INFORMATION

Device type	Conventional Tokamak
Status	Planned
Ownership	Public

6.2. EU-DEMO (EUROFUSION, EUROPEAN UNION)

6.2.1. Introduction

EU-DEMO is a DEMO concept based on conventional tokamak design being developed in Europe by EUROfusion. The European DEMO or EU-DEMO is the facility expected to follow ITER in the European roadmap to electricity from fusion. The project is currently in its conceptual design phase and is expected to start operating by 2050.

6.2.2. Purpose

EU-DEMO aims to demonstrate the technological and economic viability of fusion by producing about 500 MW of net electricity and to achieve tritium self-sufficiency.

6.2.3. Main features

Several design options are being studied. These options will have an impact on a number of plant technologies, including the divertor configuration and breeding blanket solution, among others. The pre-conceptual design of EU-DEMO tokamak foresees a major radius of ~9 m and a fusion power of ~2000 MW [91]. General information is listed in Table 129 below.

TABLE 129. GENERAL INFORMATION

Device type	Conventional Tokamak
Status	Planned
Ownership	Public

6.3. JA-DEMO (JAPANESE CONSORTIUM, JAPAN)

6.3.1. Introduction

The Japanese DEMO or JA-DEMO is a DEMO concept based on conventional tokamak design being developed in Japan by a Japanese Consortium. Construction of JA-DEMO is expected to be completed by 2050.

6.3.2. Purpose

The Purpose of JA-DEMO are to demonstrate net engineering gain[2] (Q_{eng}>1) and tritium self-sufficiency, as well as plant availability bridging the gap to commercialization of fusion energy.

6.3.3. Main features

It is expected that for reliable electric power generation, a fusion output of 1.5 GW or higher, will be required. For the magnets system of JA-DEMO, superconducting (Nb3Sn) magnets consisting of a central solenoid, 7 poloidal field coils (NbTi), 16 toroidal field coils (Nb3Sn or Nb3Al) are being considered. Some of JA-DEMO conceptual features are: major radius of 8.5 m, minor radius of 2.42 m, plasma elongation κ_{95}=1.65, plasma current of 12.3 MA, magnetic field on axis of 5.94 T, normalized beta β_N=3.4, current drive power of 83.7 MW and a predicted scientific energy gain[2] Q_{sci}=17.5 [92]. General information is listed in Table 130 below.

TABLE 130. GENERAL INFORMATION

Device type	Conventional Tokamak
Status	Planned
Ownership	Public

6.4. K-DEMO (KOREA INSTITUTE OF FUSION ENERGY, REPUBLIC OF KOREA)

6.4.1. Introduction

K-DEMO is a DEMO concept based on conventional tokamak design being developed in the Republic of Korea by the Korean Institute of Fusion Energy. Construction of K-DEMO is expected to be fully completed by 2050. In a first early phase of the project (2037–2050), K-DEMO will be used to develop and test components. In its second phase, after 2050, it is expected to demonstrate net electrical power.

6.4.2. Purpose

The Purpose of K-DEMO are to demonstrate physics and technology necessary for achieving net engineering gain[2] (Q_{eng}>1).

6.4.3. Main features

The conceptual design of K-DEMO features a major radius of 6.8 m, minor radius of 2.1 m, toroidal field of about 7 T, and plasma current larger than 12 MA. K-DEMO is expected to feature a double-null divertor configuration and the divertor X-point inside the vacuum vessel. The K-DEMO blanket sectors are subdivided into 16 inboard and 32 outboard sectors. The upper or lower divertor is also subdivided into 32 modules. The key features of the K-DEMO magnet system include two toroidal field coil winding packs with different conductors, enclosed in the toroidal field case [93]. General information is listed in Table 131 below.

TABLE 131. GENERAL INFORMATION

Device type	Conventional Tokamak
Status	Planned
Ownership	Public

6.5. DEMO-RF (RUSSIAN CONSORTIUM, RUSSIAN FEDERATION)

6.5.1. Introduction

DEMO-RF is a DEMO concept based on conventional tokamak design being developed in the Russian Federation by a Russian Consortium. Construction of DEMO-RF is expected to be completed by 2055.

6.5.2. Purpose

DEMO-RF is expected to demonstrate net engineering gain[2] (Q_{eng}>1).

6.5.3. Main features

DEMO-RF features are under development. The conceptual design currently foresees the use of the facility either as a pure fusion energy system or as a fusion–fission hybrid facility with high temperature superconducting magnets and a total magnetic field larger than 8 T and plasma current of about 5 MA. Liquid metal plasma facing components are being considered for first wall and divertor [94]. General information is listed in Table 132 below.

TABLE 132. GENERAL INFORMATION

Device type	Conventional Tokamak
Status	Planned
Ownership	Public

6.6. FDP (GENERAL FUSION INC, UNITED KINGDOM)

6.6.1. Introduction

FDP is a DEMO concept based on magnetized target fusion design (which uses pneumatic pistons to compress the plasma) being developed by Canadian company General Fusion (supported by the government of Canada) [61]. Its target completion date is 2025.

6.6.2. Purpose

The purpose of the FDP is to prove that magnetized target fusion technology can scale to a commercial pilot plant.

6.6.3. Main features

The three main components of FDP are a liquid metal chamber, compression system and plasma injector. The device is expected to operate by heating hydrogen atoms at high temperatures and compressing it with pneumatic pistons surrounding a rotating internal chamber filled with liquid metal. The generated heat from fusion reactions can then be transferred into the liquid metal, and in future commercial power plants, it can be extracted from the metal and used to produce steam, which will drive a turbine producing electricity. General information is listed in Table 133 below.

TABLE 133. GENERAL INFORMATION

Device type	Magnetized Target Fusion
Status	Planned
Ownership	Public-Private

6.7. ST-E1 (TOKAMAK ENERGY, UNITED KINGDOM)

6.7.1. Introduction

ST-E1 is a DEMO concept based on spherical tokamak design being developed by UK company Tokamak Energy Ltd [61]. ST-E1 is expected to be completed by 2030.

6.7.2. Purpose

The purpose of ST-E1 is to demonstrate net engineering gain[2] (Q_{eng}>1).

6.7.3. Main features

ST-E1 will be a compact spherical tokamak with high temperature superconducting magnets. General information is listed in Table 134 below.

TABLE 134. GENERAL INFORMATION

Device type	Spherical Tokamak
Status	Planned
Ownership	Private

6.8. STEP (UKAEA, UNITED KINGDOM)

6.8.1. Introduction

STEP is a DEMO concept based on spherical tokamak design being developed by UKAEA. The first phase of the programme is to produce a concept design by 2024. Its target completion date is 2040. STEP is expected to be smaller than ITER. The spherical shape can improve efficiency in the magnetic field and potentially minimise the plant's costs.

6.8.2. Purpose

The STEP programme aims to demonstrate the commercial viability of fusion, as well as to enable the flourishing of a fusion industry.

6.8.3. Main features

STEP will be a compact spherical tokamak able to produce net engineering gain[2] (Q_{eng}>1), although it is not expected to be a commercially operating plant [95]. General information is listed in Table 135 below.

TABLE 135. GENERAL INFORMATION

Device type	Spherical Tokamak
Status	Planned
Ownership	Public

6.9. ARC (COMMONWEALTH FUSION SYSTEMS, UNITED STATES OF AMERICA)

6.9.1. Introduction

ARC is a DEMO concept based on conventional tokamak design being developed by US private company Commonwealth Fusion Systems [61]. ARC will be the successor of SPARC (see p. 77).

6.9.2. Purpose

The purpose of ARC is to demonstrate the commercial viability of fusion with high temperature superconducting magnet technology.

6.9.3. Main features

ARC will be a compact conventional tokamak with high temperature superconducting magnets able to produce ~ 200–250 MWe, with a radius of 3.3 m, a minor radius of 1.1 m, and an on-axis magnetic field of 9.2 T. ARC will feature REBCO superconducting toroidal field coils. The coils will have joints to enable disassembly, which will allow for quick replacements of the vacuum vessel (thus mitigating first wall life-time issues) and enable the possibility of testing various vacuum vessel designs and divertor materials [96]. General information is listed in Table 138 below.

TABLE 138. GENERAL INFORMATION

Device type	Conventional Tokamak
Status	Planned
Ownership	Private

6.10. GA-FPP (GENERAL ATOMICS, UNITED STATES OF AMERICA)

6.10.1. Introduction

The GA-FPP is a DEMO concept based on steady-state, compact advanced tokamak design, which was announced in October 2022 [97].

6.10.2. Purpose

The purpose of the GA-FPP is to demonstrate the commercial viability of fusion, achieving steady-state operation, maximizing efficiency, reducing maintenance costs and increasing the lifetime of the facility.

6.10.3. Main features

The GA-FPP design approach will rely on advanced sensors, control algorithms and high performance computers for controlling the plasma, silicon-carbide breeding blankets for producing the necessary Tritium and microwave heating for powering the fusion reactions.

TABLE 139. GENERAL INFORMATION

Device type	Conventional Tokamak
Status	Planned
Ownership	Private

6.11. DA VINCI (TAE TECHNOLOGIES, UNITED STATES OF AMERICA)

6.11.1. Introduction

Da Vinci is a DEMO concept based on field reversed configuration design being developed by US private company TAE Technologies [61]. Da Vinci will be the successor of Copernicus (see p. 139).

6.11.2. Purpose

The purpose of Da Vinci is to demonstrate the commercial viability of fusion in a reversed field configuration via p-^{11}B reactions (see Table 1, p. 4).

6.11.3. Main features

Da Vinci will be a field reversed configuration device. General information is listed in Table 140 below.

TABLE 140. GENERAL INFORMATION

Device type	Field Reversed Configuration
Status	Planned
Ownership	Private

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