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INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

Enhancing nuclear safety

Nuclear Fusion Reactors

//safety and radiation protection considerations for demonstration reactors that follow the ITER facility

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L'IRSN
//in brief

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It is the nation's public service expert in nuclear and radiation risks, and its activities cover all the scientific and technical issues related to these risks. IRSN interacts with all parties concerned by these risks to contribute to public policy issues relating to nuclear safety, human and environmental protection against ionizing radiation, and the protection of nuclear materials, facilities, and transport against the risk of malicious acts. Its work also actively contributes to other major public policies in research, innovation, occupational health and environmental health.

Reference documents

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Foreword

The Institute for Radiological Protection and Nuclear Safety is a public service expert in nuclear and radiation risks. It develops research programmes and conducts studies in its fields of scientific and technical expertise, and provides technical support to the public authorities in charge of nuclear safety and security, and radiation protection. In fulfilling its missions in risk assessment and prevention, whether at its own initiative or to support safety, security and radiation protection authorities, the Institute is called upon to state its position on certain scientific and technical issues.

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Reference documents

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Preface

The goal of this document, which falls under the field of forward planning, is to explain the safety and radiation protection issues which need to be examined while designing future nuclear fusion reactors after ITER. Such reactors constitute the preparatory stage before building industrial nuclear fusion power generation facilities.

These reactors will follow on the heels of ITER (International Thermonuclear Experimental Reactor), which is currently being built at Cadarache (France) to demonstrate the technical and scientific feasibility of controlled fusion. They are currently being studied in various countries across the world (China, South Korea, India, etc.).

ITER is the first magnetic confinement fusion reactor to require a construction licence decree under French nuclear facilities regulations. The decree was passed in 2012 after the Institute for Radiological Protection and Nuclear Safety conducted an in-depth assessment of the safety and radiation protection measures adopted by the operator.

Due to its design and operation, ITER presents unique safety and radiation protection issues, including the risk of dust and hydrogen isotope explosion, plasma malfunction, magnetic system failures, etc.

The Institute has put in place the following measures to manage these specific features:

- collaboration with the Canadian Nuclear Safety Commission on tritium confinement and the effects of tritium releases into the environment;
- technical support from a plasma physics expert;
- neutronic and structure activation calculations by the European Organization for Nuclear Research in Geneva (Switzerland) and the implementation and research and development programmes within the Institute and/or in collaboration on specific safety problems raised by a nuclear fusion facility.

In performing the safety analysis for ITER, IRSN has acquired specific expertise in this field. The Institute drew from this experience and acquired knowledge and considered that it was wise to explore the safety and radiation protection issues that demonstration reactors currently under development could raise.

Future demonstration reactors will mainly differ from the ITER reactor by seeking to attain tritium self-sufficiency and achieve significantly longer operating times. These differences will have a significant impact on design and a direct influence on safety.

The thoughts expressed in this document highlight the growing importance of safety and radiation protection issues in these new designs with respect to the ITER facility.

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1/ Introduction

Current power reactors use fission of heavy nuclei, mainly uranium, to produce energy. However, fusion projects are based on fusion of the light nuclei of hydrogen isotopes, deuterium and tritium.

Since initial research on nuclear fusion began in the 1950s, the system most frequently used to produce it is the "tokamak", comprising a toroidal chamber in which a gaseous mix of hydrogen isotopes in the form of plasma is confined using the helical magnetic field resulting from the combination of magnetic fields produced by the field coils. Other projects aim to produce a helical magnetic field by giving the toroidal chamber and coils a helical form, as in the W7X stellerator developed in Germany. Another approach has been developed, which consists of using lasers to exert high pressure on a target of hydrogen isotopes, mainly in the context of nuclear weapons development (the CEA's[1] Megajoule Laser facility [LMJ], the National Ignition Facility [NIF] developed by Lawrence Livermore National Laboratory, etc.). This report only covers the most common approach, magnetic confinement in a tokamak.

Since the 1950s, over 200 tokamaks have been developed around the world to perform research on fusion. In the late 1980s, a decision was made by the European Atomic Energy Community (Euratom), Japan, the Soviet Union and the United States to construct ITER (International Thermonuclear Experimental Reactor) with a view to "the scientific and technical feasibility of fusion power". This decision is part of a long-term initiative which then planned construction of a second research reactor, called DEMO (DEMOnstration power plant), more similar to a power reactor, prior to industrialisation of nuclear fusion.

Fusion is the combination of two nuclei of light atoms, here deuterium and tritium, to form a heavier nucleus which releases a large amount of energy carried by the reaction products (nuclei, particles and radiation).

The most commonly used system for producing a fusion reaction is the tokamak, a Russian acronym meaning "toroidal chamber with magnetic coils", which uses magnetic fields to create, confine and control a hot plasma inside which the fusion reaction can occur.

(1) Commissariat à l'énergie atomique et aux énergies alternatives.

ITER is the first fusion facility using magnetic confinement requiring authorisation in accordance with the regulations applicable to nuclear installations.

Due to the quantities of radioactive substances used, far greater than those used in the tokamaks previously built, the ITER facility is the first fusion facility using magnetic confinement to require a construction license under French law governing nuclear facilities. Consequently, its construction license was issued *via* Decree no. 2012-1248 of 9 November 2012, following a detailed examination of the provisions adopted by the operator to prevent or adequately mitigate the risks and disadvantages it presents. Assessment of these provisions was performed by the French Institute for Radiological Protection and Nuclear Safety (IRSN[2]). A summary of this assessment is given in Appendix 1 of this document. The ITER facility is currently under construction. A timeline of the main past and scheduled milestones is given in Appendix 2 of this document.

(2)
Institut de radioprotection et de sûreté nucléaire.

Various reactor projects, called "DEMO reactors", using the tokamak concept are under development worldwide. IRSN is using the experience acquired during the ITER facility safety assessment to identify the safety issues they need to take into account from the design phase.

Currently, no new international projects are planned following the ITER facility. Different nuclear fusion reactor projects are currently being studied in various countries around the world (People's Republic of China, Republic of Korea, Republic of India, etc.). Most of these projects are based on the same "tokamak" concept. For simplicity, in the rest of this document, all these "tokamak" projects will be called "DEMO reactors". These facilities involve more significant safety and radiation protection issues than for the ITER facility.

The purpose of this document is to present some considerations on the safety and radiation protection issues that should be examined from the design stage of "DEMO reactors", based on publications currently available and experience acquired during the ITER facility safety assessment.

2/ The “tokamak” concept

In a nuclear fusion facility using the “tokamak” magnetic confinement concept, fusion reactions take place inside a toroidal plasma, a state of matter made up of ions and electrons. Magnetic confinement of the plasma (Figure 1) is mainly produced by a toroidal magnetic field (B_t) produced by toroidal field coils and by a poloidal magnetic field (B_p) produced by an electric current in the plasma. Helical magnetic field lines are thus produced around which ions and electrons spiral, thereby containing the plasma within a sealed toroidal vessel, called the “vacuum vessel”, into which deuterium and tritium have been introduced.

Plasma is a hot, low-density gas produced under the action of a strong magnetic field. It is made up of positive ions and electrons, which have been stripped off due to the temperature.

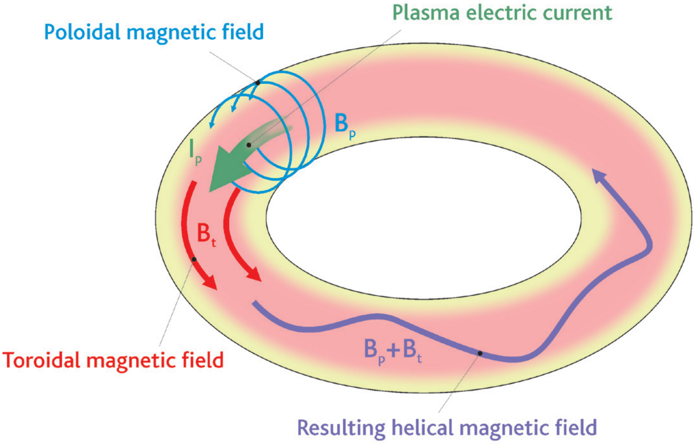

Figure 1. Principle of magnetic confinement of a plasma in a tokamak. © Georges Goué/IRSN.

The induced current, called the "plasma current", heats the plasma but is not in itself enough to reach the temperature conditions needed for the fusion of deuterium and tritium. Additional heating is therefore needed, which will be described later.

The fusion reaction of deuterium ($^2_1H^+$) and tritium ($^3_1H^+$), which produces α particles ($^4_2He^{2+}$) and neutrons (n) is as follows:

$$^2_1H^+ + {}^3_1H^+ \rightarrow {}^4_2He^{2+} (3.6\ \mathrm{MeV}) + n\ (14\ \mathrm{MeV})$$

Tritium is the radioactive isotope of hydrogen. Its nucleus decays *via* a beta emission with an average energy of 5.7 keV to produce stable helium-3. Its half-life is 12.3 years.

High energy neutrons are produced during fusion in the plasma. 80% of this energy is converted into heat.

The fusion reaction produces high-energy neutrons (14 MeV) whereas the energy of neutrons produced by fission reactions is less than 2 MeV. A fifth of the energy is released into the plasma *via* the α particles, while four-fifths is converted into heat *via* neutron interactions with the materials comprising vacuum vessel internal components and the vessel itself.

A "tokamak" (Figure 2) is made up of the following main components:

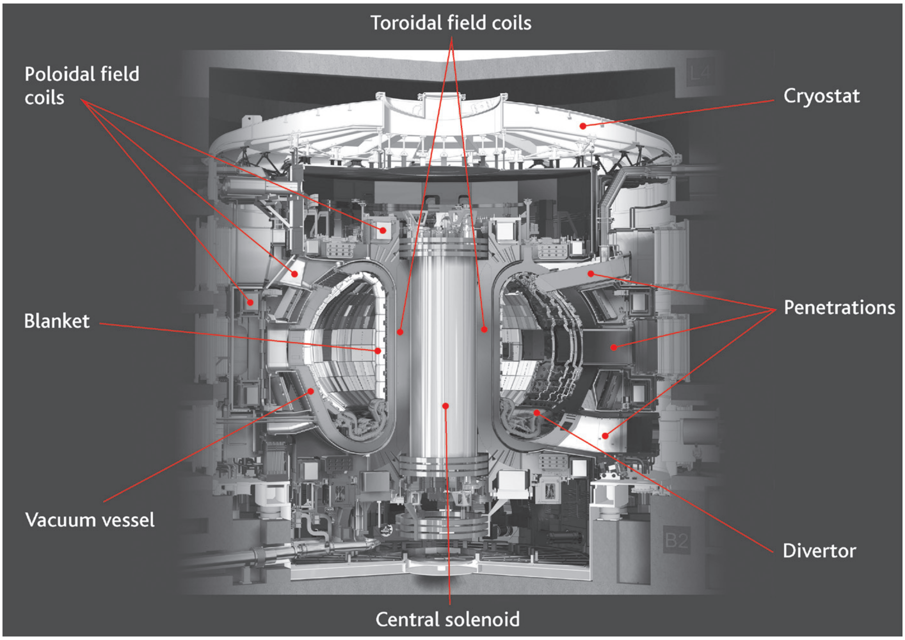

Figure 2. Schematic diagram of the ITER facility tokamak. © ITER Organization.

- **the vacuum vessel and its internal components (blanket and divertor);**

  The vacuum vessel is a sealed toroidal metal chamber which encloses two main components: the blanket and the divertor (Figure 3).

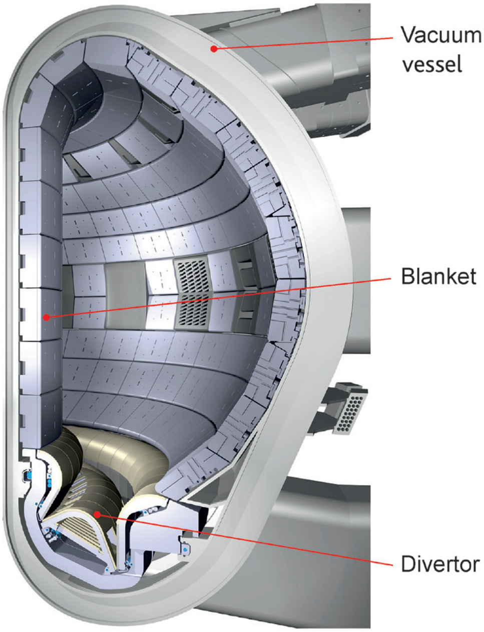

Figure 3. ITER facility vacuum vessel and internal components. © ITER Organization.

The blanket provides neutron protection for the metal walls of the vacuum vessel. It is cooled *via* a coolant to transfer the radiative heat of the plasma and a large fraction of the heat due to the slowing of neutrons in the structure to cooling towers (for the ITER facility) or to a turbine generator (for “DEMO reactors”). It also removes the heat produced by nuclear reactions in its materials caused by neutrons from the fusion reactions in the plasma (“neutron reactions”). Furthermore, very sparsely for the ITER facility and comprehensively for the “DEMO reactors”, the blanket is tritium breeding, i.e. it contains lithium (Li) which, under the neutron flux from the plasma, produces tritium *via* the following reactions:

$$^{7}_{3}\mathrm{Li} + \mathrm{n} \rightarrow {}^{4}_{2}\mathrm{He} + {}^{3}_{1}\mathrm{H} + \mathrm{n} - 2.46\ \mathrm{MeV}$$

$$^{6}_{3}\mathrm{Li} + \mathrm{n} \rightarrow {}^{4}_{2}\mathrm{He} + {}^{3}_{1}\mathrm{H} + 4.8\ \mathrm{MeV}$$

Most of the heat removed from ITER, which will ultimately be converted into electricity in a DEMO reactor, comes from the slowing of neutrons in the structures of the vacuum vessel and its internal components.

(3) Tungsten is a highly refractory metal which offers very high resistance to heat and wear.

The wall of the blanket that faces the plasma, called the first wall, is subject to heavy loads due to the neutron flux, thermal flux and thermal shocks, which erode it and create dust (see Chapter 3). Furthermore, this first wall adsorbs tritium. It is made of beryllium or tungsten(3), and must be changed regularly during operation. To do this, various sized parts of the blanket must be removed from the vacuum vessel using robots, then transported, also by robotic means, into hot cells in which the first wall is changed.

Tritium breeding blankets promote tritium production while also providing neutron protection for the metal walls of the vacuum vessel.

The tritium breeding blankets (see Section 6.1) are essential equipment for future nuclear fusion facilities. They are the subject of intense research and development activities worldwide. Several types of blanket are being studied (some examples are given in Figure 4). They always include the following:

– lithium which, under neutron flux from the plasma, reacts to form tritium. The lithium may be present under various forms (liquid lithium, lithium-lead eutectic, lithium-based oxide ceramics, molten lithium salts, etc.);

– a neutron booster material, such as beryllium or lead, which can increase tritium production. Interactions between neutrons and these materials cause nuclear reactions, called (n, 2n) reactions, which lead to the emission of twice as many neutrons;

– a coolant (water, helium, liquid metal, etc.) to transfer part of the radiative heat of the plasma, heat due to the slowing of neutrons in the structure and heat from "neutron reactions" to cooling towers (for the ITER facility) or to a turbine generator (for "DEMO reactors").

The divertor, located in the lower part of the vacuum vessel, extracts helium, fuel that has not undergone fusion (and can be reused) and impurities.

The divertor, located in the lower part of the vacuum vessel, extracts helium-4 (α particles from the fusion reactions that have captured plasma electrons), fuel (tritium and deuterium) that has not undergone fusion and impurities (in particular the dust from erosion of the first wall of the blanket and the divertor). As for the blanket, the divertor is cooled by a coolant to transfer part of the radiative heat of the plasma, heat due to the slowing of neutrons in the structure and heat from "neutron reactions" to cooling towers (for the ITER facility) or to a turbine generator (for "DEMO reactors"). The divertor also has a first wall facing the plasma, whose material (tungsten) may different from that of the first wall of the blanket, due to the greater thermal flux from the plasma. This first wall must also be changed during operation, in the same way as for the blanket.

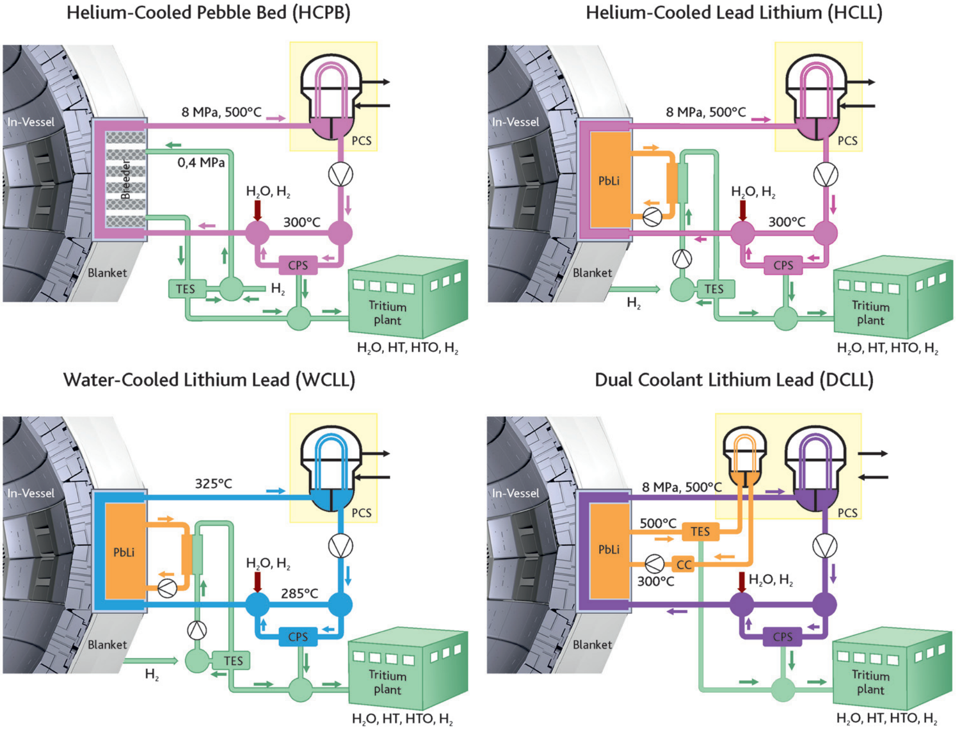

Figure 4. Various types of tritium breeding blankets, from [1]. TES: tritium extraction system; CPS: coolant purification system; PCS: power conversion system). © Georges Goué/IRSN.

The vacuum vessel is a metal chamber. Cooling water flows between the two walls to transfer the heat due to the slowing of neutrons in the walls and in the borated stainless steel blocks installed between these walls, along with heat from "neutron reactions", to cooling towers (for the ITER facility) or to a turbine generator (for "DEMO reactors"). Penetrations are used to install equipment for monitoring plasma behaviour and performance, for additional heating, for the fuelling system and for cooling and vacuum systems (see Figure 3). Penetrations are also planned for extracting parts of the blanket or divertor for maintenance purposes. These penetrations can have large cross-sections when the parts of the blanket or divertor to be extracted are very large, which has a significant impact on the overall design of the tokamak.

The vacuum vessel has a pressure-suppression system to cope with all foreseeable accident situations involving a rise in internal pressure (water ingress, air ingress, etc.). This system is made up of a relief line fitted with valves and rupture discs, leading to a relief tank half filled with water, which uses sparging to condense most of the tritiated water (HTO) formed *via* reaction with the tritium in the event of a cooling water leak into the vacuum vessel. The non-condensed fraction of the tritiated water would mostly be trapped by the ventilation and detritiation system which keeps the atmosphere of the relief tank under negative pressure. The fraction that is not trapped would be released into the environment *via* a facility stack (Figure 5). In the event of air ingress into the vacuum vessel, the non-condensable tritiated gases (HT) would mostly be trapped by the ventilation and detritiation system alone. Other radioactive materials present in the vacuum vessel and cooling systems (activated dust and activated corrosion products) would mostly be trapped by the ventilation and detritiation system filters.

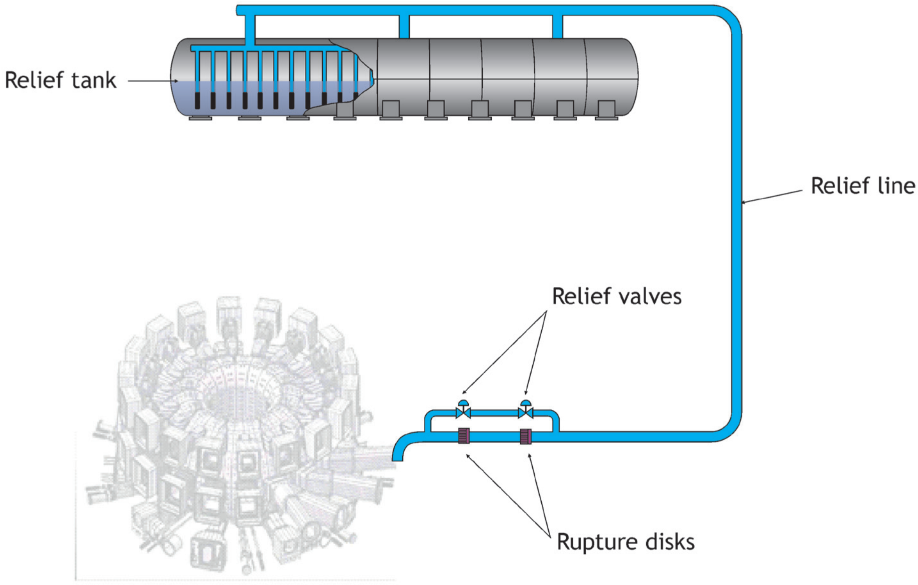

Figure 5. ITER facility vacuum vessel pressure suppression system (VVPSS). © Georges Goué/IRSN.

- **the cryopumps;**

  The cryopumps (Figure 6), located under the vacuum vessel downstream of the divertor, remove gases and dust from the vacuum vessel (mainly tritium and deuterium that have not reacted, helium-4 from fusion reactions and dust from erosion of the first wall of the blanket and the first wall of the divertor) and transfer them to an auxiliary building.

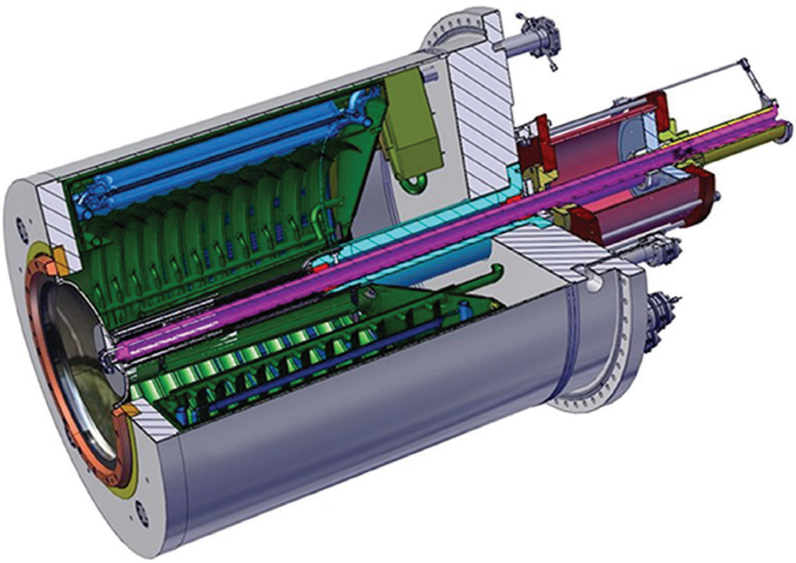

Figure 6. ITER facility cryopump. © ITER Organization.

- **the magnetic system;**

  The magnetic system is made up of superconducting field coils cooled by circulation of liquid helium at a temperature of 4.5 K. It is mainly made up of toroidal field coils, poloidal field coils and the central solenoid, which produce the magnetic field needed to confine the plasma within the vacuum vessel.

- **the cryostat;**

  All magnetic field coils are enclosed in a cylindrical stainless steel vacuum enclosure, the "cryostat" (Figure 7), whose main function is to maintain the extremely low temperature conditions needed for superconducting coils to operate. The cryostat itself is entirely enclosed in a biological shield housing which limits worker exposure to ionising radiation from the vacuum vessel.

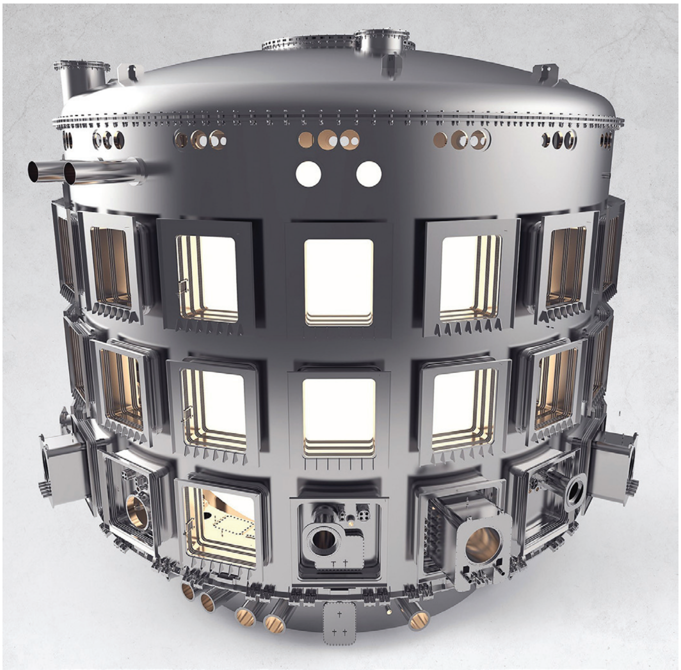

Figure 7. ITER facility cryostat. Approximately 30 m in height and diameter, with a total mass of 3,800 tonnes of steel, it has 280 penetrations to provide access for pipework and electrical power and for heating, diagnostics and remote handling systems. © ITER Organization.

- **equipment for monitoring plasma behaviour and performance**;

  The main equipment for monitoring plasma behaviour and performance in the vacuum vessel is as follows:

  – magnetic measurement systems for monitoring plasma shape and position,

  – optical systems for monitoring plasma temperature and density profiles,

  – spectroscopic devices and particle analysers to determine the characteristics of various particles (ions, electrons, α particles, impurities, etc.),

  – neutron measurements to calculate the quantity of fusion power produced.

The vacuum vessel is surrounded by a large number of auxiliary systems – equipment for cryogenics, measurement and monitoring, additional heating, fuel supply, cooling, vacuum and electrical power – which work together to create and maintain the conditions required for fusion.

- **equipment for additional heating of the plasma;**

  Additional heating of the plasma is provided by various components using two different heating methods: injection of a beam of high-energy neutral atoms and emission of electromagnetic waves.

- **fuel supply lines (for deuterium and tritium) ;**

  The deuterium and tritium introduced into the vacuum vessel come from the tritium building, which will be described later in this document.

- **cooling systems;**

  Radiative heat from the plasma, heat due to the slowing of neutrons and heat produced by neutron reactions in the vacuum vessel and its internal components is removed by three primary cooling systems: one for the vacuum vessel, one for the blanket and one for the divertor. These primary cooling systems are themselves cooled *via* heat exchangers by a secondary cooling system, which transfers its heat to a third system, made up of cooling towers for facilities like ITER which do not produce electricity, or feeding a turbine generator for "DEMO power reactors".

- **disruption mitigation systems;**

  A plasma is subject to multiple types of instability. Small-scale instabilities lead to turbulence similar to that observed in a fluid. This has the effect of mixing the hot particles (ions and electrons) from the centre of the plasma with the colder ones nearer to the edges, but does not destabilise the plasma enough to make it lose its magnetic confinement. However, large-scale instabilities affect the plasma as a whole. These include oscillation, wave propagations and vertical displacements up or down. If the plasma touches the first wall of the blanket or the first wall of the divertor, it completely loses its magnetic confinement in a few milliseconds. This sudden plasma termination is called "disruption".

Disruption is the name given to loss of plasma confinement in the vacuum vessel. Systems to mitigate this are being studied.

Under the current operating conditions of a tokamak, disruptions are frequent, which is one of the reasons why operating sequences with plasma, called "plasma discharges", remain very short (usually around a few tens of seconds, although the Tore-supra tokamak achieved a duration of around 7 minutes in 2003).

Given the large amount of energy stored in the plasma, when a disruption occurs it causes physical phenomena (thermal shocks,

In summary:
ITER will be the largest facility in the world using a tokamak and is designed to demonstrate the scientific and technical feasibility of fusion power. The hot plasma in which the nuclear fusion reaction occurs is formed by a strong magnetic field. The fusion reaction between deuterium and tritium atoms is the one that produces the most energy at the "lowest" temperatures. The plasma will be confined in the vacuum vessel. The blanket protects the vacuum vessel and the superconducting magnets from the energy carried by the neutrons produced. For ITER, this energy will be removed but ultimately, for a DEMO reactor, it will be converted into electricity. The divertor located in the vacuum vessel removes fusion reaction products. All of the vacuum vessel internal components are subject to very large energy fluxes which require penetrations to allow their removal for maintenance. Auxiliary systems located around the central vessel create the conditions required for fusion. A major future issue is the development of tritium breeding blankets to provide reactors with self-sufficiency in tritium.

electron beams, eddy currents, etc.) which have definite consequences (electromagnetic loads in the vacuum vessel internal components and the vessel itself, production of dust from erosion of the first wall of the blanket) or lead to a risk of water leaks.

To prevent the occurrence of disruptions as far as possible, systems are being studied to shut down the "plasma discharge" (for example by a large injection of gas) without a disruption. Such systems could be added to the ITER tokamak and will most likely be planned from the design phase for future nuclear fusion facilities.

3/ Design of a nuclear fusion facility

In a nuclear fusion facility, the tokamak, located in the main “tokamak building”, is surrounded by rooms that mainly house the primary cooling systems (Figure 8) and the vacuum vessel relief tank.

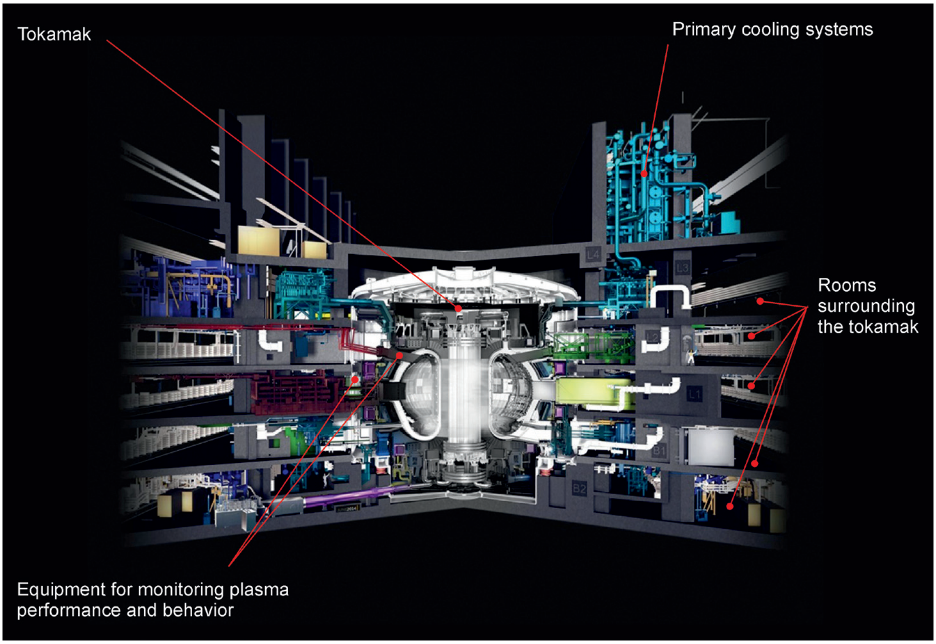

Figure 8. ITER facility tokamak building. © ITER Organization.

The vacuum vessel fuelling lines (deuterium and tritium) and the cryopumps that remove the fusion reaction products, are located in the "tokamak building", but are only part of the fuelling system. This system recycles any unburnt fuel present in the products removed from this vessel, along with any produced in the tritium breeding blankets back to the vacuum vessel. Other fuelling system equipment is located in an auxiliary building, called the "tritium building" (Figure 9). This equipment (see Figure 18):

- processes products extracted from the vacuum vessel and tritium breeding blankets, i.e. separates hydrogen isotopes (hydrogen, deuterium, tritium) from other gases (He, Ar, etc.) and impurities (dust, etc.);
- performs hydrogen isotope separation. The isotope separation system comprises cryogenic distillation columns, which operate based on the different boiling points of the various isotopes. The separated hydrogen isotopes or mixtures of these isotopes are then stored. Fuel is stored in metal hydride beds (zirconium, titanium, uranium etc.);
- provides fuelling for the vacuum vessel.

The buildings around the vacuum vessel provide cooling for the components located around the vessel, and extract the reaction products for removal as waste or for recycling back into the vessel.

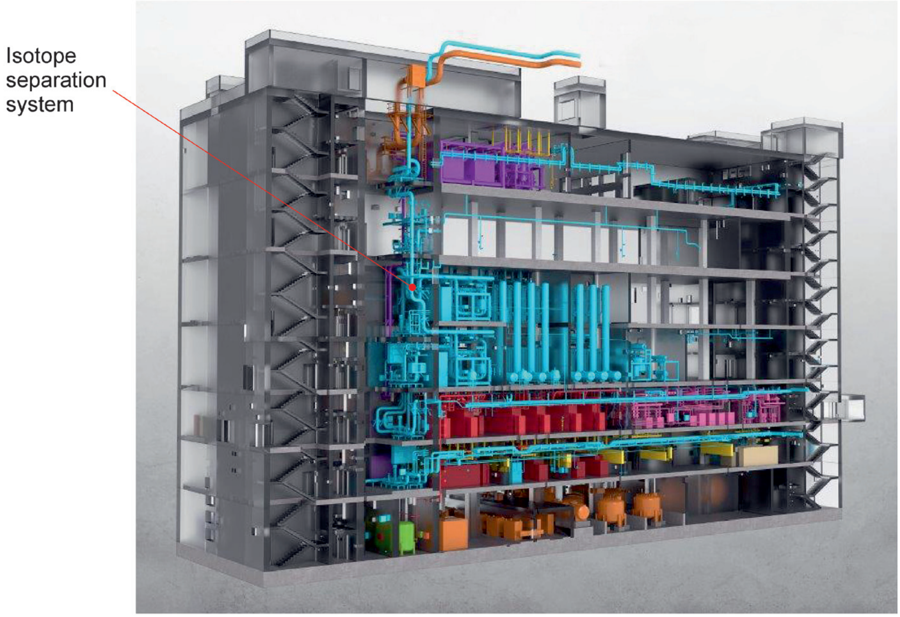

Figure 9. ITER facility tritium building. © ITER Organization.

As seen above, vacuum vessel internal components require periodic maintenance. This maintenance is performed in the cells located in an auxiliary building, called the "hot cell building" (Figure 10). This building may also be used to process and store intermediate level long-lived waste (ILW-LL) and tritiated waste.

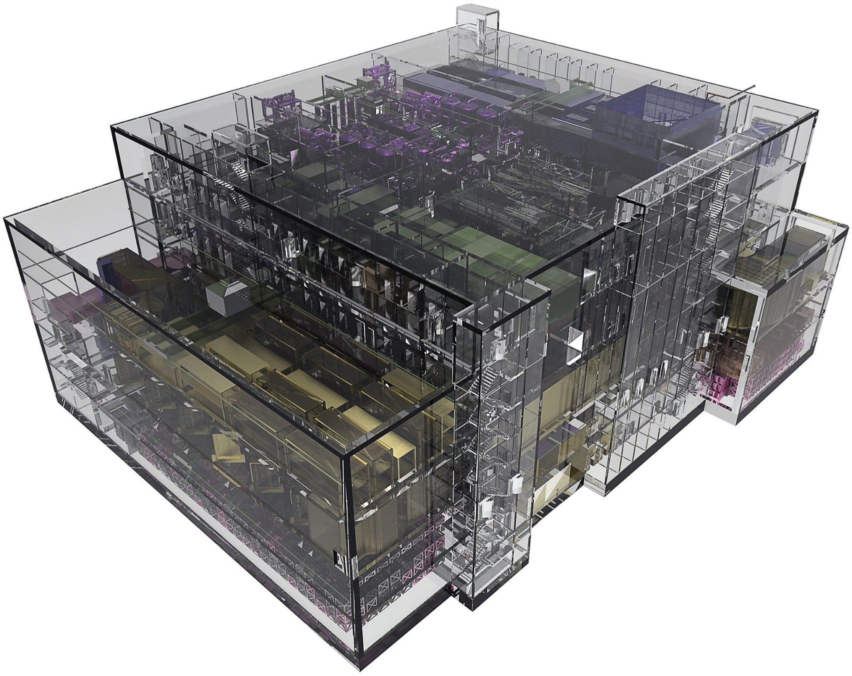

Figure 10. ITER facility hot cell building. © ITER Organization.

It should be noted that the dose rates resulting from activation of vacuum vessel internal components are such that parts of this equipment can only be removed by robotic means and that their transfer from the vacuum vessel to the cells can only be performed by automated transfer casks (Figure 11).

Other waste, whether low and intermediate level short-lived waste (LILW-SL) or very low level waste (VLLW), is processed and stored in another auxiliary building, called the "radwaste building".

All buildings are fitted with ventilation systems which, along with the static confinement provided by the process equipment, rooms and buildings, provide confinement of radioactive substances. The ventilation systems of a large number of rooms and equipment items (glove boxes, etc.) include equipment for air detritiation under normal and accident operating conditions.

Due to high dose rates, robotic means are used to transfer equipment requiring periodic maintenance.
All nuclear buildings (tokamak, tritium, radwaste, hot cell) have ventilation systems that provide confinement of radioactive substances.

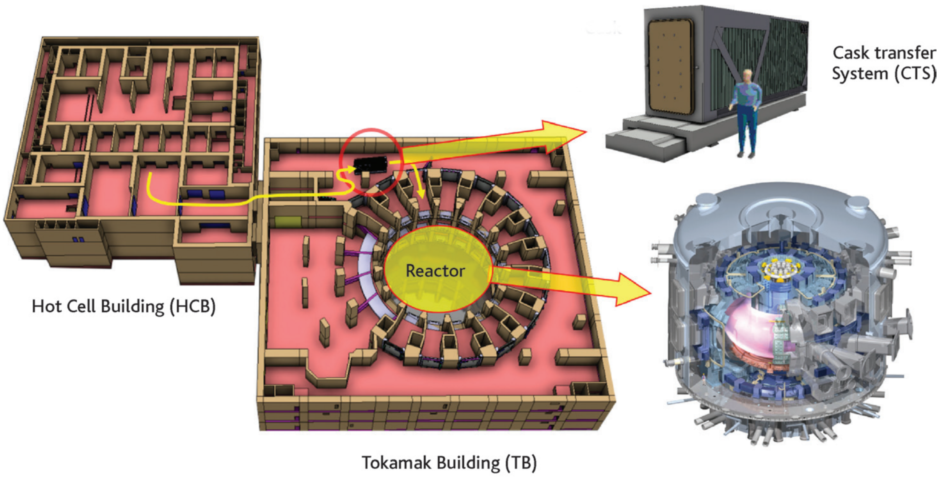

Figure 11. ITER facility automated transfer cask [2]. © 2017 Elsevier B.V.

Two types of radioactive substance are found in a nuclear fusion facility:

- **tritium;**

  Tritium is most often found as a gas (T_2), which easily diffuses across seals and certain materials, or as tritiated water (HTO), which is often corrosive for the equipment involved.

  The tritium needed for fusion reactions will come from outside the ITER facility, and will be produced in tritium breeding blankets for DEMO reactors.

  Most of the tritium is located in the "tritium building" (several kg) but a significant quantity is also found in the vacuum vessel (in the first wall of the blanket and in the lithium-bearing part of the blanket), and to a lesser extent, in the hot cell building due to the maintenance operations performed on vacuum vessel internal components and waste storage.

  Without intervention, the quantities of tritium adsorbed onto the first wall of the vacuum vessel internal components could become very large. The operators of nuclear fusion facilities limit the quantities of tritium adsorbed in order to avoid losing too much fuel and to avoid excessive consequences in the event of accidents involving tritium. To reduce the quantities of tritium adsorbed, they use thermal desorption to remove the tritium from the first wall of the vacuum vessel internal components, by operating the primary cooling systems at a temperature much higher than under normal operation.

- **activated materials or products;**

  Interactions between neutrons from fusion reactions in the plasma and the nearby environment lead to the production of the following:

  - activated structural materials,

    The blanket and divertor are highly activated, as are to a lesser extent the vacuum vessel, the field coils, the cryostat and the equipment for monitoring plasma behaviour and performance, for additional plasma heating, for the fuelling system, and for the cooling and vacuum systems. If the blanket is tritium breeding, it will also contain activated fluids, gases and solids, which vary depending on the type of tritium breeding blanket used.

  - activated dust,

    The dust resulting from erosion of the first wall of the vacuum vessel blanket and the first wall of the divertor is activated; it also contains adsorbed tritium. Without intervention, the quantities of activated dust in the vacuum vessel could become very large. As with tritium above, operators of nuclear fusion facilities limit the quantities of dust as it disturbs the plasma. In addition, the consequences of accidents that could involve this dust must not be too serious. For these reasons, dust is regularly removed from the vacuum vessel using robotic vacuum cleaners. However, this results in significant storage of activated dust in the hot cell building.

  - water activation products,

    The activation products of the water in the primary cooling systems are tritium, carbon-14, nitrogen-16 and nitrogen-17. However, tritium production by activation of the water in primary cooling system is negligible compared with the quantities of tritium which diffuse from the vacuum vessel into the cooling systems *via* their walls.

  - activated corrosion products in water,

    Corrosion products formed in the primary cooling systems which could be mobilised in the event of a water leak are ions in solution and non-fixed deposits on the walls, which are activated during passage through the vacuum vessel internal components and the vessel itself.

Two types of radioactive substance are produced during fusion: tritium which will be produced in DEMO reactors and is introduced in ITER, and activated materials or products. As with tritium, activated dust must be periodically removed, in particular to mitigate the consequences in the event of an accident.

In summary:
The buildings around the vacuum vessel provide cooling for the components located around the vessel, and extract the reaction products for removal as waste or for recycling back into the vessel.
Due to high dose rates, robotic means are used to transfer equipment requiring periodic maintenance.
All nuclear buildings (tokamak, tritium, radwaste, hot cell) have ventilation systems that provide confinement of radioactive substances.
Two types of radioactive substance are produced during fusion: tritium which will be produced in DEMO reactors and is introduced in ITER, and activated materials or products. Activated dust and tritium must be periodically removed, in particular to mitigate the consequences in the event of an accident.

– activated gases,

The air present between the cryostat and the biological shield housing, and to a lesser extent, the air in the tokamak building rooms, is activated by the neutron flux produced by the fusion reactions. The main isotopes entailing irradiation risks are carbon-14 and argon-41.

Inert gases (nitrogen, neon, etc.) which are introduced in small quantities between the plasma and the divertor to reduce radiative heat transfer to the first wall of the divertor are also activated.

4/ The ITER experimental fusion facility

The general goal of the ITER facility is to "*demonstrate the scientific and technological feasibility of fusion power by producing around 500 MW for pulses of several hundreds of seconds*".

During the 20 years of operation planned for this facility, around 40,000 plasma discharges could be performed.

Other than the fusion power targeted, the main characteristics of the ITER facility tokamak are as follows:

- a vacuum vessel with a volume of approximately 1,600 m^3;
- a nominal plasma current of 15 MA;
- a beryllium first wall for the blanket;
- a tungsten first wall for the divertor;
- six tritium breeding modules out of the 440 modules of the blanket; the six places reserved for these modules mean that the seven members of the international ITER Organization for fusion energy (the People's Republic of China, the European Atomic Energy Community, the Republic of India, Japan, the Republic of Korea, the Russian Federation and the United States of America) can test the various types of tritium breeding blankets they are developing.

In summary:
The general objective of the ITER facility is to "*demonstrate the scientific and technological feasibility of fusion power by producing around 500 MW for pulses of several hundreds of seconds*". Almost 40,000 plasma discharges are planned, constrained by the limited quantities of tritium authorised in the facility. This facility offers the possibility of studying the performance of the auxiliary systems (heating, remote handling, cryogenics, etc.) and performing tritium production experiments *via* the six places reserved for tritium breeding modules among the 440 blanket modules.

The authorised operating domain for the ITER facility is limited by:

- a quantity of tritium in the whole facility less than or equal to 4 kg,
- a quantity of tritium in the vacuum vessel less than or equal to 1 kg,
- a mass of dust in the vacuum vessel less than or equal to 1,000 kg (tungsten and beryllium).

5/ Nuclear fusion reactor projects

Different countries have taken different approaches with regard to how to follow on from the ITER experimental facility with a view to nuclear fusion power generation. However, two general strategies emerge:

- before building a fusion reactor for power generation, some countries are aiming to demonstrate its feasibility using smaller experimental reactors (fusion power below 1,000 MW); this is the case for the People's Republic of China, the United States of America and the Republic of India. In the remainder of this document, these projects are collectively referred to as "intermediate experimental reactors";
- other countries are aiming to directly build an experimental fusion reactor for power generation, with a fusion power output over 1,000 MW; this is the case for the Republic of Korea, the European Atomic Energy Community and Japan. In the remainder of this document, these projects are collectively referred to as "fusion power reactors";

Internationally, countries have opted either to directly build fusion power reactors or to first build experimental reactors.

To date, the Russian Federation has no known plans for a tokamak nuclear fusion facility.

5/1 Intermediate experimental reactors

5/1/1 China's "CFETR" project

The People's Republic of China considers that before building a fusion power reactor, it is necessary to:

The Chinese project aims to study the behaviour of the tritium breeding blanket and the resistance of the materials, and to optimise the design of the divertor, with a fusion power output of up to 1,000 MW. Maintenance *via* middle-plane, upper and lower penetrations has been retained.

- demonstrate the real effectiveness of the implementation of a tritium breeding blanket (self-sufficiency in tritium, tritium processing, blanket cooling);
- demonstrate the resistance of the materials used under high neutron flux;
- improve the design of the divertor.

In view of this, the People's Republic of China plans to build a facility called the China Fusion Engineering Test Reactor (CFETR). Operation will start with a fusion power output of 200 MW, rising to 1,000 MW in the second phase of CFETR facility operation. This project has been the subject of quite a large number of publications.

The People's Republic of China has not yet determined the type of tritium breeding blankets to be used on the CFETR facility. They have been studying several types of tritium breeding blanket.

The tokamak configuration adopted is one with middle-plane, lower and upper penetrations of the vacuum vessel, used for extracting very large sectors of the blanket and divertor for maintenance (Figure 12).

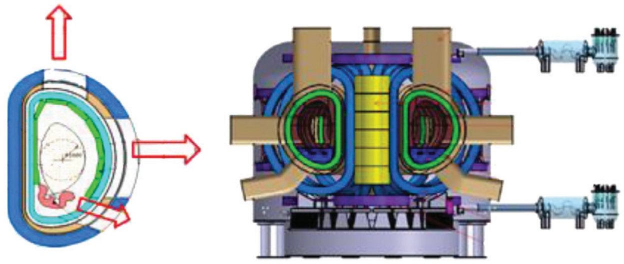

Figure 12. Design of the Chinese CFETR reactor, from [3] and [4]. © DR.

5/1/2
America's "FNSF-AT" project

To date, there has been no US government decision to build a DEMO reactor. However, a strategy has been proposed by the Department Of Energy (DOE). This suggests that, before deciding to build an experimental fusion power reactor, a facility called the Fusion Nuclear Science Facility Advanced Tokamak (FNSF-AT) should be built, to demonstrate that:

- nuclear fusion facilities can be self-sufficient in tritium while producing electrical power;
- the blanket and divertor can be used for a relatively long time without needing maintenance;
- the physical phenomena involved are well understood (damage caused by neutron interactions with materials, tritium behaviour in the first wall of the blanket and divertor, etc.).

Even if design studies for the FNSF-AT were launched in the next few years, the facility would not be operational before 2030. An experimental fusion power reactor is therefore not conceivable before 2050.

To be adaptable based on developing knowledge, the FNSF-AT is to be modular, in particular thanks to implementation of vertical penetrations for the extraction of vacuum vessel internal components (vertical maintenance, Figure 13). The fusion power output of 125 MW at commissioning could be raised to 250 MW then 400 MW. The target is for a mean operating time with plasma of approximately 30% per year.

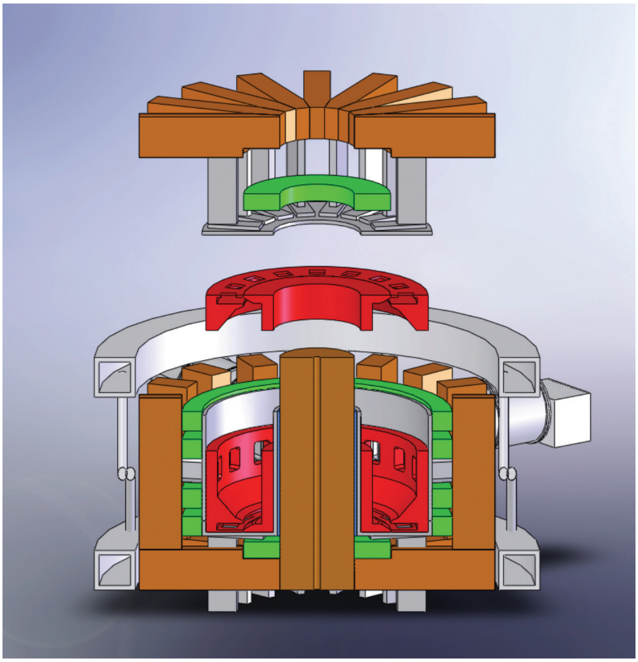

Figure 13. Design of the American FNST-AT reactor project [5]. © A. M .A. Garafalo/General Atomics.

The American project aims to develop a system that is self-sufficient in tritium by studying two types of tritium breeding blanket which also allow for electrical power generation, to optimise the service life of vacuum vessel components and to study the physical phenomena involved, for fusion power output of up to 400 MW. Maintenance *via* vertical penetrations is preferred. The quantity of tritium used would be approximately 4 kg for an operating time with plasma of approximately 30% per year.

Two types of tritium breeding blanket are under consideration:

- Helium Cooled Ceramic Breeder (HCCB), using a lithium-based oxide ceramic for tritium production, beryllium as a neutron booster and helium to cool the structures;
- Dual Coolant Lithium Lead (DCLL) using a LiPb eutectic for tritium production, as a neutron booster and for cooling the structures, along with helium for cooling the structures. The DCLL blanket seems to be the preferred option.

The quantity of tritium present in the whole facility would be around 4 kg.

5/1/3
India's "SST-2" project

India's project aims to study various types of tritium breeding blanket and various components for a fusion power output of up to 500 MW.

The Republic of India's programme envisages the construction of a facility called the Steady State Superconducting Tokamak-2 (SST-2) by 2027, and a fusion power reactor with fusion power output of 3,300 MW by 2037 (Figure 14). The main purpose of the SST-2 facility would be to test the equipment for the future power reactor, in particular the tritium breeding blankets. Operation will start with a fusion power output of 100 MW, rising to 500 MW in later SST-2 facility operation. Very little has been published on this programme to date.

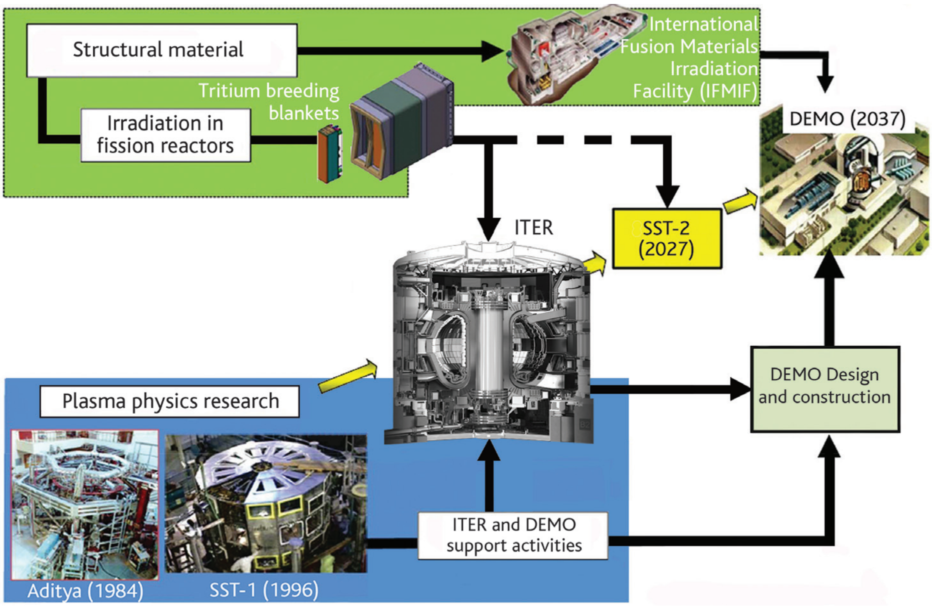

Figure 14. Republic of India Roadmap, from [6]. © DR.

5/2
Fusion power reactor projects

5/2/1
South Korea's "K-DEMO" project

By 2037, the Republic of Korea plans to build a reactor called "K-DEMO" whose vacuum vessel will have a similar volume to that of the ITER facility. It would be operated in two phases:

- an initial phase would serve to test various components (tritium breeding blankets, equipment for monitoring plasma behaviour and performance, etc.). During this phase, the fusion power output would be limited to 10% of maximum capacity, with an availability of around 10-20%;
- following replacement of certain components, the aim of the second phase would be to produce significant electrical power, with availability of at least 70%, with a view to the design of future industrial fusion power reactors.

The South Korean project aims to build a DEMO reactor of similar size to ITER, which would ultimately produce significant electrical power, with availability of at least 70% for fusion power output of up to 2,400 MW. Vertical maintenance is preferred. The tritium breeding blanket would be water cooled and made of tritium-producing lithium-based oxide ceramic.

Two fairly similar options are being studied, with 1,700 MW and 2,400 MW fusion power output respectively, and corresponding net power of 100 MWe and 300 MWe. This project has been the subject of quite a large number of publications.

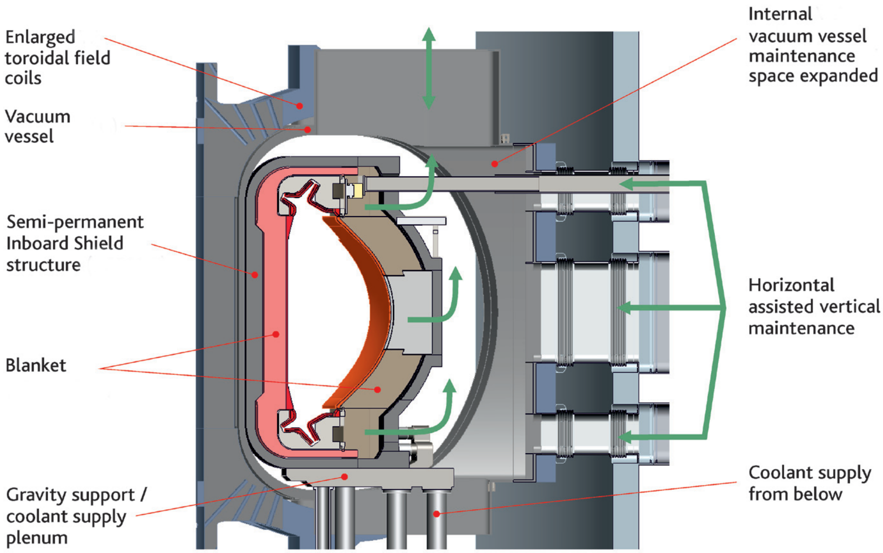

Figure 15. Design of the South Korean K-DEMO reactor, from [7]. © DR.

Water Cooled Ceramic Breeder (WCCB) tritium breeding blankets would be used, with a lithium-based oxide ceramic for the production of tritium, beryllium as a neutron booster and pressurised water for cooling. Vertical maintenance is preferred (Figure 15).

5/2/2
Europe's "DEMO" project

The European Atomic Energy Community is aiming to build a DEMO reactor with a design similar to ITER, which would ultimately produce 500 MW of electrical power with good availability for a fusion power output of up to 1,950 MW.
Vertical maintenance seems to be preferred. Several types of tritium breeding blanket are under consideration.

The European Atomic Energy Community aims to operate a European DEMO reactor around 2040. Its operation will start with an initial DEMO 1 phase, during which, based on designs similar to those of the ITER facility, the tritium breeding blankets and the divertor will be improved and the availability will be much higher (approximately 30%). The purpose of the second operating phase (DEMO 2) will be to supply a net electrical power of around 500 MWe from fusion power output of approximately 1,950 MW, with good availability. This project has been the subject of a large number of publications.

Several types of tritium breeding blanket are being studied with a view to making a choice around 2020.

For the general design of the European DEMO project tokamak, vertical maintenance seems to be preferred (Figure 16).

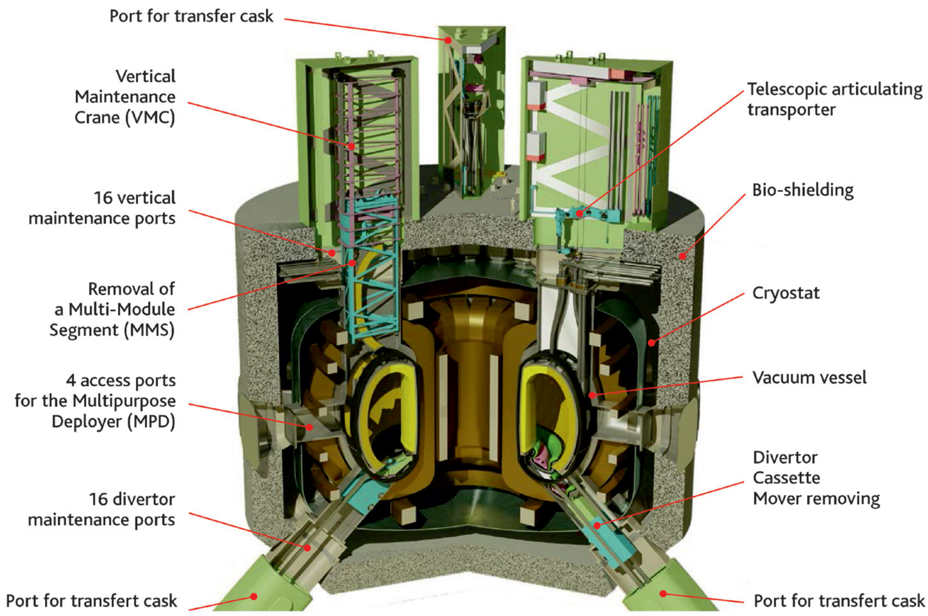

Figure 16. Design of the European DEMO reactor [8].

5/2/3
Japan's project

Until 2014, Japan had long presented the SlimCS ("slim" central solenoid) project, with fusion power output of approximately 3,000 MW, as the next fusion facility for this country. Then, Japan turned to a new project with fusion power output of 1,300 to 1,500 MW. For want of an official name, this project, which has been the subject of a few publications, will be referred to as the "new Japanese project" in the remainder of this document. Finally in late 2015, Japan published a roadmap, drawn up by a team made up, at the request of the of Ministry of Education, Culture, Sports, Science and Technology (MEXT), of members of the National Institute for Fusion Science (NIFS) and the Japan Atomic Energy Agency (JAEA), which foresees construction in around 2030 of a DEMO reactor with fusion power output of around 1,000 MW [9].

The Japanese project aims to build a DEMO reactor with fusion power output of up to 1,000 MW.

6/ The main differences between the planned reactors and the ITER experimental facility

While the conceptual design of the tokamak is the same for the ITER experimental facility and the planned nuclear fusion facilities, the latter differ mainly by aiming to be self-sufficient in tritium and by significantly longer operating times. These differences have a significant impact on the design of the new facilities.

The main development priorities associated with planned nuclear fusion reactors are aiming to obtain self-sufficiency in tritium and to significantly increase operating times with plasma.

6/1 Self-sufficiency in tritium

Tritium from the detritiation of heavy water from current CANDU reactors, which will be used for the ITER facility, will not be available in sufficient quantities to supply DEMO reactors which will need much more tritium. A DEMO reactor with fusion power output of 1,000 MW operating continuously will consume approximately 60 kg of tritium annually, whereas the ITER facility will only need 20 kg of tritium over its 20 years of operation during which it will operate with plasma on average approximately 1% of the time.

To supply tritium for DEMO reactors, the tritium breeding blankets of these reactors are now set to produce more tritium than the reactor consumes, in order to take account of tritium losses due to its radioactive decay and those due to its permeation in the reactor equipment. A nuclear fusion facility should also be able to produce the tritium needed for start-up of the new nuclear facilities built after it, until these facilities themselves become self-sufficient. The tritium breeding blankets are therefore characterised by the ratio between tritium production and consumption in the facility, called the Tritium Breeding Ratio (TBR), which should be greater than 1. In various publications, the minimum TBR for a DEMO reactor to be self-sufficient ranges from 1.1 to 1.2.

For self-sufficiency in tritium, the ratio between tritium production and consumption in the facility must be greater than 1 and as low as possible.

The lower the minimum TBR to be achieved, the easier it will be to obtain the desired self-sufficiency. In this respect, an important parameter is the percentage of tritium introduced into the vacuum vessel that undergoes a nuclear fusion reaction with deuterium, called the "burn-up fraction" in the plasma. This fraction is low because the fuel that is introduced into the vacuum vessel does not all reach the interior of the plasma and because only part of the fuel that enters the plasma undergoes fusion reactions, as the presence of helium in the plasma (product of the fusion reactions) hinders fusion reactions. When fuel is introduced into the vacuum vessel as gas, only 5% of the fuel enters the plasma. However, when the fuel is introduced as pellets, 50-90% of the fuel reaches the interior of the plasma. For the ITER facility, the burn-up fraction in the plasma is not expected to exceed 0.3%. Each plasma discharge of several minutes will involve approximately 100 g of tritium, of which approximately 0.3 g will be burnt. Several grams will remain adsorbed on the first walls of the vacuum vessel internal components, and the rest must be recycled in the fuelling system. For DEMO reactors, the projected burn-up fraction in the plasma ranges from 1 to 5% in various publications [10-12]. For the same fusion power, the greater the burn-up fraction, the less tritium needs to be introduced into the vacuum vessel and consequently the lower the quantity of tritium to process in the tritium building. The quantities of tritium lost by radioactive decay and by permeation are also lower and the minimum TBR to be achieved is lower. In summary, the higher the burn-up rate in the plasma, the lower the minimum TBR to be achieved and the more feasible self-sufficiency in tritium appears.

Everything that increases the tritium burn-up rate in the plasma, the tritium production rate in the blankets and the surface area covered by these blankets inside the vacuum vessel, along with everything that reduces tritium recycling time, facilitates the facility being self-sufficient in tritium.

Another factor, which is less influential than the burn-up rate in the plasma, may be considered for reducing the minimum TBR to be achieved. This is the time needed to process tritium in the tritium building before recycling; the shorter this time, the lower the minimum TBR to be achieved. In the ITER facility, there is no tritium recycling during plasma discharges given their short duration. For DEMO projects, the time needed for tritium recycling has been estimated, as an initial approximation, to be between 1 and 24 hours, depending on the recycling process adopted and the associated technologies [13].

Furthermore, to achieve the minimum TBR, tritium production should be as high as possible. For this, the choice of tritium breeding blankets, among the numerous types being studied, will be essential, and the surface area covered by the tritium breeding blankets inside the vacuum vessel should be as large as possible. However, the tritium breeding blankets cannot cover the whole internal surface of the vacuum vessel due to the presence of the divertor and various components that access the interior of the vacuum vessel to inject fuel and additional heat, and to monitor plasma behaviour and performance, along with disruption mitigation systems and those for removing and replacing internal components.

Numerous publications covering the ability of a nuclear fusion facility to be self-sufficient in tritium show that this objective, which is essential for industrial operation of tokamak fusion power reactions, is difficult to achieve.

IRSN considers that the impact on safety and radiological protection of the choices made to become self-sufficient in tritium also need to be taken into account. As will be highlighted later in this document, tritium releases into the environment will strongly depend on the type of tritium breeding blanket adopted and, to a lesser extent, on the target burn-up fraction in the plasma.

6/2 Significant operating times

For the ITER experimental facility, the average operating time with plasma will only be about 1% of the time. For the planned DEMO reactors, the target operating times with plasma range from 30% to 70% of the time.

One of the consequences of this much greater operating time with plasma compared with that of the ITER facility, is that the number of displacements per atom[4] (dpa) due to 14 MeV neutrons in the materials of the structures surrounding the plasma, around 2 to 3 dpa for the ITER facility, could reach 150 dpa for a high-power DEMO with high availability. Furthermore, the quantities of helium and hydrogen produced from neutron reactions in these materials, which will be negligible for the ITER facility, will be significant for DEMO reactors and will constitute an additional source of damage for these materials. The choice of materials able to withstand

(4) Mean number of displacements to which the atoms of a material are subject under a neutron flux. This number measures the modification of the structure of the material, which leads to a deterioration of its initial properties.

intense neutron bombardment is therefore essential for the design of DEMO reactors.

To increase the plasma operating time it is necessary to reduce the duration of maintenance operations in the vacuum vessel. This consists in increasing the size of the components that make up the blanket and therefore the penetrations, the transport vehicles and the hot cells where they will be processed.

The significant operating times sought with plasma also imply major design modifications compared with the ITER facility. The time needed to extract some or all of the ITER facility's 440 blanket modules and 54 divertor cassettes from the vacuum vessel and transport them to the hot cells is totally incompatible with the long operating times targeted with plasma for DEMO reactors (for the ITER facility, it is estimated that it will take about two years to replace the entire first wall of the blanket and the first wall of the divertor; changing the divertor alone would require around six months of shutdown). To reduce maintenance operation times in the vacuum vessel, the designs of the planned DEMO reactors are leaning towards the installation of blankets made up of a few sectors or half-sectors that can be removed from the vacuum vessel *via* very large penetrations (see Figures 12, 15 and 16). The volume and weight (30-720 tonnes) of these sectors or half-sectors implies having automated transport vehicles that are themselves extremely large and heavy. The hot cells that will receive the sectors or half-sectors for maintenance will also therefore be very large (six times the surface area of the ITER facility hot cells for the European DEMO project).

7/ Safety and radiation protection issues to be examined from the design-phase of DEMO reactors

7/1 Residual heat removal

7/1/1 During operation without plasma

For the ITER facility, there is relatively little residual heat to be removed (11 MW at plasma shutdown and 0.6 MW after one day). The internal structures of the cryostat are at very low temperatures. In the event of failure of all cooling systems, the temperature rise in the tokamak structures would be slow, especially since in this situation air would be introduced into the cryostat to further slow this rise in temperature. It would take about four months for the temperature of the divertor to become hot enough to cause failure of its cooling system and consequent water ingress into the vacuum vessel. There could be a large proportion of dust sufficiently hot to be oxidised by water (temperature above about 350°C) and hydrogen production in the vacuum vessel would be significant. A hydrogen or dust explosion in the vacuum vessel would release tritium and dust into the environment.

For DEMO reactors, the residual heat to be removed would be much greater than for the ITER facility, due to the longer operating times with plasma and, for some projects, the greater fusion power. Thus, for a high power DEMO reactor with high availability using the same materials as the ITER facility, the residual heat to be removed could be one or two orders of magnitude greater than for the ITER facility [14]. However, use of lower activation materials(5) (martensitic steel, vanadium alloy, silicon carbide composites, etc.), which is planned for all DEMO reactor projects to limit risks of exposure to ionising radiation, should also reduce the residual heat to be removed. In these materials, elements that undergo rapid radioactive decay after activation, such as tungsten or tantalum, would replace certain elements currently used such as molybdenum, niobium and nickel.

(5)
Roughly speaking, a low activation material should be able to be handled without special precautions after a period of 100 years.

Table 1 below gives the residual heat values from a number of recent publications for several DEMO reactors. They are difficult to compare as the operating times with plasma are rarely specified.

Planned facility	Fusion power	Residual heat		Publication
		At shutdown	After one day	
Européen DEMO	2,000 MW	80 MW	30 MW	[15]
New Japanese project	1,350 MW	38 MW	8.3 MW	[16]
Former Japanese project (SlimCS)	3,000 MW	54.4 MW	11.3 MW	[17–18]
ITER	500 MW	11 MW	0.6 MW	

Tableau 1. Some residual heat values from a number of recent publications for several DEMO reactors.

To give a sense of scale, for a French N4-series pressurised water reactor, providing 1,450 MWe of electrical power, the residual heat is 264 MW immediately after reactor trip and approximately 24 MW after one day.

On the basis of estimates of the residual heat to be removed, designers should assess the possible consequences of total failure of the tokamak's cooling systems. In this regard, a study performed by the Materials Assessment Group (MAG) for the European DEMO reactor project calculated a temperature rise for the tungsten first wall of the blanket of approximately 1,000°C after 10 days, which in the event of probable air ingress into the vacuum vessel due to failure at a penetration given the temperatures reached, could lead to the formation of significant quantities of radioactive tungsten trioxide (WO_3): 10 to 100 kg/hr depending on the type of tungsten alloy for a first wall surface area of 1,000 m². These highly volatile aerosols could be dispersed into the environment in the event of deterioration of the confinement barriers following total failure of a tokamak's cooling systems [19, 20].

An assessment by analogy should be made for other DEMO reactor projects, including those for which the residual heat to be removed is equivalent to that of the ITER facility, as the consequences of the rising temperature in the tokamak structures could be different, in particular due to the presence of tritium breeding blankets.

The consequences of the failure of cooling systems in nuclear fusion facilities depend on the residual heat to be removed. For DEMO reactors, the residual heat will be greater than for the ITER facility.

The tokamak design, and in particular the design of its cooling systems, is therefore very heavily dependent on the assessment of issues concerning residual heat removal. IRSN therefore considers that it is essential to cover these issues from the safety options stage.

7/1/2
During sector transfer, and storage and maintenance in hot cells

As mentioned above, DEMO reactor blanket sectors to be transferred to hot cells will be very large. For these reactors (or at least some of these reactors), it is therefore probable that the residual heat of each of these blanket sectors will be significant and that, in contrast to what is planned for the ITER facility, these sectors will need to be cooled during the various transfer operations between the vacuum vessel and the hot cells.

For example, the residual heat from one sector of the former Japanese SlimCS DEMO reactor project was estimated at 4.55 MW at shutdown and 0.26 MW after one month. Under these conditions, the designers considered that it would be necessary to wait one month before transporting such a sector, and that it would need to be cooled during transfer because, without cooling, the sector would reach a temperature of approximately 1,000°C after around 40 days [18, 21].

Designers should therefore examine, from the safety options stage, the possible consequences of total failure of the cooling of a sector when it is transferred between the vacuum vessel and the hot cells, and during processing and storage in the hot cell building (Figure 17).

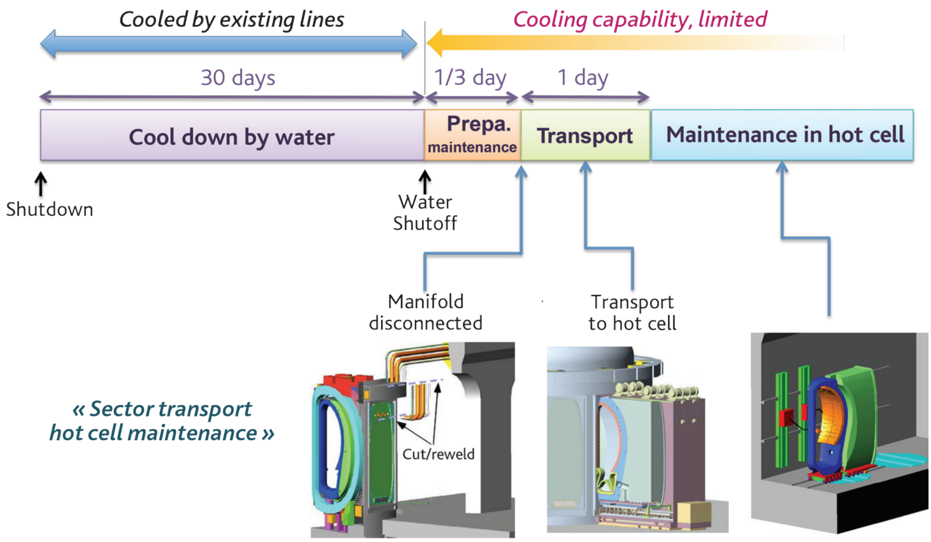

Figure 17. Cooling for the blanket sectors on the Japanese SlimCS reactor project [18]. © Kenji Tobita/JAEA.

- **During preparation of a sector for transfer:**

  During this operation, the sector is disconnected from its normal cooling system to be connected to a transport-specific cooling system. The possible consequences of failure of the transport-specific cooling system should therefore be examined, taking into account possible reconnection of the normal cooling system.

  Furthermore, it should be noted that when a sector to be transferred is disconnected from the normal cooling system, the other sectors in the vacuum vessel must continue to be reliably cooled. The design of the cooling system(s) for blanket sectors must take this requirement into account.

- **During sector transfer between vacuum vessel and hot cells:**

  Study of the potential consequences of a cooling failure during the transfer must consider not only the maximum time such a transfer could take, but also incidents that could lead to the transfer being stopped for a long period. These assessments should take into account the design of the cooling systems installed on the automated transfer systems on which the design of these means of transfer depends, and that of the facility areas to be crossed (size of the automated transfer casks, possible emergency cooling system, etc.). For example, the JAEA considered it necessary to install cooling panels on

Due to the expected size of the blanket sectors of DEMO reactors, the residual heat to be removed will be high. They should be specifically cooled at each step of transfer. Transfer times, cooling systems specific to automated transfer systems and the places of maintenance and storage must be considered.

the walls of a corridor of the former Japanese SlimCS DEMO reactor project so that the concrete of this corridor remained at a temperature below 65°C to avoid deterioration of this confinement barrier due to evaporation of the water contained in the concrete [18].

- **During sector maintenance and storage operations in the hot cells:**

  A certain number of sectors with potentially high total residual heat may be grouped in the hot cell maintenance and storage zones. The designers of the European DEMO reactor considered [8, 22] that stored waste from renovation of a complete set of vacuum vessel internal components should be cooled for approximately 18 months. Again, the possible consequences of failures must be examined from the safety options stage.

The design of the blanket sectors, and in particular their cooling systems in the vacuum vessel, in the automated transfer systems and in their dedicated maintenance and storage areas, is highly dependent on the assessment of residual heat removal issues. IRSN therefore considers that it is essential to cover these issues from the safety options stage.

7/2 Ionising radiation exposure risks

Like the ITER facility, exposure of workers to ionising radiation in DEMO reactors would mainly be associated with human intervention during maintenance operations. Just as for the ITER facility, DEMO reactors will undergo maintenance that largely relies on robotics in order to reduce the doses received by workers. Furthermore, as stressed above, research is underway regarding the use of materials with lower activation under neutron flux than those used on the ITER facility.

The optimisation work conducted for the design of the ITER facility adopted individual internal dose targets close to zero (a no dose target being incompatible with the presence of a large quantity of tritium, which easily diffuses through certain seals and materials) and a mean individual dose of 2.5 mSv/year, with a maximum individual dose of 10 mSv over a year and an annual collective dose of 500 person.mSv. For DEMO reactors, it is not certain that use of low neutron activation materials will be enough to achieve similar values. It is therefore important that the design optimisation work be widely developed, in particular for areas where the ionising radiation exposure risks could be significantly greater than for the ITER facility.

Particular consideration must be given to possible exposure risks resulting from activation of tritium breeding blankets. Activation of the LiPb eutectic liquid used in some types of blanket under consideration, produces mercury-203 (^{203}Hg) and polonium-210 (^{210}Po), whose respective dose factors per ingestion are 100 and 100,000 times higher than for tritiated water. For maintenance phases requiring the opening of a system having carried LiPb, the risk of dispersion of the volatile isotopes ^{203}Hg and ^{210}Po into the room should be taken into account [23]. Furthermore, LiPb is corrosive for the materials of the structures in which it is transported [24]. Precipitation of activated corrosion products in the cold parts of systems (heat exchangers, etc.) is therefore a possibility, which could lead to the creation of areas of high exposure to ionising radiation [13].

Due to the expected large size of DEMO reactor blanket sectors, activation of the tritium breeding blankets and plasma operating times much longer than for ITER, the ionising radiation exposure risks for operators are greater.

Furthermore, there are exposure risks associated with high-energy γ radiation from nitrogen-16 (with a half-life of 7.1 s) produced by neutron reactions with the oxygen in water. For the ITER facility cooling system rooms, given its low operating time with plasma, these risks are avoided by simply prohibiting access to the rooms during operation with plasma. For DEMO reactors, given that operating times with plasma will be much longer, it may be necessary to implement biological shielding in the corresponding rooms [15].

The dose rates from DEMO reactor blanket sectors or half-sectors will be much greater during their extraction for maintenance than those associated with the ITER facility blanket modules. Thus the dose rates associated with a European DEMO reactor half-sector have been estimated, as an initial approximation, at around 3 kGy/hr at contact [25]. Design provisions must be made to limit worker exposure during transfer of these components between the vacuum vessel and the hot cells and during their maintenance and storage in these cells.

As for any nuclear facility, the designer of each DEMO reactor is responsible, from the design phase, for conducting optimisation work for the doses received by operators, leading to the proposal of dosimetry targets.

7/3
Types of accidents to consider

A tokamak is a complex device housing equipment carrying high energy which could cause accidents or have an impact during an accident. This could lead to failure of the first confinement barrier for radioactive substances, made up of the vacuum vessel and its extensions (relief tank, etc.), and even the second confinement barrier, made up of the rooms surrounding the tokamak. The main energies to be considered are:

- plasma energy (approximately 700 MJ for the ITER facility), equally shared between thermal and magnetic energy. Plasma disruption leads to thermal shocks in the first wall of the blanket, with production of dust and the appearance of eddy currents and electromagnetic loads, which are taken into account in the design of vacuum vessel internal components. In some cases, such a disruption can create an electron beam which hits the first wall of the blanket creating damage which can cause cooling system failure leading to water ingress into the vacuum vessel, which could cause release of radioactive substances (tritium, activated dust and activated corrosion products) into the environment (see Chapter 2);
- the magnetic energy of the field coils (around 50 GJ for the ITER facility). In the event of loss of coil superconductivity (for example due to a leak of their liquid helium coolant), the drop in electric current in the coils could lead to the appearance of eddy currents and electromagnetic loads, which are taken into account in the design-basis used for the vacuum vessel internal components and the vessel itself. Furthermore, a short circuit in a coil could cause local deformation of this coil. Finally, an electric arc could occur, causing local melting of the metal structures of the coil and nearby equipment such as the vacuum vessel or the cryostat. The integrity of the vacuum vessel is not affected, unlike that of the cryostat. However, air ingress into the cryostat would not lead to a release of radioactive substances into the environment;
- thermal energy from the primary cooling systems. Any water leak from a cooling system into the vacuum vessel, into the cryostat or into the rooms surrounding the tokamak causes vaporisation and rising pressure phenomena. Water ingress into the vacuum vessel leads to a release of radioactive substances (tritium, activated dust and activated corrosion products) into the environment (see Chapter 2). Water ingress into the cryostat

does not affect its integrity. A cooling water leak into a room around the tokamak would lead to a release of activated corrosion products into the environment *via* the room's ventilation system;

- energy released in the event of an explosion of hydrogen isotopes (hydrogen, deuterium, tritium) or in the event of a dust explosion:
  - an explosion involving deuterium and tritium could occur in the event of air ingress into the vacuum vessel. Furthermore, in the event of water ingress into the vacuum vessel, an explosion of the hydrogen produced by oxidation of dust by water, along with deuterium and tritium, could occur in the vacuum vessel relief tank,
  - a dust explosion could also be triggered by a deuterium and tritium explosion in the vacuum vessel.
- the cooling energy from the system that supplies liquid helium at 4.5 K (the tokamak building of the ITER facility contains about 20 tonnes of liquid helium). Any leak of helium into the cryostat or the rooms surrounding the tokamak would cause sudden vaporisation of the liquid helium, with large volumetric expansion and a pressure rise in the affected area;
- energy from activation products (see Section 7.1.1).

Identification of accidents that could occur in the ITER facility was made difficult by the novelty of the issues to be covered. An initial list of accidents was drawn up pragmatically, based on foreseeable failures that could lead to significant consequences (water leak, air ingress, etc.). More detailed methods were then used, which consist in starting either from component failures or possible causes of releases. Finally, the risk of accidents that could result from an external hazard (earthquake, etc.) or an internal hazard (fire, etc.) were examined. Ultimately, 10 design-basis accidents were covered by the ITER facility operator in a preliminary safety report, which added 12 beyond design-basis accidents, which are either design-basis accidents for which one or more additional failures were considered, or accidents deemed to be of very low probability. Technical discussions during examination of this report led to the operator modifying certain accident scenarios and studying a few more. The list of accidents finally adopted for the ITER facility, and the doses which could be received by the public 2.5 km from the facility are as follows:

- For design-basis accidents:

  - water ingress into the vacuum vessel (long-term dose at 2.5 km: 10^{-4} mSv),
  - air ingress into the vacuum vessel (long-term dose at 2.5 km: 1.3 10^{-2} mSv),
  - vacuum vessel primary cooling system leak outside the vacuum vessel (long-term dose at 2.5 km: 10^{-5} mSv),
  - divertor primary cooling system leak outside the vacuum vessel (long-term dose at 2.5 km: 1.8 10^{-2} mSv),
  - leak in the isotope separation system (long-term dose at 2.5 km: 1.1 10^{-4} mSv),
  - failure of a fuelling line (long-term dose at 2.5 km: 4.9 10^{-3} mSv),
  - loss of confinement in the hot cell building (long-term dose at 2.5 km: 3.6 10^{-4} mSv),
  - leak from a tritiated water tank (long-term dose at 2.5 km: 1.6 10^{-2} mSv),
  - loss of confinement on an automated transfer cask (long-term dose at 2.5 km: 4.3 10^{-3} mSv),
  - failure of the largest opening of a glove box (long-term dose at 2.5 km: 3.1 10^{-5} mSv).

- For beyond design-basis accidents:

  - air ingress into the vacuum vessel and leak into the vacuum vessel from the blanket primary cooling system (long-term dose at 2.5 km: 4.3 10^{-3} mSv),
  - water ingress into the vacuum vessel and failure of a vacuum vessel penetration (long-term dose at 2.5 km: 0.13 mSv),
  - loss of control of the plasma without fusion power shutdown and water ingress into the vacuum vessel (long-term dose at 2.5 km: 3.4 10^{-5} mSv),
  - blanket primary cooling system leak outside the vacuum vessel and failure of the fusion power shutdown system (long-term dose at 2.5 km: 10^{-2} mSv),
  - air ingress into the vacuum vessel causing an explosion (long-term dose at 2.5 km: 0.2 mSv),
  - significant breach of vacuum vessel and cryostat (long-term dose at 2.5 km: 0.3 mSv),

- vacuum vessel primary cooling system leak outside the vacuum vessel and loss of coolant flow in the other systems (long-term dose at 2.5 km: 3.6 10^{-4} mSv),
- water and helium leaks into the cryostat (long-term dose at 2.5 km: 2.4 10^{-3} mSv),
- failure of a fuelling line and failure of the detritiation system (long-term dose at 2.5 km: 1.6 mSv),
- fire in the tritium building (long-term dose at 2.5 km: 0.17 mSv),
- leak from the isotope separation system into its room then explosion (long-term dose at 2.5 km: 0.13 mSv),
- fire in the waste treatment area with propagation towards the storage area (long-term dose at 2.5 km: 0.3 mSv),
- tritium explosion during the regeneration phase of a vacuum vessel cryopump (long-term dose at 2.5 km: less than 0.3 mSv).

The vacuum vessel is the first confinement barrier while the rooms surrounding the tokamak constitute the second. Foreseeable failures or external hazards that could cause radioactive releases into the environment have been considered for the ITER facility. The increased complexity of DEMO reactors and the larger quantities of matter and energy involved mean that the likelihood and consequences of accidents are different than those for the ITER facility.

▸ In addition, the following accidents were excluded:

- presence of an operator near an automated transfer cask carrying a vacuum vessel internal component;
- fall of an elevator with an automated transfer cask carrying a vacuum vessel internal component;
- fire jeopardising the storage of activated dust;
- fire jeopardising the storage of purely tritiated waste;
- spurious opening of a rupture disc or a valve on the vacuum vessel relief line during a plasma discharge.

For the handful of DEMO projects that have been the subject of publications on the accidents considered, it would appear that the types of accidents considered are almost identical to those adopted for the ITER facility. However, the operator must take into account the fact that the likelihoods and consequences could be significantly different, in particular due to the greater quantities of radioactive substances or energy involved. Furthermore, given the increased complexity of DEMO reactors compared with the ITER facility, it is possible that other types of accident may need to be studied (see Sections 7.3.2, 7.3.3, 7.3.6 and 7.3.7 below).

These issues must be examined from the design phase. A non-exhaustive outline of such an examination is given below.

7/3/1
Changes in the quantities of tritium and dust in the vacuum vessel

The consequences of accidental water or air ingress into the vacuum vessel of the ITER facility have been deemed acceptable, given the limitations on the quantities of tritium (1 kg) and dust (1,000 kg) in the vacuum vessel adopted by the operator. On the basis of current knowledge, these values could be reached after one or two years of operation, i.e. after 100-200 hours of operation with plasma. The duration of the thermal desorption operations that will then be performed to reduce the quantities of adsorbed tritium and the cleaning operations to reduce the quantities of dust should not affect the expected availability for this experimental facility.

For DEMO reactors, where the operating times with plasma are much longer, limitations identical to those adopted for the ITER facility would require frequent shutdowns, incompatible with the desired availability rates.

If the designers adopt the same limitations, they will need to put forward provisions to reduce the adsorption of tritium on the structures and to reduce dust creation, or reduce the time it takes for vacuum vessel detritiation and dust removal. Operating with the first wall of the blanket at a higher temperature would promote lower adsorption of tritium. In addition, a higher burn-up fraction in the plasma would also be favourable. Furthermore, the designers could look into ways to control the inventory by attempting to collect dust at the divertor during operation with plasma [26].

The design of DEMO reactors, associated with a limitation on the quantities of tritium and dust in the vacuum vessel, must ensure that the consequences of water or air ingress into the vessel are acceptable for the environment.

The designers could also examine other choices with regard to the limitations on the quantities of tritium and dust in the vacuum vessel and show that, with the new limitation adopted, the consequences of accidents, in particular air ingress into the vacuum vessel followed by an explosion of hydrogen isotopes and dust, would be acceptable, given the design provisions adopted elsewhere.

It is therefore necessary for the corresponding elements to be provided from the safety options stage.

7/3/2
Presence of tritium breeding blankets

The presence of much more extensive tritium breeding blankets in the DEMO reactors compared with the ITER facility means that designers must re-examine the consequences of the accidents considered, and even study other accidents.

The accidents to be considered for DEMO reactors depend on the type of tritium breeding blanket adopted.

To this end, it is necessary to determine the quantities of radioactive substances that could be present in the tritium breeding blankets , which could be very different depending on the type of tritium breeding blanket adopted.

For the ITER facility blanket, the main accident adopted was the risk of a water leak from its cooling system into the vacuum vessel. For the planned DEMO reactors, the tritium breeding blankets could be cooled by liquids (water, LiPb eutectic) or gases (helium). The designers should therefore examine the possible consequences of leaks of these fluids into the vacuum vessel and outside this vessel, and possible violent reactions (including during transfer and maintenance in the hot cells). For example, the risks of interactions between water and beryllium or the LiPb eutectic (leading to hydrogen production) and between helium and this eutectic are to be considered for certain types of blanket [28, 29].

The corresponding elements should therefore be provided from the safety options stage.

7/3/3

Increase in the number of possible cases of loss of control of the plasma

To manage the multiple types of instability that may occur in the plasma, the designers plan to install equipment to monitor plasma behaviour and performance (monitoring the neutron flux, the magnetic field, etc.) and actuators (fuelling control valves, additional means of heating, etc.) with the corresponding instrumentation and control. In the event of malfunction of plasma control, the plasma stops abruptly (disruption).

For the ITER facility, plasma discharges are short discharges, called "inductive scenarios", because the plasma current is produced by gradually increasing the current in the central solenoid. For DEMO reactors, to obtain longer operation with plasma, permanent discharges without any particular limitation of duration are planned, called "non-inductive scenarios". The plasma current is produced by additional heating, to which is added a significant fraction of the current generated within the plasma itself by certain types of instability. This self-generated current is called "bootstrap current".

The plasma control system should be much more developed than for the ITER facility to obtain good control of plasma stability

The number and type of possible malfunctions of the plasma control system for DEMO reactors will be different from those considered for the ITER facility.

under these operating conditions. This means that the possible malfunctions of the plasma control system will be more numerous and different from those considered for the ITER facility (sudden increase in fuelling rate or additional heating, etc.), and some of these malfunctions could be more severe.

Furthermore, the designers must also examine all cases of malfunction of the disruption mitigation systems (gas injection), in particular the consequences of spurious trip of these systems, which could lead to a plasma disruption whose severity needs to be assessed.

It would seem necessary that the possible consequences of the most severe disruptions that could occur, given the design provisions adopted, be estimated at the safety options stage, with suitable substantiation of the choice of scenarios studied.

7/3/4

Increased magnetic energy of the toroidal field coils

The ITER facility was designed with dimensions such that the integrity of the vacuum vessel could not be affected by an electric arc in one of the 18 toroidal field coils. For certain DEMO reactors, the magnetic energy of the toroidal field coils could be significantly higher than in those of the ITER facility. For example, the magnetic energy of the toroidal field coil of the European DEMO project could be around 10 GJ (compared with 2.28 GJ for an ITER facility field coil). Loss of integrity of the vacuum vessel due to an electric arc on a field coil could lead to ingress of vacuum vessel cooling water into both the vessel itself and the cryostat, an accident which has not been examined to date.

For DEMO reactors, the risk of loss of integrity of the vacuum vessel due to an electric arc in a toroidal field coil should therefore be examined from the safety options stage.

7/3/5

Increase in the quantities of helium used

Any significant leak of liquid helium, a fluid used for cooling the superconducting magnetic field coils of the ITER facility, can put the room or equipment where it occurs under pressure and affect its integrity. In particular, the operator examined if there is a room where there could be a simultaneous occurrence of dissemination of radioactive substances and a helium leak threatening the

The presence of helium in DEMO reactors, in particular as coolant for the tritium breeding blankets, must be examined given the risk of pressure on the confinement barriers in the event of leakage.

integrity of the room. Specific provisions, in particular the design-basis earthquake used for the helium system, were taken to avoid this possibility. Furthermore, for the ITER facility, the quantity of helium that could be accidentally spilt into the vacuum vessel is limited by design to 25 kg. This is because, in the event of water ingress into the vacuum vessel and more than 25 kg of helium, which is a non-condensable gas, the integrity of the vacuum vessel and of the vacuum vessel relief tank could be threatened.

For some future DEMO reactors, the quantities of helium in the tokamak building will likely be greater, in particular for those which use gaseous helium as the coolant for the tritium breeding blankets. Several publications stress the need to provide expansion volumes to limit pressure on the confinement barriers in the event of helium leakage [30, 31].

It is therefore necessary that for the DEMO reactors concerned, the designers take into account the risks of helium leakage from the safety options stage.

7/3/6 Increase in the number of rooms where there could be significant quantities of hydrogen isotopes outside the vacuum vessel

For the ITER facility, the operator studied the case of an explosion of all the hydrogen isotopes in the room housing the isotope separation system for the fuelling system in the tritium building.

The total quantities of tritium used in some DEMO reactors will be much greater than for the ITER facility. For these reactors, it can therefore be expected that several rooms outside the vacuum vessel could contain significant quantities of hydrogen isotopes.

The risks of explosion of these large quantities of hydrogen isotopes in these rooms must therefore be examined from the safety options stage.

7/3/7 Vertical port in the vacuum vessel

For the DEMO reactor designs that plan for extraction of blanket sectors or half-sectors *via* vertical penetrations, **the risks of such a sector or half-sector being dropped during its removal from the vacuum vessel must be examined from the safety options stage.**

Furthermore, use of a cover slab to protect the tokamak from airplane crashes, as planned for the ITER facility, could be difficult to implement for these reactors. This protection must therefore be examined from the design options stage, taking into account the provisions adopted for the extraction of internal components.

7/3/8 Specification of protection with regard to extreme events

Specification of all essential equipment designed to withstand extreme events must be considered for DEMO reactors from the design phase.

For ITER, this was, of course, only taken into account at a later stage.

Determination of the extreme events to be adopted should take into account uncertainties for the phenomena considered, including the climate changes which could occur by the end of operation of the currently planned DEMO reactors.

7/4 Releases into the environment under normal operation

The radiological impact of releases into the environment due to normal operation of a fusion facility mainly concern gaseous tritium releases. The annual gaseous releases from the ITER facility should not exceed 2.5 g of tritium in years of heavy maintenance (long-term shutdown to change the divertor, planned two or three times in the service life of the facility) and 0.6 g of tritium for other years. To calculate the consequences, tritium is assumed to be in the form of tritiated water (HTO), which is the worst-case inorganic form. Organically-bound tritium (OBT) releases are not expected.

During heavy maintenance years, tritium releases will mainly be due to the degassing of internal components in the vacuum vessel during the long period of this being open and in the hot cells. The release estimates stated above assume that a thermal desorption operation will be performed before opening the vacuum vessel, which should greatly reduce the quantities of tritium present. For years without heavy maintenance, two-thirds of the tritium releases will come from the Tritium Recuperation System (TRS), whose role is to recover most of the tritium present in the activated dust and tritiated ILW-LL waste from the vacuum vessel internal components.

The radiological impact of releases into the environment due to normal operation of a fusion facility is mainly due to gaseous releases of tritium, for which the greater quantities of tritium and the larger number of release paths implemented in DEMO reactors must be taken into account. Similarly, for DEMO reactors there should be a reduction in releases of liquid effluents into the environment from the higher number of air detritiation systems and greater volume of water detritiation systems.

The remaining releases could be caused by permeation of tritium *via* equipment into the rooms.

For DEMO reactors, the use of greater quantities of tritium and additional paths for gaseous tritium releases should be taken into account.

In accordance with the optimisation principle, designers should seek to reduce as much as possible the quantities of tritium in the facility and to examine the main release paths from the design phase.

In addition, for the ITER facility, liquid effluents containing tritium will mainly come from very slightly tritiated liquid effluents produced by the air detritiation systems and from the treatment of water from the cooling systems. The concentration of tritium in these effluents is too low for them to be treated by the Water Detritiation System (WDS), which recovers tritium for reinjection into the fuelling system.

For DEMO reactors, which could contain numerous rooms fitted with air detritiation systems and large volumes of cooling water, **the designers should examine the possibilities of reducing the quantities of very slightly tritiated liquid effluents to be released into the environment [15].**

For the ITER facility, the maximum radiological impact due to radioactive liquid and gaseous effluents is around 2.3 µSv/year[(6)].

(6)
The estimated radiological impact can be compared to the annual permissible dose limit for exposure of artificial origin for the public of 1 mSv/year specified in Article R.1333-8 of the French public health code.

7/4/1
Reducing tritium quantities in the facility

As stated above, the quantity of tritium present in the ITER facility at any one time is limited to 4 kg. The quantities mentioned in publications regarding DEMO reactors vary between 2.5 kg and 7.5 kg [14, 26, 32, 33], most of which is found, just as for the ITER facility, in the tritium buildings, with the remainder mainly in the vacuum vessel. In this regard, during the presentation made by the University of Science and Technology of China at the "3D versus 2D in hot plasma[(7)]" Seminar from 30 April to 2 May 2013, it was shown that, when the quantities of tritium increase in a fusion reactor, releases of tritium into the environment would also increase as it is unlikely that lower leakage rates than those adopted for the ITER facility can be targeted in the design of DEMO reactor equipment [3].

(7)
Seminar on 2 and 3 dimensional modelling of plasmas.

In any case, to reduce the quantities of tritium present, several publications note the interest of introducing a bypass into the fuelling system to supply the vacuum vessel with fuel that has not passed through the isotope separation system (Figure 18). One method considered would produce fuel pellets to be shot into the centre of the plasma, using products extracted from the vacuum vessel by the cryopumps from which only the helium-4 and impurities (dust, etc.) would be removed [26, 33]. Such a modification to the fuelling system could significantly reduce the quantities of tritium in the facility and consequent releases into the environment.

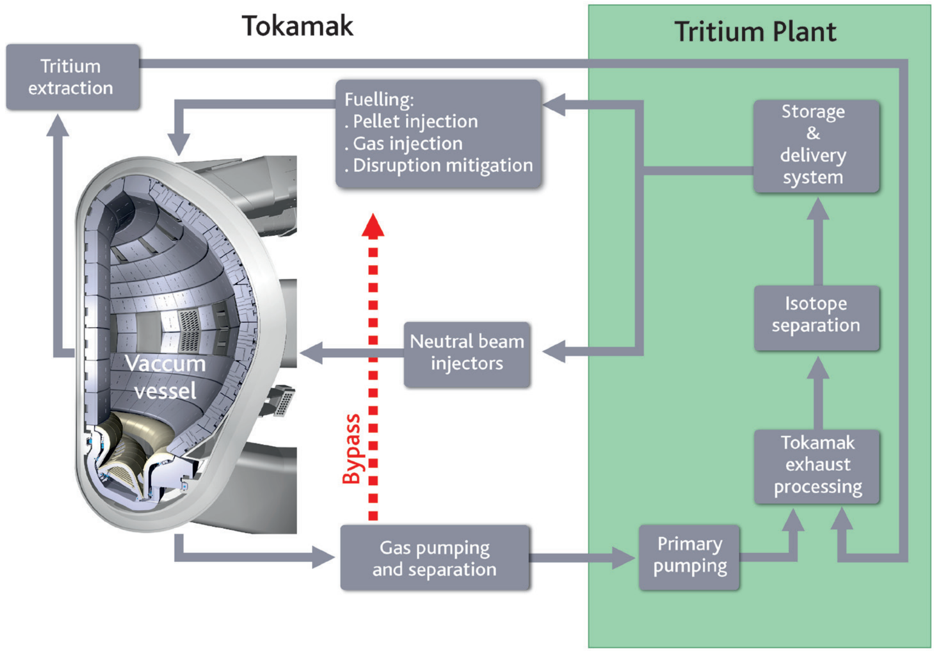

Figure 18. Fuelling system bypass, from [33]. © Georges Goué/IRSN.

Furthermore, it should be noted that, the higher the burn-up fraction in the plasma, the less tritium needs to be introduced into the vacuum vessel and consequently the lower the quantities of tritium to be recycled in the tritium building.

In order to limit tritium releases into the environment, the quantity of tritium used should be reduced as much as possible (bypass, high burn-up fraction, etc.).

Finally, the quantity of tritium present in the tritium building will be smaller the shorter the time needed for tritium recycling in this building.

7/4/2
Examination of the main possible paths for gaseous tritium releases

7/4/2/1
Releases associated with the cooling systems for tritium breeding blankets

In DEMO reactors, part of the tritium produced in the tritium breeding blankets will diffuse *via* the walls of the cooling systems that pass inside the blankets near the areas where tritium is produced. Several publications mention the possibility of around a hundred grams of tritium being found in these systems [34], while the quantities of tritium present in the two primary cooling systems of the ITER facility blanket and divertor are approximately 0.7 g per system. Even if purification systems are implemented to permanently extract tritium from the cooling systems of DEMO reactor tritium breeding blankets, it is clear that some of the tritium present in these systems will diffuse *via* the cooling system heat exchangers into the environment, a release path that does not exist for the ITER facility (Figure 19). Research is underway regarding lining the walls of the cooling systems inside the tritium breeding blankets with a protective layer of alumina (Al_2O_3) or erbium oxide (Er_2O_3) [13], which would reduce permeation by a factor of 50 to 1,000 [19, 35]. However, this research is still only at laboratory scale.

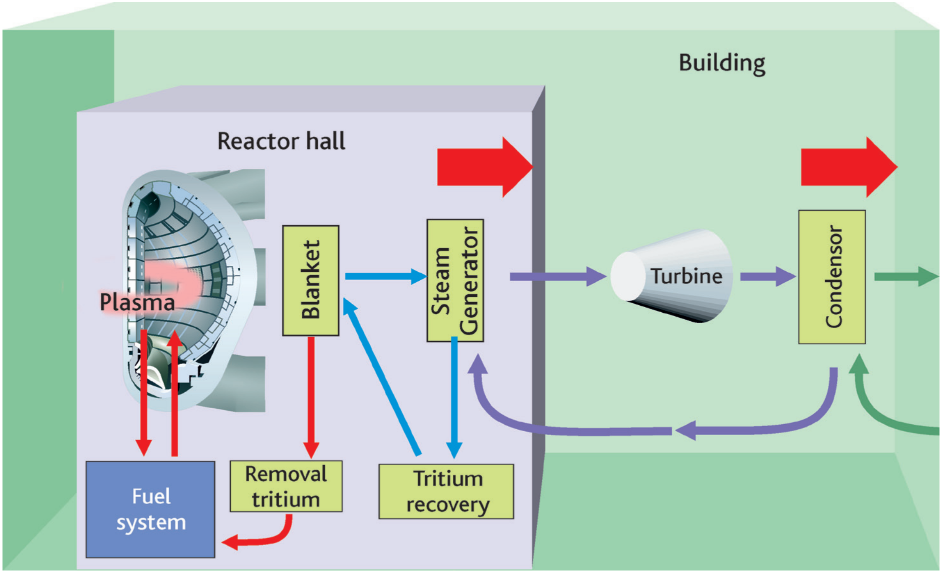

Figure 19. Releases associated with the cooling systems for tritium breeding blankets, from [36]. © Georges Goué/IRSN.

Initial estimates of releases from planned DEMO reactors vary greatly depending on the type of tritium breeding blanket adopted. Thus, for the European DEMO reactor, the releases associated with the use of WCLL tritium breeding blankets (lithium-based oxide ceramics for tritium production, beryllium as neutron booster and water for cooling) are much lower than those that would be associated with the use of HCPB blankets (lithium-based oxide ceramics for tritium production, beryllium as neutron booster and helium for cooling) or HCLL blankets (LiPb eutectic for tritium production and as neutron booster, helium for cooling). The stated factors are 55 and 160 respectively [37].

In choosing the type of tritium breeding blanket to achieve self-sufficiency in tritium, there is a need to consider the tritium releases into the environment associated with their cooling systems.

7/4/2/2
Releases associated with transfers of internal components to the hot cells and their processing in these cells

As mentioned above, under normal operation, tritium releases from the ITER facility are mainly associated with transfers of internal components to the hot cells and their processing in the cells during years of heavy maintenance, and with treatment of tritiated waste in other years. It is difficult to get a relative appreciation of the releases associated with planned DEMO reactors as the design of their blanket in sectors or half-sectors is so different from that of the ITER facility and because there are many factors that could affect the volume of releases (quantity of tritium present in the internal components transferred, frequency of transfers, effectiveness of thermal desorption to reduce the quantities of tritium in the vacuum vessel, concentration of tritium in the waste resulting from maintenance of this equipment, effectiveness of the waste detritiation systems).

The designers of DEMO reactors should therefore carefully examine, from the safety options stage, the tritium releases into the environment which could be associated with transfers of internal components to the hot cells and their processing in these cells.

7/4/2/3
Releases associated with waste detritiation equipment

If waste detritiation equipment is used (see Section 7.5 below), this equipment could be a source of releases.

This path of releases into the environment should be considered from the design phase.

7/5
Waste

The type and quantity of tritiated waste produced should be considered based on the general policy of the host country with regard to waste management.

Part of the waste from the ITER facility will contain quantities of tritium that are too high for it to be sent directly to waste repositories in France. It is currently planned that they be stored for approximately 50 years, in the planned INTERMED facility, to reduce the quantities of tritium *via* radioactive decay. Furthermore, the most tritiated waste will undergo pre-processing in the tritium recovery system (TRS).

It is probable that waste produced by DEMO reactors will be more tritiated than that produced by the ITER facility. This could lead designers to add specific detritiation equipment (thermal furnace, melting furnace, incinerator, etc.). Furthermore, the tritium breeding blankets could be a source of waste different from that from the ITER facility, depending on the type of tritium breeding blanket adopted. In any case, management of DEMO reactor waste will depend on the general policy of the host country in this regard.

DEMO reactor designers should examine waste management constraints from the safety options stage, taking into account the general policy of the country that will host the reactor.

8/ Conclusion

The considerations above have been established on the basis of the experience acquired during the safety assessment of the ITER facility and publications available on planned DEMO reactors in late 2017. Following this work, IRSN stresses that designers should examine the following subjects as a priority:

- residual heat removal, taking into account the design envisaged for the tokamak cooling systems and those for the blanket sectors when they are transferred between the vacuum vessel and the hot cells and during their storage in these cells;
- optimisation of the doses which workers receive depending on robotisation and the choice of materials;
- the types of accident considered for the ITER facility and the specific types of accident that could be associated with the design of DEMO reactors;
- possibilities for limiting the overall quantity of tritium present in the facility and releases by various main paths for gaseous tritium releases. In this respect, the choice of tritium breeding blankets would appear to be a key factor;
- identification of the waste management constraints based on the general policy of the country hosting the reactor.

9/ Glossary

Blanket: a vacuum vessel internal component which provides neutron protection for the metal walls of the vessel.

Collective dose: the dose is the quantity of energy absorbed by a medium from ionising radiation. The collective dose is the sum of the individual doses received by a given group of people (here expressed in person.millisieverts ([person.mSv]).

Cryopump: a vacuum pump system that removes helium and impurities *via* condensation on a cold surface.

Cryostat: a metal chamber maintained at very low temperature that contains the tokamak.

Design-basis: determination of the characteristics of a facility during its design to meet pre-established criteria and regulatory practice.

Disruption: the name given to loss of plasma confinement in the vacuum vessel.

Divertor: an internal component located in the lower part of the vacuum vessel which extracts helium, fuel that has not undergone fusion (and that can be reused) and impurities.

Dose rate: the radioactive dose rate determines the intensity of irradiation (energy absorbed by matter per unit of mass and time). It is measured in grays per second (Gy/s).

Field coils: systems made up of superconducting magnets which generate a magnetic field that creates, confines and shapes the plasma. A distinction is generally made between toroidal field coils (confinement), the central solenoid (field generation and contribution to heating) and poloidal field coils (literally: which develop between magnetic poles). These coils are cooled by pumping liquid helium at a temperature of 4.5 K.

Fusion: the combination of two nuclei of light atoms to form a heavier nucleus, which releases a large amount of energy carried by the reaction products (nuclei, particles and radiation).

Magnetic confinement: plasma moving under the effect of a magnetic field, held at a distance from the walls of the vacuum vessel. The magnetic field only acts on charged particles. All neutral secondary fusion products (impurities, helium-4, etc.) are extracted by the divertor. Neutrons are absorbed by the walls of the vacuum vessel, giving up heat which is removed for ITER and will ultimately be transformed into electricity for a DEMO reactor.

Plasma: hot low-density gas made up of positive ions and electrons, which have been stripped off due to the temperature, produced under the action of a strong magnetic field.

Plasma discharge: the name given to the operating sequence of the tokamak with a plasma.

Residual heat: heat still produced by the activated materials after the shutdown of the fusion reaction.

Solenoid: a device that consists of a coiled electric wire used to create a magnetic field.

Stellerator: a magnetic confinement nuclear fusion device that differs from tokamaks, in which plasma confinement is achieved by a helical magnetic field created by a complex arrangement of field coils outside the torus. This configuration is designed to operate continuously.

Tokamak: a Russian acronym for тороидальная камера с магнитными катушками, meaning “toroidal chamber with magnetic coils” which uses magnetic fields to create, confine and control a hot plasma inside which the fusion reaction can occur.

Toroidal form or torus: having the shape of an inner tube.

Tritium breeding blanket: a blanket that produces tritium while also providing neutron protection for the metal walls of the vacuum vessel.

Vacuum vessel: a sealed toroidal stainless steel chamber in which the fusion reactions (plasma) occur. It is the first confinement barrier for radioactive substances and contributes to plasma stability. It contains the blanket and divertor. It has a double steel wall in which cooling water flows to remove the heat given off by the fusion reactions. This heat will be used to produce electricity in fusion power plants.

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11/ Summary of the IRSN report on the construction license application for the ITER experimental facility

1) Context

In 2010, the ITER Organisation (ITER/O) submitted a construction license application to the French Nuclear Safety Authority (ASN[8]) to build the ITER experimental facility at Cadarache. This construction license application was accompanied by a file containing a "preliminary safety analysis report" outlining the provisions to manage risks related to the facility and measures to limit the likelihood of an accident and its impacts. An "impact study" was also included in the construction license application. This study presented the health and environmental impacts caused by releases of radioactive and chemical effluents produced under normal operation of the facility. The Institute for Radiological Protection and Nuclear Safety (IRSN) examined both the report and the study, and presented its conclusions before the Nuclear Safety Advisory Committee for Laboratories and Plants (GPU[9]) during meetings held on 30 November 2011 and 7 December 2011. Several members

(8) Autorité de sûreté nucléaire.

(9) Groupe permanent d'experts pour les laboratoires et usines.

(10)
Groupe permanent d'experts pour les réacteurs nucléaires.

(11)
Groupe permanent d'experts pour les déchets.

(12)
Groupe permanent d'experts pour les équipements sous pression nucléaires.

(13)
The thermal output of a nuclear reactor is expressed in megawatt thermal (MWth).

from the Advisory Committee for Reactors (GPR[10]), the Advisory Committee for Waste (GPD[11]) and the Advisory Committee for Nuclear Pressure Equipment (GPESPN[12]) were also in attendance at these meetings.

The ITER facility is an experimental installation with the aim of demonstrating the feasibility of controlling nuclear fusion energy during experiments lasting several hundred to several thousand seconds, generating an output of around 500 MWth[13]. The initial preliminary design studies for the ITER facility were led by an international team for a generic site. The studies were completed in 2001 and French authorities proposed Cadarache as a potential location for the new site. In 2002, based on the final documented report prepared by the aforementioned international team and after adapting it to the Cadarache site (the site's seismic spectrum, etc.), the French Alternative Energies and Atomic Energy Commission (CEA) prepared a safety options report. The Institute for Nuclear Safety and Protection (IPSN, which became IRSN) assessed these safety options and presented its conclusions to the GPU, GPR and GPD on 20 November 2002. The safety options were deemed acceptable and a number of requests were asked to be taken into account in the preliminary safety report. Cadarache was chosen as the site for the new ITER facility on 24 June 2005.

2) The ITER experimental facility

The basic concept for the ITER facility is an experimental tokamak, or magnetic fusion device. The fusion reaction takes place inside a torus-shaped plasma made from deuterium and tritium. The plasma is confined by a magnetic field produced by a set of coils. This contains the plasma inside a sealed toroidal chamber, or "vacuum vessel" preventing the plasma from touching the vessel walls.

fusion reactions between the deuterium and tritium produce α particles that transfer their energy to the plasma, and high-energy neutrons (14 MeV) that are slowed down and absorbed in the structures surrounding the plasma (blanket, divertor, vacuum vessel), where a cooling system extracts the energy.

In terms of safety, the facility is characterised by the presence of high quantities of tritium, divided between several large buildings. Besides the building that houses the tokamak, the facility has a "tritium building", used to treat products removed from the vacuum vessel, a "hot cell building", where equipment from inside the vacuum vessel is overhauled, and a "radwaste building".

3) IRSN opinion on safety and radiation protection for the facility

To assess the safety and radiation protection of the world's first fusion facility classified as a regulated nuclear facility, IRSN set up a specific organisational structure characterised by the following:

- technical support from a plasma physics expert, neutronic and activation calculations by the European Organization for Nuclear Research (CERN[14]) in Geneva, and the signature of a collaboration agreement with the Canadian Nuclear Safety Commission on tritium confinement and the impacts of tritium releases on the population;
- the launch of research and development within the Institute on specific safety problems raised by a nuclear fusion facility.

(14) Centre européen pour la recherche nucléaire.

The main conclusions drawn from the assessment of safety and radioprotection of the future ITER facility are presented below. It should be noted that this assessment does not take into account feedback that could be drawn from analysis of the accident that occurred at the Fukushima Daichii plant on 11 March 2011 as ITER/O was asked by the ASN to submit a report on this feedback for 15 September 2012.

Progress of preliminary design studies for the facility

IRSN considers that preliminary designs for equipment in the "tokamak building" were on par at this stage of the facility building process but that they were incomplete for equipment in the "tritium building", the "hot cell building" and "radwaste building". The operator must therefore submit a new, more comprehensive version of preliminary designs for the equipment of the other nuclear buildings prior to the assembly phase for "tokamak building" equipment.

IRSN also considers that the preliminary designs for the automated transfer casks used to move highly activated equipment between the "tokamak building" and "hot cell building" should also be finalised within the same timeframe. A demonstration of the effectiveness of emergency measures planned for any potential failure of these casks should also be submitted with the designs.

Finally IRSN deemed that the limitations on the facility's operating range, which is essential for demonstrating safety and radiation protection, were clearly defined by the operator.

The inventory of radioactive substances in the facility

Besides tritium, it should be noted that nuclear reactions between neutrons from fusion reactions and the environment near the plasma (structures, cooling water, air, etc.) produce activated substances and products. The total inventory of activation products assessed by the operator was considered adequate by IRSN. However, IRSN considered that the operator should continue its activation calculations in order to obtain detailed inventories by radionuclide for the main equipment. This will make it possible to assess the risks associated with exposure to ionising radiation at each work station, and to characterise the produced waste more accurately in order to ensure that it can be disposed of through waste management solutions.

Risk management measures for the facility

IRSN assessed the design of containment systems adopted by the operator to manage risks involving the spread of radioactive substances such as tritium. The evaluation found that the design requires that rooms have relatively substantial leaktightness. IRSN considered that to ensure the stated leaktightness objective, the operator will need to measure the leak rate of these premises when the facility is commissioned and take measures to ensure that these leaktightness levels are maintained throughout operation. Furthermore, the majority of rooms are equipped with detritiation systems that clean the air quite effectively in the event of accidental dispersion of tritium. IRSN considered that the operator must ensure that no accident situations cause ambient conditions or chemical emissions that could significantly reduce the expected efficiency of these detritiation systems.

With regard to risks of exposure to ionising radiation, the assessment found that operation of the ITER facility will involve relatively short experimental campaigns interspersed with long maintenance periods where a large number of staff will be onsite. Maintenance operations on high irradiating components can only be carried out with robotic equipment and access to the areas in question must be prohibited. IRSN considered that the robust design of access control systems for these prohibited areas needed to be demonstrated. In addition, despite the presence of relatively large amounts of tritium gas with a high diffusion level, the operator has set a target as close as possible to zero for the internal dose level for workers. IRSN considered that demonstrating the achievement of this target would require detailed predictive dose assessments at each work station.

IRSN examined risks related to plasma disruptions. Plasma is subject to multiple instabilities, similar to those observed in a fluid. However, the largest instabilities affect the overall stability of the plasma until it collapses. Sudden plasma termination creates electromagnetic loads in the vacuum vessel, which must be taken into account in its design. It can also lead to damage on the first wall of the vacuum vessel blanket and divertor. IRSN considered that the operator needed to continue studies aimed at characterising the most severe instabilities that could affect the plasma of the ITER reactor.

With regard to internal fire risks, IRSN examined the general fire protection measures and considered that these risks should be further analysed in order to demonstrate that the measures are sufficient. The Institute also considered that the operator should examine the risk of filtration systems upstream of the detritiation systems clogging from soot and smoke in the event of a fire.

The risks of internal explosions in the ITER reactor are mainly due to the presence of hydrogen isotopes (tritium and deuterium) in numerous equipment items. IRSN viewed that these risks cannot only be managed through prevention measures. Therefore, during the assessment, the operator supplemented its file by studying the consequences of an explosion of hydrogen isotopes in rooms which house equipment used to transfer or store such isotopes. IRSN considered that for the room housing the cryogenic column distillation system, the operator should take measures to ensure that an explosion of the entire inventory of hydrogen isotopes in these columns does not affect the level of confinement required for the room.

Accidents in the facility

IRSN examined studies of potential accidents considered by the operator. The consequences of these events remain limited. However, IRSN considered that the conservative nature of the scenarios adopted for some of these accidents, in terms of their impacts, should be verified. IRSN also felt that the operator needs to show that the explosion of hydrogen isotopes or dust, which could be caused in the event of air or water ingress into the vacuum vessel, would not affect the confinement of the vacuum vessel and its extensions.

Finally, IRSN viewed that the list of accident situations considered for the on-site emergency plan design-basis studies needed more work.

IRSN also identified some unlikely accident situations that cause loss of the first confinement barrier in the "tokamak building" for which the operator needs to demonstrate that the confinement level of the second confinement barrier is not affected.

Design of important safety elements

The tokamak is supported by 18 metal columns that are embedded in the main base slab of the "tokamak building". The main base slab is part of the building's second confinement barrier. IRSN considered that the operator needs to aim for a more robust design of the supporting elements than that proposed.

The vacuum vessel is the main element of the first confinement barrier of the "tokamak building". The vacuum vessel is a double walled chamber filled with pressurised water. Due to the elements around the vacuum vessel (magnetic coils, penetrations, etc.), it will not be possible to inspect a large portion of the outer wall during operation. IRSN therefore considered that compensatory measures needed to be taken for the design and manufacture of these uninspectable areas.

Waste and effluents produced by the facility

Radioactive waste from the ITER facility will be activated and/or contaminated waste, particularly by tritium, for which waste management solutions have been identified. IRSN felt that the operator needed to give a precise description of how this waste would be managed at the facility, of the design of conditioning for this waste in the facility's storage buildings, and show the compatibility of this conditioning with the waste management solutions. IRSN also considered that work to ensure that this waste is integrated into the waste management solutions needed to be pursued. The Institute also underlined the need to ensure that these waste management solutions are available when waste packages from the ITER facility are ready to be disposed of.

The release of effluents in normal operation was the subject of optimisation work which IRSN considered acceptable. IRSN and Canadian Nuclear Safety Commission repeat calculations confirm that the expected release levels result in very low impacts on the environment.

4) Conclusion

In conclusion, it is IRSN's opinion that the provisions adopted for the ITER facility are acceptable on the whole, given the commitments made by the operator and subject to the response provided for the points raised by the Advisory Committees. Supplementary information requested to improve the design of the facility and its demonstration of safety must be submitted before the start of the tokamak equipment assembly phase.

12/ Main past and scheduled milestones for the ITER facility

Main past milestones:

- 11 and 12 October 1986: decision of the European Atomic Energy Community (EURATOM), Japan, the Soviet Union and the United States to jointly pursue design studies for a large fusion facility, ITER;
- 21 July 1992: signature of the agreement to launch engineering design of ITER;
- July 2001: finalisation of the ITER conceptual design;
- 28 June 2005: Cadarache chosen as the site for ITER;
- January 2007: work begins on the ITER site;
- 9 November 2012: publication of the construction licence decree for ITER in the Official Gazette of the French Republic.

Main scheduled milestones:

- 2020: start of tokamak assembly,
- 2026: first plasma with hydrogen,
- 2037: first plasma with deuterium and tritium.

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