POLYTECHNIC UNIVERSITY OF TURIN

Department of Energy Master of Science in Energetic and Nuclear Engineering

Master Thesis

Advanced high-magnetic field nuclear fusion reactors
Reattori nucleari a fusione avanzati ad alto campo magnetico

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Abstract

The public acceptance of nuclear fusion as a clean and safe energy source is inextricably linked with the radioactive waste it will produce. Innovative solutions must be taken into account to address the challenges of materials activation and the use of Tritium in the fuel cycle.
The relatively short half-life of Tritium makes its supply assessment crucial for future DT plants operation. If it is chosen as fuel, its breeding inside fusion plants and its production from other sources must be sufficient for the operation of the currently planned experimental plants (such as ITER), but a start-up inventory has to be prepared for future plants operation as well.
The DT fusion reaction produces large amounts of highly energetic neutrons that damage the materials and cause induced radioactivity. For this reason it is necessary to evaluate the performance of reduced-activation materials for fusion applications.
The development in high temperature superconducting magnets technology can lead to progress towards more compact and efficient tokamak designs, such as ARC [1], and the possibility of achieving higher plasma parameters. This could open the prospect for burning different fuel mixtures that can help resolve the problems linked with Tritium and the neutron flux. Compact high-field experiments such as IGNITOR and CANDOR were among the first to propose the use of Deuterium and Helium-3.
The aim of this thesis is to address these issues and reconsider the reliance on Deuterium and Tritium as sole fuel choice.
The first sections are dedicated to the description of some physical concepts necessary for the discussion. Then some of the most important tokamak reactors designs will be presented (ITER, ARC, DEMO) along with some compact reactors that were designed to burn Deuterium-Helium3 (IGNITOR, CANDOR, ARIESIII).
The issues related to Tritium supply and demand are described, and the alternative use of advanced fuel mixtures is discussed. In particular Tritium-poor fuels and Deuterium-Helium3 will be analysed, because of their lower neutron production.
The final chapter is devoted to the analysis of materials activation in ARC and some possible solution for its reduction are proposed, namely the use of advanced fuels, reduced-activation materials, or a combination of both.
The studies on neutron-induced activation have been performed on FISPACT-II, considering the first layer of structural material of the ARC reactor. The ultimate goal of this analysis is to find a possible "zero-waste" scenario for this component, that is the satisfaction of the recycling and clearance criteria.

Contents

Chapter 1

Introduction

Nuclear fusion is the process powering the stars in the Universe, that combines lighter nuclei into heavier ones at extremely high temperatures. In a star, like our Sun, tremendous gravitational forces are responsible for the confinement of this reaction and the establishment of suitable conditions for fusion reactions to happen in the first place.
Replicating these conditions and "Bringing the power of the Sun to Earth"[2] is what makes fusion energy a promising option for a sustainable and non-carbon emitting energy supply. Although this fascinating prospect, it still presents so far arduous scientific and engineering challenges.
Nuclear fusion is already achievable in a reactor, albeit not using gravity as in the stars, but mainly using inertial or magnetic confinement; however a net electricity gain has yet to be reached (power used for the reactor operation must be less than the power coming from the reaction, converted in electricity).
Ambitious experimental plants are planned, with the ultimate goal of reaching this objective.
The recent development in plasma physics, materials technology and high-temperature superconductors improve the prospect to reach these goals before the end of the century, and integrate fusion in the power mix.
The public acceptance of nuclear fusion will necessarily be linked with the radioactive waste produced in future plants. The use of radioactive Tritium as part of the fuel cycle seems the best option to obtain a net power gain as soon as possible, but may raise concerns in the future as fusion will become a viable energy source; furthermore the highly energetic neutrons produced will cause materials activation and increase the amount of waste produced.
In order to mitigate these problems, several solutions have been proposed, among which the use of advanced nuclear fuels that may help reduce the amount of Tritium inside the plant and the neutron flux simultaneously.

Chapter 2

Nuclear Fusion Reactors

The aim of this chapter is to give a brief introduction to some main characteristics of nuclear fusion, analysing in details only a few aspects that will be useful in the subsequent chapters. A description of current fusion reactors technology will then be carried out, giving a detailed account of the Tokamak configuration and the most important projects using this design: ITER, DEMO and ARC.

2.1 Theoretical Background

Nuclear fusion is a nuclear reaction occurring when two nuclei overcome the Coulomb force that repels them and form one or more new nuclei and subatomic particles (usually neutrons and protons). If the starting nuclei are relatively light (up to 56Fe and 62Ni), and therefore have a high binding energy per nucleon, this reaction releases energy.
In order to achieve such conditions, the fuel mixture is usually heated up until it reaches the state of plasma, an highly ionised gas characterised by collective effects.
One of the quantities of interest in nuclear reactions is the microscopic cross section σ, that is a measure of the probability of interaction between two nuclei. For the following discussion, it is useful to consider an average over the product of the distributions of cross-section and velocity, obtaining the so-called "reactivity", denoted $\langle \sigma v \rangle$. This quantity depends on temperature and can be used to qualitatively identify the dominant reaction in the range of temperature of interest. Figure 2.1 shows the reactivity of some fusion reactions.
A fundamental equation for any magnetic fusion reactor design is the scaling law:

$$\frac{P_f}{V_p} \propto 8\langle p \rangle^2 \propto \beta_T^2 B_0^4$$

relating the volumetric fusion power density P_f/V_p with the volume-averaged plasma pressure $\langle p \rangle$, the parameter $\beta = (2\mu_0\langle p \rangle)/B_0^2$, relating plasma pressure and magnetic pressure, and the magnetic field B_0. This equations shows that there are two possible paths in order to achieve a high fusion power density: increase β or increase the magnetic field. The second option is often the one chosen for high-field, compact tokamaks,

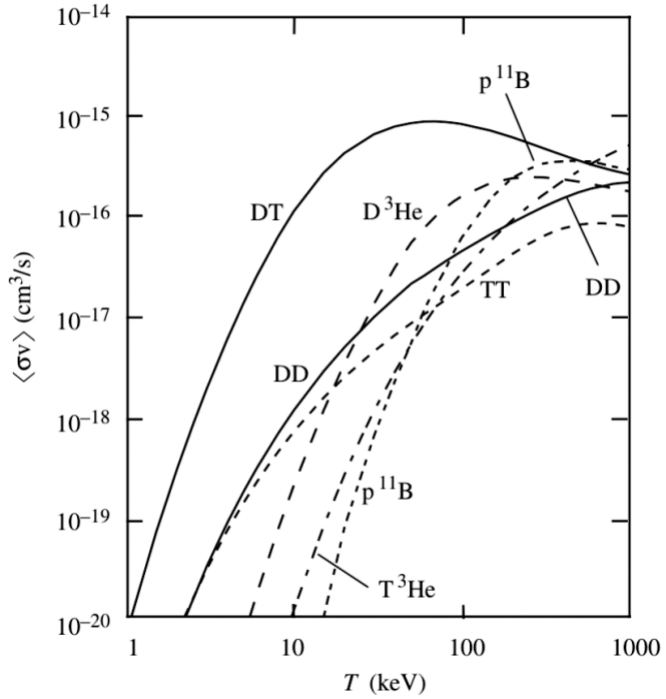


Figure 2.1: Average reaction rate (reactivity) for the principal fusion reactions, as a function of temperature. The temperature is reported in keV ($1\ keV =\sim 11.6e6\ K$) [3].

such as ARC [1]: increasing this parameters usually comes at the risk of exciting MHD modes. The operation in the "second stability region" (at sufficiently high values of β) is often a solution adopted for advanced reactors, where the need for higher pressures and temperatures becomes necessary to approach ignition with advanced fuel mixtures such as Deuterium and Helium-3.
The concept of plasma ignition can be summarised as the point where the plasma temperature can be maintained against the energy losses solely by internal heating, mainly due to the "slowing down" of alpha particles generated during the DT reaction. This condition is difficult to obtain without the initial help of external heating, for this reason another parameter is introduced: the physics gain factor "Q". This parameter is the ratio between the net thermal power coming from the reactor and the total heating power inserted [4]. The condition of Q = 1, that is when the power being released by the fusion reactions equals the heating power, is referred to as breakeven, or better "scientific breakeven", as this does not equate to a net power gain in electricity.

2.1.1 The Deuterium-Tritium fusion reaction

With current technologies, the most feasible reaction is between two heavy nuclei of Hydrogen: Deuterium (D) and Tritium (T):

$$D + T \longrightarrow {}^4He(3.5\ MeV) + n(14.1\ MeV)$$

Therefore each fusion reaction of this kind produces an alpha particle (the Helium nucleus) and a high-energy neutron. The total energy released is 17.6 MeV ($2.8E-12\ J$). Deuterium is a stable isotope, naturally occurring in seawater (0.0312% by weight), which would make it a naturally abundant fuel, with respect to other energy sources. It can be extracted by processing seawater in order to extract the Heavy water (D20) by chemical exchange methods. D2O is then subjected to electrolysis using Hydrogen Sulphide which results in the formation of Deuterium Gas. In addition to this common methodology, other alternate mechanisms include saltwater distillation using Hydrogen and Ammonia or fractional distillation of liquid hydrogen.
Tritium, on the contrary, is naturally present only in trace quantities in the environment: its continuous production in the upper layers of the atmosphere is balanced by the fact that it is radioactive. Tritium decays β^- (emitting an electron and an electron antineutrino) with a half-life of about 12 years. It can be artificially produced in suitable fission reactors (most commonly in heavy water reactors), or can be bred by a reaction between Lithium and neutrons. Lithium has two stable isotopes ${}^6Li$ and ${}^7Li$, with natural abundance of 7.59% and 92.41%, respectively [5]. They are transmuted to Tritium in the following reactions:

$${}^7Li + {}^1n \longrightarrow {}^4He + {}^3T + {}^1n - 2.47\ MeV$$

$${}^7Li + {}^1n \longrightarrow 2{}^3T + 2{}^1n - 10.3\ MeV$$

$${}^6Li + {}^1n \longrightarrow {}^4He + {}^3T + 4.78MeV$$

${}^6Li$ has a higher cross-section in the low-energy region, whereas ${}^7Li$ has a higher one in the high-energy region.
The neutrons produced in a DT fusion reaction can then be exploited to breed Tritium inside the plant. It has to be noted that Tritium could be also produced by Deuterium-Deuterium (DD) fusion reactions [6].
An important quantity needed to define the efficiency of a breeding technique is the TBR (Tritium Breeding Ratio):

$$TBR = \frac{T\ produced\ in\ the\ blanket}{T\ burnt\ in\ the\ plasma}$$

This ratio can be larger than one since the reaction with ${}^7Li$ produces a neutron that can trigger another breeding reaction and can be enhanced by the use of neutron-multiplying materials. This value should be sufficiently larger than one to account for: Tritium decay, losses in the fuel cycle and providing start-up inventory for another reactor.

2.2 Confinement

Fusion can only occur under some specific conditions of immense heat and pressure. In order to achieve such conditions, it is necessary to confine the plasma long enough to achieve a sufficient number of reactions. At the moment, the two most successful confinement techniques are inertial and magnetic confinement. Each of these techniques has been studied and experimented extensively. A few possible configurations will be presented in the following subsections.

2.2.1 Inertial Confinement

The first to propose the idea of achieving the fusion ignition condition by compressing a Hydrogen pellet and using high-power lasers was Nuckolls in 1972 [7]. This is based on the same general principle as that used in the hydrogen bomb: fuel is compressed and heated so quickly that it reaches the conditions for fusion and burns before it has time to escape. The inertia of the fuel keeps it from escaping, hence the name inertial-confinement fusion (ICF). In principle an inertial fusion power plant would have a chamber where these mini-explosions would take place repeatedly in order to produce a steady output of energy. In some ways this would be rather like an automobile engine powered by mini-explosions of gasoline [8].
At the National Ignition Facility (NIF), the 5th of December 2023, a shot produced about 3.15 MJ of fusion energy from the 2.05 MJ of laser light that reached the small cylindrical chamber [10] and on the 30th of July 2023 this record was surpassed, obtaining about a 20% increase in fusion energy. In both cases the yield surpasses the criteria for ignition established by the National Research Council in 2007; taking into account that only about 250 kJ of energy reach the fuel capsule, a value of $Q > 10$ was reached. The technique used at NIF is called "Indirect-Drive" Inertial confinement, as the lasers are not directly pointed on the fuel capsule, but they first hit the inner surfaces of a gas-filled, cylindrical structure that encloses the capsule. As the capsule surface absorbs energy and ablates, pressure accelerates the shell of remaining ablator and fuel inwards. [9].
Even though this is a milestone in fusion technology, taking into account the total power used to power the NIF lasers (about 300 MJ), it is still far from being a viable source of energy.
Inertial confinement fusion is tested not only with the prospect of energy production, but also "to maintain the reliability, security, and safety of the nuclear deterrent without full-scale testing" [11], as it can replicate the conditions relevant to the understanding of the operation of modern nuclear weapons.

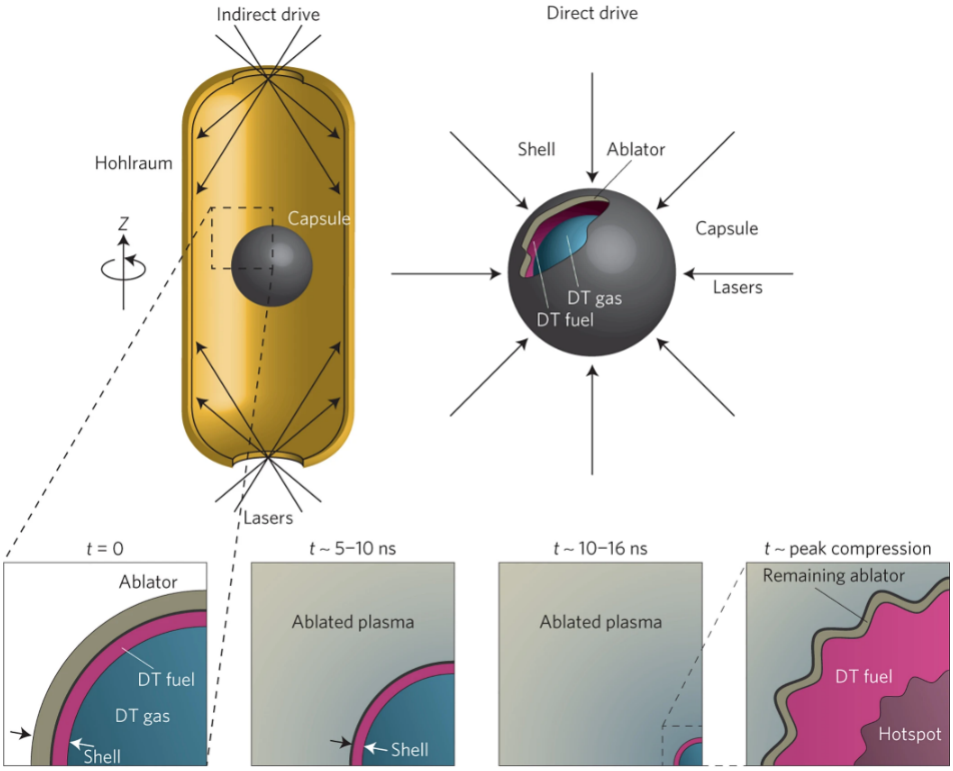


Figure 2.2: Schematic of Indirect-Drive (left) and Direct-Drive (right) Inertial Confinement. [9]

2.2.2 Magnetic Confinement

Another possibility for fusion confinement is the use of powerful magnets that can create a magnetic field acting on the charges inside a plasma. The plasma is kept at low densities and heated to fusion temperature. This idea has been investigated since the 1950s, but two designs appear to be more successful nowadays: the Stellarator and the Tokamak. Both are toroidal configurations, in which the magnetic field is curved around to form a closed loop.
A first design for a Stellarator was proposed by Lyman Spitzer in 1951 [12]. These designs are based on the absence of an internal current; in principle, this can remove some of the instabilities of the Tokamak, so the Stellarator should be more stable at similar operating conditions and work in steady-state operation. On the downside, since it lacks the poloidal field generated by the plasma current, the Stellarator needs stronger magnets to achieve comparable confinement. The complexity linked with its geometry has diverted research on Tokamaks many years, but in the late 1990s new interest arose in Stellarators, as new materials and construction methods, as well as the possibility to use powerful computers to plan the design, have increased the quality and power of the magnets.

The Tokamak design (from a russian acronym, that means "toroidal chamber with magnetic coils"), was designed in 1951 by Soviet physicists Andrei Sakharov and Igor Tamm. The main difference with Stellarators is the use of a central solenoid that works as the primary in a transformer, inducing a current inside the plasma. In turns this currents generates a poloidal field around the toroidal-shaped plasma, ensuring confinement. Its operation is defined by a set of parameters outside which disruptions (sudden losses of confinement) can occur, causing major thermal and mechanical stresses to the walls.

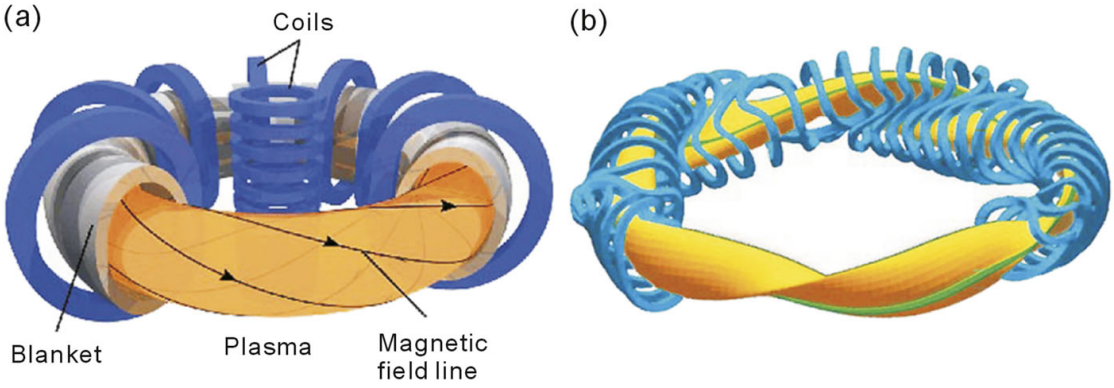


Figure 2.3: Schematics of magnetically confined plasmas in (a) tokamaks; (b) stellarator configurations. In the tokamak, the rotational transform of a helical magnetic field is formed by a toroidal field generated by external coils together with a poloidal field generated by the plasma current. In the stellarator, the twisting field is produced entirely by external non-axisymmetric coils. [13].

2.3 the ITER project

> "At the Geneva Summit Meeting in November 1985 a proposal was made by the Soviet Union to build a next generation tokamak experiment on a collaborative basis involving the world's four major fusion blocks. In October 1986 the United States, in consultation with Japan and the European Community, responded with a proposal on how to implement such an activity. The ensuing discussions between diplomatic and technical representatives of the four prospective participants resulted in the establishment, under the auspices of the IAEA, of the International Thermonuclear Experimental Reactor (ITER) Conceptual Design Activity [14]".

At the end of the 1980s the first designs for an "International Thermonuclear Experimental Reactor" (ITER), with the aim of proving that fusion could be controlled to produce useful energy, were proposed. The final design was approved in 2001 by IAEA and the six partner nations (the European Union, Japan, the United States, Russia, the People's Republic of China and the Republic of Korea) agreed to site it in Cadarache, in the South of France.
Today ITER is funded by seven member parties: China, the European Union, India, Japan, Russia, South Korea and the United States; The United Kingdom and Switzerland participate through Euratom and F4E, and the project has cooperation agreements with 4 more countries: Australia, Canada, Kazakhstan and Thailand.
In 2007 in Cadarache the construction of the "largest and most powerful fusion device in the world [2]" began and initially it was set to become operational in 2019 and later postponed to 2025. Currently the ITER cost and schedule baseline is under review [2], but the most realistic date for the first DT experiment should be 2035.
The goal of this project is to obtain a $Q > 10$ with DT plasma, while testing components and researching plasma physics during its planned 30 years of operation. Of great importance will be the testing of "Tritium breeding module concepts that would lead in a future reactor to Tritium self-sufficiency and to the extraction of high-grade heat and electricity production", as is specifically stated in [14].
The main components of the ITER tokamak are:

- The stainless steel Vacuum Vessel (VV), providing the first containment barrier, removing the nuclear heating and maintaining high quality vacuum inside the plasma chamber.

- The Blanket, shielding the vacuum vessel and the other components from the high-energy neutrons produced during the fusion reactions. Some Breeding Blanket concepts for the production of Tritium will be tested out during ITER operation.

- The Divertor, at the bottom of the VV, allowing for the removal of waste and impurities present in the plasma during operation. It is estimated that this component must endure 20 MW/m^2 heat spikes.

- The Cryostat, surrounding the vacuum vessel and superconducting magnets, ensuring ultra-cool, vacuum environment.
- The Superconducting magnets, needed to confine and shape the plasma.

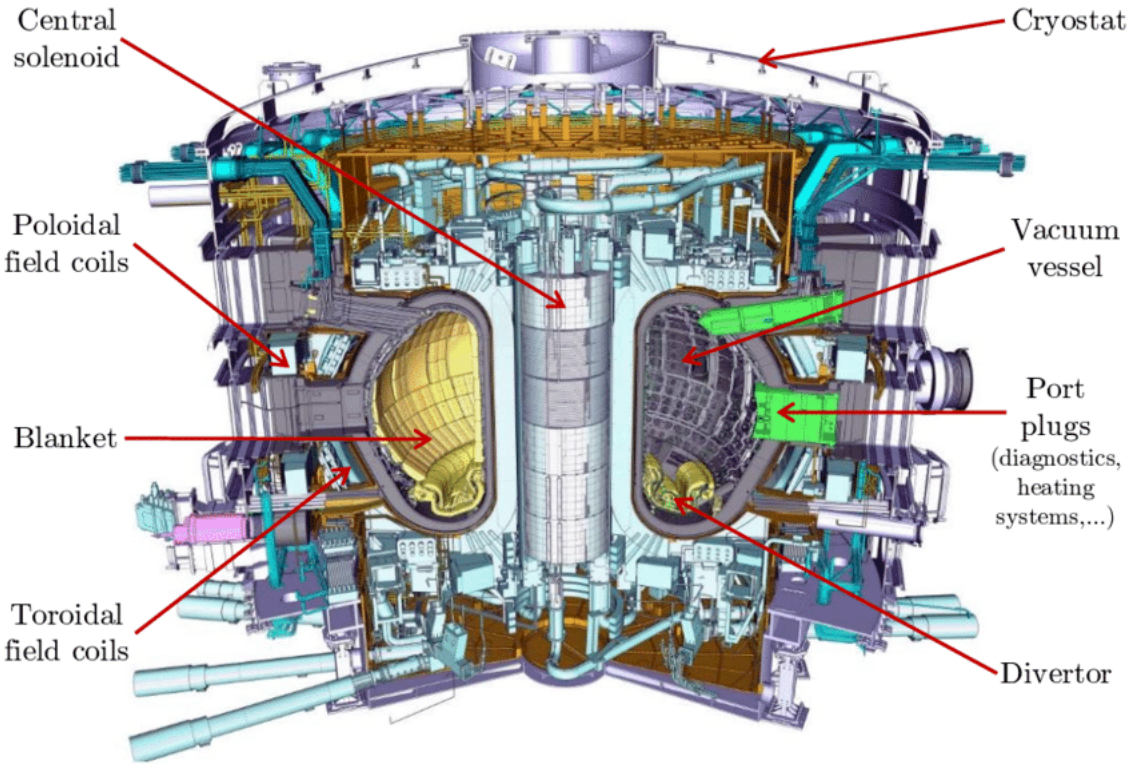


Figure 2.4: Main components of the ITER tokamak[15].

Some interesting parameters for the ITER plant are reported in table 2.1.

2.3.1 ITER magnets

The ITER Magnet system consists of 18 Toroidal Field (TF) coils, a Central Solenoid (CS), six Poloidal Field (PF) coils and 18 Correction Coils (CCs).
The TF coils have the shape of a "D" and provide the toroidal magnetic field that confines the plasma, the PF coils provide the position equilibrium of plasma current, the CCs allow corrections, a current ramp inside the CS induces a magnetic field responsible for the generation of the plasma current, that in turns generates a poloidal field, confining the plasma. The resulting field presents helical field lines; the value of the field on the torus axis is about 5 T and it goes up to 12 T at the coils.
The TF magnets as well as the CS are made of Nb_3Sn-based Cable-In-Conduit Conductors (CICC), wile the PF coils are made of $NbTi$ CICC. While being both low temperature superconductors (i.e having a critical temperature lower than approximately 20 K),

NbTi has a lower critical field and for this reason has been adopted for the PF coils, subject to a lower magnetic field.

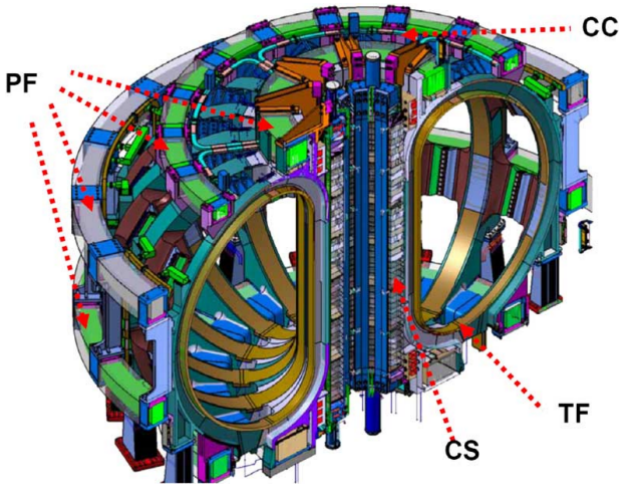


Figure 2.5: A section of the ITER magnets system [16].

2.3.2 Breeding Blanket concepts for ITER

The main functions of a breeding blanket are the Tritium breeding and extraction, removal of the heat load generated by neutron flux, shielding. In ITER the Test Blanket Module (TBM) program will test four different modules simultaneously, positioned in two different ports facing the plasma. Each module will be connected to the cooling system, coolant purification system, tritium extraction system and instrumentation and control system, in order to verify their operation. [17] The modules included in the TBM program are:

- Helium-Cooled Lithium Lead (HCLL) TBS, expected to be proposed by EU;
- Helium-Cooled Pebble-Bed (HCPB) TBS, proposed by EU;
- Water-Cooled Ceramic Breeder (WCCB) TBS, proposed by Japan;
- Helium-Cooled Ceramic Reflector (HCCR) TBS, proposed by Korea;
- Helium-Cooled Ceramic Breeder (HCCB) TBM, proposed by China;
- Lithium-Lead Ceramic Breeder (LLCB) TBS, proposed by India.
- Water-Cooled Lithium Lead (WCLL) TBS, proposed by EU;
- Dual-Coolant (Pb16Li + Helium) Lithium Lead (DCLL), proposed by US.
- Helium-Cooled Solid Breeder (HCSB) TBS, proposed by India (similar to the other HCSB-TBS type listed above).

2.4 the European DEMO project

A Demonstration Fusion Power Plant (DEMO) will be the next step after ITER towards the creation of a commercial fusion reactor. It will be a design capable of producing electricity, operating with a closed fuel-cycle. At the moment the more realistic date for the start of its construction is 2050, since several design options are under investigation by the European consortium Eurofusion.
The main difference from ITER are its larger size and output power; this last aspect is linked with a higher Tritium consumption, for this reason the use of a breeding blanket becomes mandatory. In DEMO The demonstration of fuel self-sufficiency has to be demonstrated.
DEMO will be characterised, in contrast to ITER, by increased pulsed length, possibly going towards steady-state operation. Fully non-inductive steady-state operation, that is plasmas without a transformer-induced plasma current, is researched at the Japanese Tokamak 60 Super Advanced (JT-60SA), built in the context of the Broader Approach Agreement between Europe and Japan [18].
The design and construction of DEMO are closely related to the results of ITER: The most critical and final major validation input from ITER, is the demonstration and operation of D-T burning plasma scenarios with Q =10 (after 2035) and the results of the TBM program (between 2037–2039) [19].
A comparison between some ITER and DEMO parameters is reported in table 2.1:

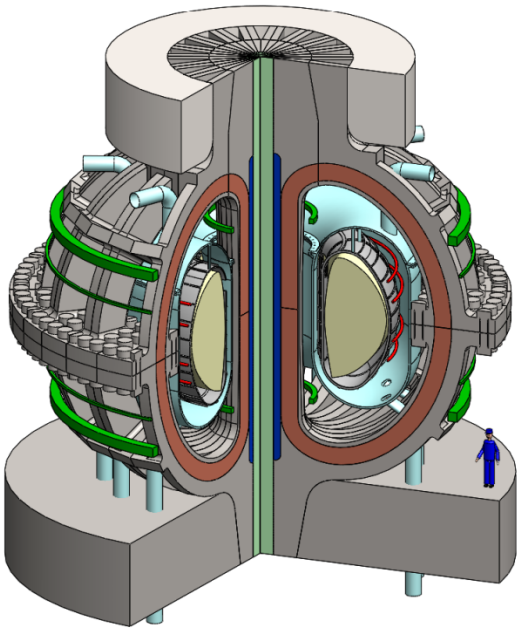
Parameter	**ITER**	**DEMO**
Plasma external radius	6.2 m	8,2 - 13,1 m
Plasma internal radius	2 m	4,1 - 6,6 m
Fusion Power	500 MW	2000 MW
Electric Power	/	1,33 - 1,55 GW
Pulse Duration	3000 s	7200 s
Tritium production	0.5 g/day (one module)	About 400 g/day
Plasma Current	15 MA	17.75 MA
Field on Axis	5.3 T	5.86 T
Irradiation damage	<3 dpa (First Wall)	<100 dpa (First Wall)
Neutron Wall Load	0.78 MW/m$\hat{2}$	2.24 MW/m$\hat{2}$

Table 2.1: EU-DEMO 2018 [20] compared with the ITER Q = 10 scenario [21]

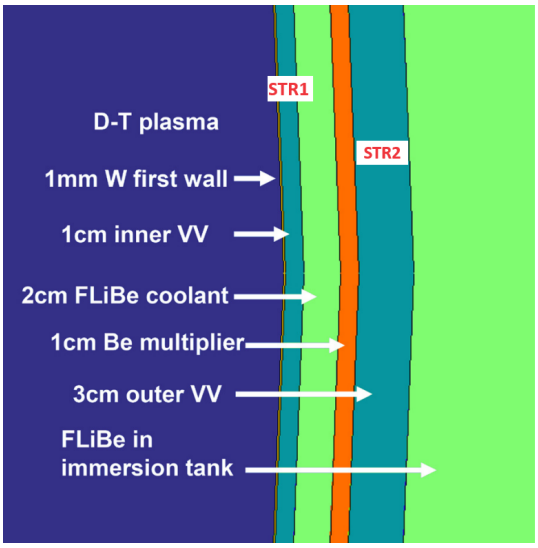
The data reported for DEMO have to be taken as indicative for current design proposals, as a final option has not been selected yet.

2.5 The ARC project

The ARC (Affordable, Robust, Compact) fusion reactor design is a tokamak developed by the researchers at the Plasma Science and Fusion Center (PSFC) of the Massachussets Institute of Technology (MIT), in collaboration with Commonwealth Fusion Systems (CFS) and it will be the successor of SPARC (as Soon as Possible ARC), a smaller, experimental reactor, currently under construction and set to begin operation in 2025. The main objective of the ARC reactor is to obtain a Q>10, and producing three times the electricity required to operate it.
The main difference with ITER and DEMO is the extremely reduced dimensions (about $1/8^{th}$ of ITER), reducing the construction costs considerably. This can be done by using High Temperature Superconductors (HTS) instead of traditional low temperature ones, as is the case in ITER. In small tokamaks, such as ARC, it is necessary to increase magnetic fields in order to maintain sufficient confinement and stability.
Another peculiarity of this design is the implementation of a FLiBe molten salt blanket, that will act not only as a shield for neutrons and as a heat removal system for the vacuum vessel, but also as Tritium breeder.



(a) The ARC reactor, shown with the plasma in yellow and the TF superconducting tape in brown. The neutron shield and divertor are omitted from this representation.[1]



(b) Radial build of the double wall vacuum vessel. A 1 cm thick Be layer is placed on the surface of the outer VV layer to function as a neutron multiplier [22]. The first structural layer has been denoted as "STR1", while the second as "STR2".

2.5.1 High Temperature Superconductors for ARC

ARC will use REBCO (Rare-Earth Barium Copper Oxide) High-Temperature Superconducting (HTS) tapes, capable of remaining in superconducting conditions at far higher temperatures with respect to traditional Low-Temperature Superconducting (LTS) materials. Furthermore they can also sustain higher current densities and external magnetic fields.
The use of HTS gives the possibility to both increase the operating magnetic field and build a more compact machine. These two aspects can solve numerous issues, such as increase the power density (scaling as B^4) and increase the plasma density (scaling as $R^{-1}B^1$) [23].
Since REBCO come in the shape of tapes, where the superconducting material is a micrometers-thick layer, it is necessary to design new cable geometries to accommodate them. A selection of designs of REBCO tapes-based cables is reported in figure 2.7. The main drawbacks related to HTS technology is its relative underdevelopment with respect to LTS, and there are still issues related to costs, mechanics, quench and anisotropy.

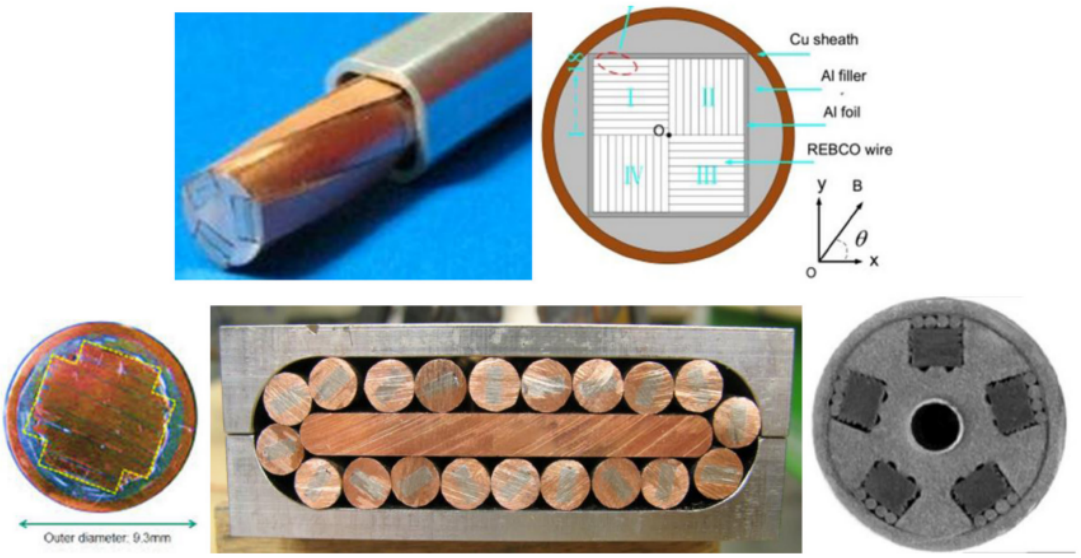


Figure 2.7: Clockwise from top left: the MIT concept, the quasi-isotropic strand, the ENEA five stack cable, the 60 kA, 12 T prototype of SPC and the staggered twisted stack of KIT. [22].

In order to obtain 9.2 T on the axis of ARC, the maximum magnetic field in the conductors in the TF coils will be 23 T, at this maximum value of the magnetic field and 4.2 K, the critical current density of REBCO tape superconductors produced by SuperPower Inc.[24] is between one and two orders of magnitude higher than Nb3Sn, making REBCO the perfect superconductor for ARC TF coils [1].
The ARC TF coils will be demountable and modular, with an expected life of operation

of a few years.

2.5.2 Liquid Breeding Blanket concept for ARC

Several designs of liquid breeding blanket have been proposed. This type of designs generally have the advantage that cannot be damaged, as the blanket is continuously "refreshed" with new material, can work simultaneously as coolant, neutron multiplier, breeder and can carry Tritium to be extracted. ARC will use F_4Li_2Be, a molten salt made from a eutectic mixture of lithium fluoride (LiF) and beryllium fluoride (BeF2). The most probable reactions when invested by a neutron flux are:

$$^6Li + n \longrightarrow {}^4He + {}^3T + 4.8\ MeV$$

$$^7Li + n \longrightarrow {}^4He + {}^3T + n - 2.5\ MeV$$

$$^9Be + n \longrightarrow {}^8Be + 2n$$

$$^{19}F + n \longrightarrow {}^{20}F$$

So it can act both as a breeder and neutron multiplier, to further increase the TBR, but also as coolant and shield for the superconducting magnets.
The presence of a liquid breeder simplifies the magnets disassembling during maintenance and maximises the breeding volume.
It presents reduced activation issues with respect to other breeders, however the presence of Beryllium raises concerns in terms of chemical hazards and reactor cost-effectiveness [25]. Furthermore FLiBe can cause material corrosion issues that cannot be neglected. In ARC, FLiBe is contained in a large low-pressure tank and slowly flows inside the vacuum vessel, inside a double wall compartment. FliBe will enter from the top of the reactor, at 800 K, in order to stay at a safe margin from its freezing point (459°). Having reached the bottom of the reactor, it will enter the surrounding FLiBe tank and moves towards the by natural circulation. Once reaching the top of the reactor again, a pump will extract the hot FLiBe and send it to the heat exchanger. This design is extremely flexible to various ranges of heat load, since in low thermal loads conditions the flow in excess can be directly pumped to the FLiBe tank [22].
Furthermore FLiBe appears to have low neutron-induced activation, that would permit its recycling [26].
A schematic of this system is shown in figure 2.8.

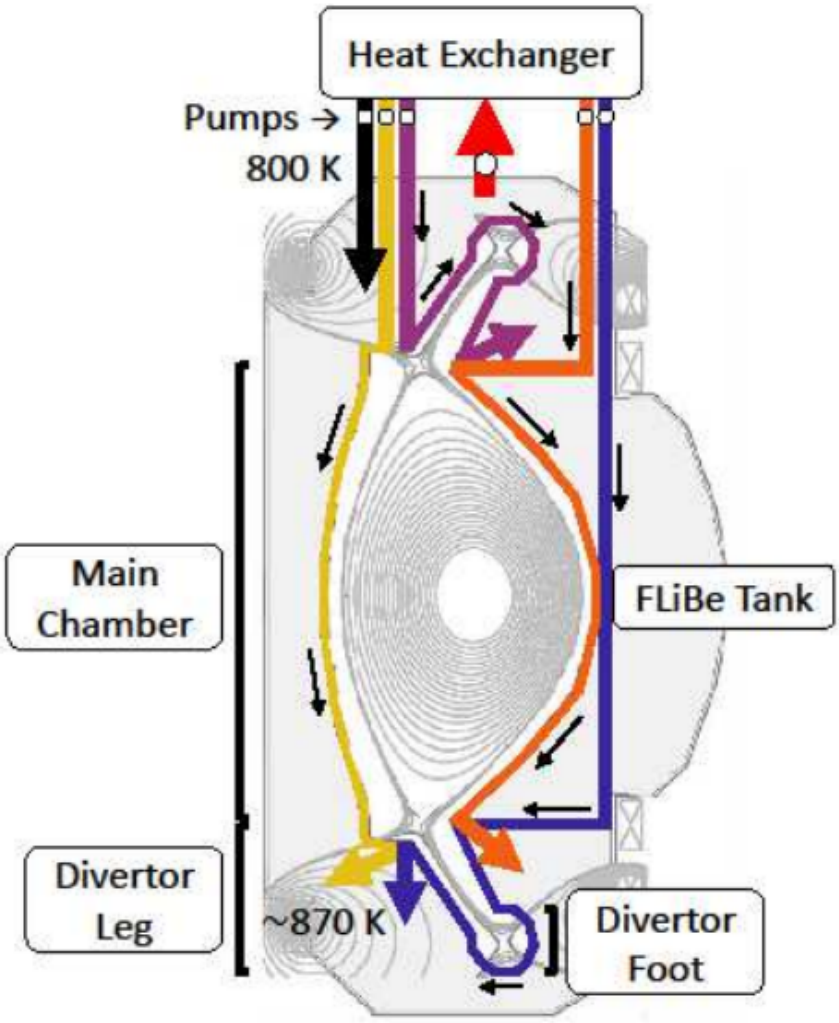


Figure 2.8: Schematic of the VV geometry and pumping system. External pumps are represented by white circles. Six representative FLiBe flows are shown with arrows [22].

Chapter 3

Advanced fusion fuels

At the present time, DT fusion is the most investigated solution to achieve breakeven. Despite the easier working conditions required for a DT mixture, the presence of Tritium complicates the engineering design of the reactor. Furthermore, the large neutron flux coming from the DT reaction poses serious challenges to the vacuum vessel materials, and a considerable shielding is required for the superconducting magnets.
Moreover, since Tritium is a radioactive material, it is necessary to ensure safe transport and flow inside the plant. Its relatively short half-life, in addition to its scarce presence in nature and low artificial production rate, makes the development of a breeding blanket technology for the startup of future plants a crucial step towards the development of fusion power.
The breeding blanket adds complexity to the reactor, a Tritium extraction system must be present and the excess Tritium must be stored for future use. Finally, the Vacuum Vessel materials are subjected to neutron irradiation, causing activation of the materials. If the levels of induced radioactivity are too high, they must be handled as radioactive waste.
The aim of this chapter is to provide a comprehensive summary of all these issues, and present some alternative approaches.

3.1 Tritium supply assessment

As previously outlined in chapter 2.1.1, Tritium is a rare isotope of Hydrogen. It is a low-energy beta emitter (maximum energy to the electron is $\sim 18.6keV$) with a half-life of 12.31 years. Having the same physicochemical form of Hydrogen, and despite the difference in mass, it easily bonds with other Hydrogen atoms to form tritiated hydrogen (HT, in gaseous form), but its most common form is tritiated water (HTO); it is also present bonded to organic matter (organically bound Tritium, or OBT). It is naturally and continuously produced in the upper layers of the atmosphere by its interaction with cosmic rays, but is also the dominant radionuclide in releases from nuclear plants [27]. It is also used for diagnostic and therapeutic purposes, in research laboratories and in some industrial applications. Tritium is also used for military applications, but at the

moment is not included in the IAEA Safeguard: the risk of diversion towards non-civil uses is a major concern, and must be studied in depth.

3.1.1 Tritium production and demand

At present, Tritium is mostly produced inside fission reactors using heavy water as moderator, coolant or both, due to neutron capture by deuterium. The most efficient for this aspect are by far CANDU 6 (CANadian Deuterium Uranium) reactors, that in principle can produce 130g of Tritium per year [28], but this was based on physics, not production data.
In order to produce and commercialise Tritium it is necessary to use a Tririum Removal Facility (TRF), of which only two are currently in operation, one in Canada and one in South Korea [29], while a third one is set to be constructed in Romania [30]. It is not clear if in the future new TRF will be built and if future CANDU supply will equal or exceed present production, as the current reactors age and reach their end-of-life. China produces a small quantity of Tritium as oxide at Qinshan and is exploring alternate methods of commercial Tritium production, and India produces a significant quantity of Tritium in its heavy water reactors, but has no commercial TRF[31].
Currently the total available Tritium stockpile is estimated to be about 25 kg and its cost is about $30000 per gram.
The most widespread types of reactors, Pressurised Water Reactors (PWR) and Boiling Water Reactors (BWR), both light water reactors, are incapable of producing significant amount of Tritium without the use of Tritium Producing Burnable Absorber Rods (TPBARs), but these methods are often employed in nuclear weapons manufacturing and may be considered unattractive as a Tritium-producing method, given this association [31]. The risk of proliferation must be taken into account for all extraction facilities nonetheless.
Another possibility to allow the startup of future DT fusion reactors, without the need of Tritium, has been proposed by Zheng et Al. [6]. Initially the reactor would burn DD fuel: this would inevitably cause a reduction in power production, but with the ongoing reactions would produce Tritium that will be burn in the subsequent operations. Two possible reactions, with equal probability can initially take place:

$$D + D \longrightarrow T + p$$

$$D + D \longrightarrow {}^3He + n$$

The first will directly produce Tritium, the second will produce it by the interaction of the neutrons with a breeding blanket. In principle this approach may cancel Tritium consumption from CANDU stockpiles, while simultaneously reducing the neutron irradiation during the DD phase, but M. Kovari et Al.[32] argue that this approach can result in extremely high costs, due to the long period of low power production and long times needed to build a sufficiently large Tritium supply.
Numerous studies have been conducted in order to see if future demand of Tritium for future power plants will be satisfied by its production, taking into account the previously

described difficulties, M.Kovari et Al. [32], in 2018, consider three possible scenarios, denoted as: "A" - pessimistic, "B" - moderate and "C" - optimistic. In all three cases the eventual starting of operation of the Chinese Fusion Engineering Test Reactor (CFETR) has been considered; this reactor would demonstrate steady state operation and tritium self-sufficiency and, in a second phase, generate power of over 1 GW. The results on Tritium supply assessment by M.Kovari et Al. are summarised in figure 3.1. A similar study has been conducted by R.J. Pearson et Al. [32], arriving at similar conclusions: the Tritium required for ITER will be supplied by the Canadian CANDU production, despite its delay in schedule. The possible start of new fusion experiments further increases the uncertainty on Tritium demand. The availability of Tritium for future fusion reactors heavily depends on the success of breeding technologies, increased production and construction of new TRF, improvement in fuel efficiency and burn-up in order to minimise Tritium inventory, or the use of low Tritium start-up techniques.

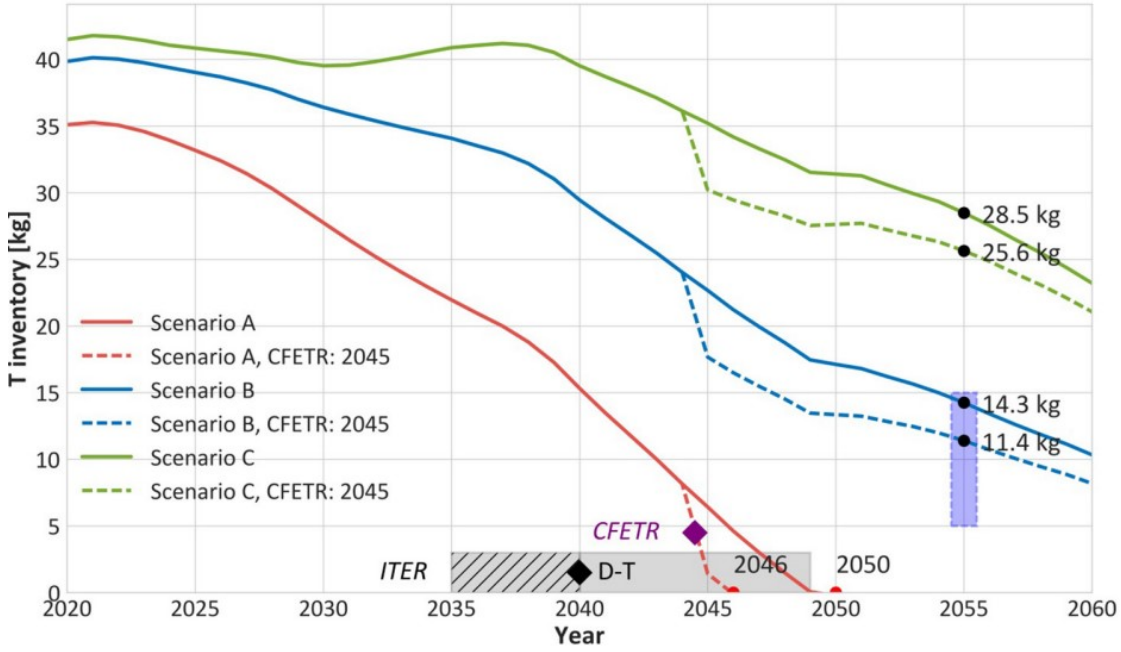


Figure 3.1: Total Tritium stockpile in Canada, Romania, and Republic of Korea, with ITER D–T operations commencing in 2040. CFETR consumption is assumed to take place five years later, in 2045. From [32]

3.2 The Deuterium Helium-3 fusion reaction

As described in the previous chapters, the use of Tritium is not only a problem from the engineering point of view, but it is also linked to a radiological hazard (a Tritium inventory of the order of kilos translates to tens of MCi of activity). Furthermore the DT reaction produces a large amount of high-energy neutrons that cause activation in the materials making up the reactor.
A possible alternative is the use of the Deuterium-Helium3 (DHe3) fusion reaction:

$$D + {}^3He \longrightarrow {}^4He(3.6\ MeV) + p(14.7\ MeV)$$

But inside a DHe3 plasma the two other DD reactions can also happen (both having same probability):

$$D + D \longrightarrow T(1.01\ MeV) + p(3.02\ MeV)$$
$$D + D \longrightarrow {}^3He(0.82\ MeV) + n(2.45\ MeV),$$

Furthermore, the production of Tritium nuclei from a DD reaction, also DT reaction would occur:

$$D + T \longrightarrow {}^4He(3.5\ MeV) + n(14.1\ MeV)$$

It has to be noted that also a ${}^3He - {}^3He$ reaction is possible, producing two protons and an alpha particle. Since it has a much lower cross-section it won't be considered in the present discussion.
A reactor using a 50% Deuterium, 50% Helium3 fuel mixture would therefore have 4 different major reactions happening inside it, but the Deuterium-Helium3 reaction would remain the dominant one. Considering a He3-rich fuel mixture, a reactor using this fuel mixture would have an almost aneutronic fuel cycle [33][34]. The production of a large amount of protons from the DHe3 reactions (but also He3He3 reactions) can potentially be exploited for the direct production of electricity, with a high efficiency [35] [36].
The reasons why this type of fuel cycle has seldom be taken into consideration are mainly two: the scarcity of Helium3 on Earth and the lower reactivity with respect to the DT reaction. For this last reason, a DHe3 reactor will need to operate at substantially higher temperatures and densities. In order for a D-He3 plasma to reach ignition, and reaching a $Q \sim 10$ in order to maintain the rate of reaction, it is necessary to reach a temperature of 50 keV.
These two reasons, that diverted attention from DHe3 reactors for a long time, could be overcome in the next decades, as the ARTEMIS program expresses the will to set an outpost on the moon [37] and recent development in High Temperature Superconductors allows to reach higher magnetic fields.

3.2.1 Lunar Mining of Helium-3

> "Extrapolation of the Apollo 11 sample data by remote sensing indicates that the 84,000 square kilometers (53,000 sq. mi.) of the highest grade regolith on Mare Tranquillitatis contains at least 5000 tonnes (5500 tons) of recoverable helium-3[38] [39]. That amount would provide 50 years supply (assumed plant life) for 100, 1000-megawatt helium-3 fusion power plants on Earth. Near the lunar poles, 84,000 square kilometers (53,000 sq. mi.) may supply three times the above number of power plants. Future direct remote sensing and/or sample data from high latitude regions of the Moon may positively influence the calculation of inferred helium-3 reserves as well as the production costs. Where deep permanent shadow exists, helium-3 also may be contained in clathrates and non-lunar fullerenes."[40]

Helium-3 was created through the interaction between lunar soil and solar winds over billion of years, and can be found in abundance on the surface and up to a depth of a few meters.
The cost of Helium-3 from lunar mining activities was estimated to be about 1 billion dollars per ton [34] in 1987, but with further advancement in mining technologies, such as the MARK IV miner [41], and improvements in Earth-Moon transportation, that could arise from the ARTEMIS program [37], this figure could be significantly lowered.
An assessment on the cost to build a DHe3 reactor, taking into account the cost of components as well as the price of fuel, has been carried out by S. Meschini et Al. [42]. They conclude that, despite the larger size, the design simplifications related to the D-He3 fuel lead to a cheaper reactor and to lower cost of electricity. The design simplification considered in the study are: the removal of the breeding blanket as well as the entire Tritium fuel-cycle-related components, since the use and the breeding of Tritium are no longer necessary; The simplification of neutron shielding, due to the lower neutron flux and overall lower energy, and the components could be designed with a longer lifetime; The Balance Of Plant (BOP) could be simplified, considering only the direct conversion channel.
The impact of Helium-3 price seems to have a low impact on the whole reactor economics, up to tens of million dollar per kilogram.
Despite the necessary simplifications and scaling involved in the study, due to the economic analysis of technologies still in development, it shows how, if the suitable conditions are met, DHe3 reactors would not only be better from a waste management viewpoint, but from the economical one too. The eventual construction of Dhe3 reactors on Earth would at the same time create the demand of He3 to further develop lunar mining.

Chapter 4

Deuterium-Helium 3 Tokamaks

Several DHe3 reactor designs have been developed during the 1980s and 1990s.
The Apollo series of reactors, developed by the the Fusion Technology Institute of the University of Wisconsin, highlighted the reduced presence of neutrons that this type of reaction would generate, along with the possible reduced cost of electricity [62]. The latest design of this series, Apollo L-3 would operate in the first stability regime and exploit both direct and thermal conversion [63].
Another DHe3 fusion reactor, ARIES-III, developed on the ARIES trend, would instead operate in the second stability region [64]. The ARIES program performs conceptual design studies for future fusion power plants, while the ARIES-III design in particular focuses on a compact, DHe3 fueled tokamak. F. Najmabadi et al. [61] compared ARIES-III with the other ARIES projects based on DT plasmas and highlighted the attractive engineering features that an operation with DHe3 would generate, provided that the demanding physics conditions are met.
Another conceptual high-density, high-field and compact fusion reactor was proposed by Prof. B. Coppi in 1975: IGNITOR, part of a line of research that began with the Alcator machine at MIT in the 1970's. IGNITOR was initially set to be built in Italy, the chosen site was then moved to Russia, but the construction has not started yet [66]. The main goals of the IGNITOR experiment are [67]:

- The study of the containment of alpha particles and the consequent plasma heating, with the final aim of reaching ignition.
- The investigation of collective modes and transport processes characterising DT fusion plasmas.
- Obtain fusion burning and ignition at relatively low temperatures (less than 15 keV), avoiding the need of reliance on an external heating system.
- Study Ion Cyclotron Resonance Frequency (ICRF) heating system.
- Test diagnostic systems and study new methods for plasma control, heating and fueling.

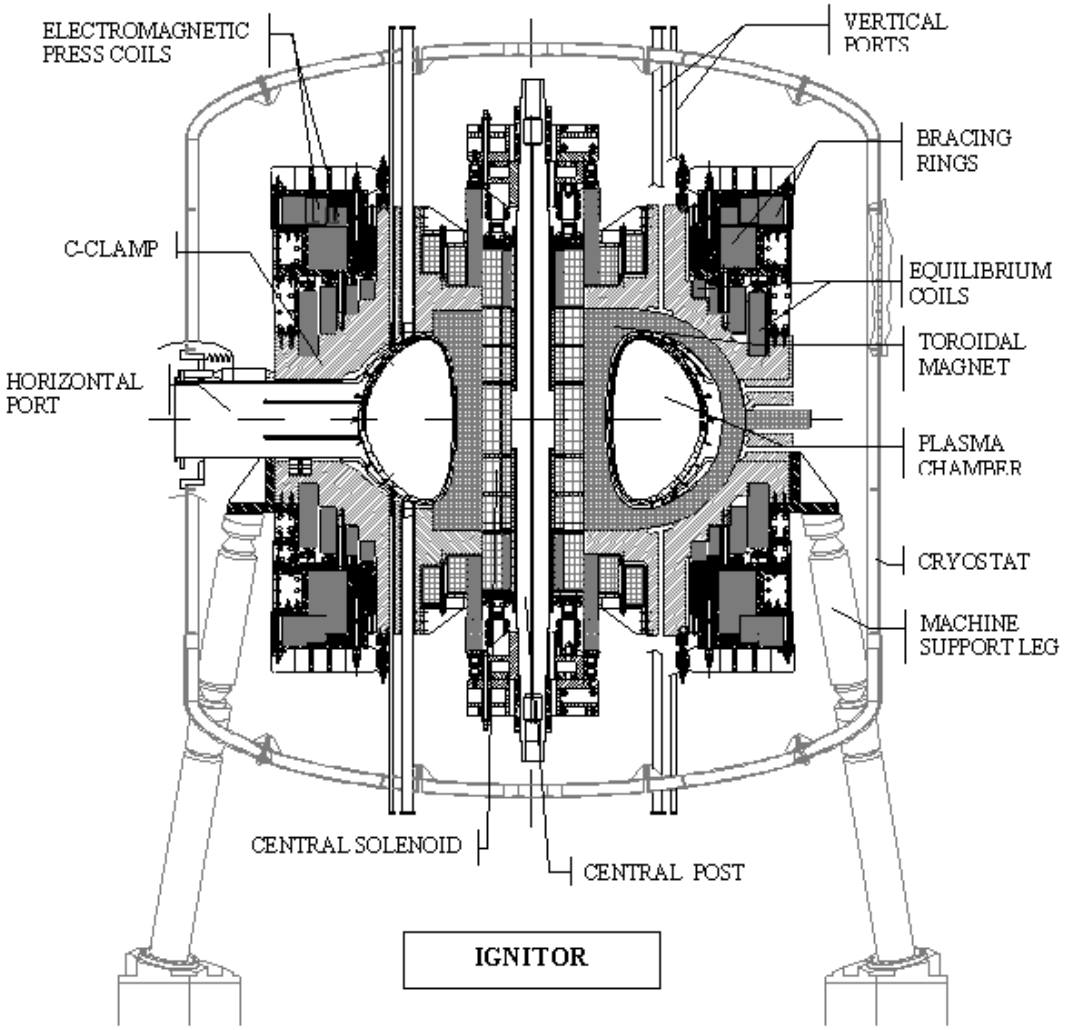


Figure 4.1: Vertical cross section of the Ignitor machine. [68]

- Explore subignited (Q<5) plasmas.
- Study the behaviour of the plasma in the second stability region (high β), by operation at low field and plasma current, with aid of the ICRF heating.
- Test operation with D-He3 fuel mixture, thanks to the addition of a radio-frequency heating systems to the design.

A design evolution of IGNITOR, designed to burn DHe3, called CANDOR has been later proposed [69] [70]. Being larger and more powerful, it should allow the core of the plasma to operate in the second stability region. In order to have a larger magnetic

field, the magnets disposition is different from IGNITOR. There would be two set of toroidal field coils, one internal and one external, and the central solenoid would be placed between them in the inboard part.

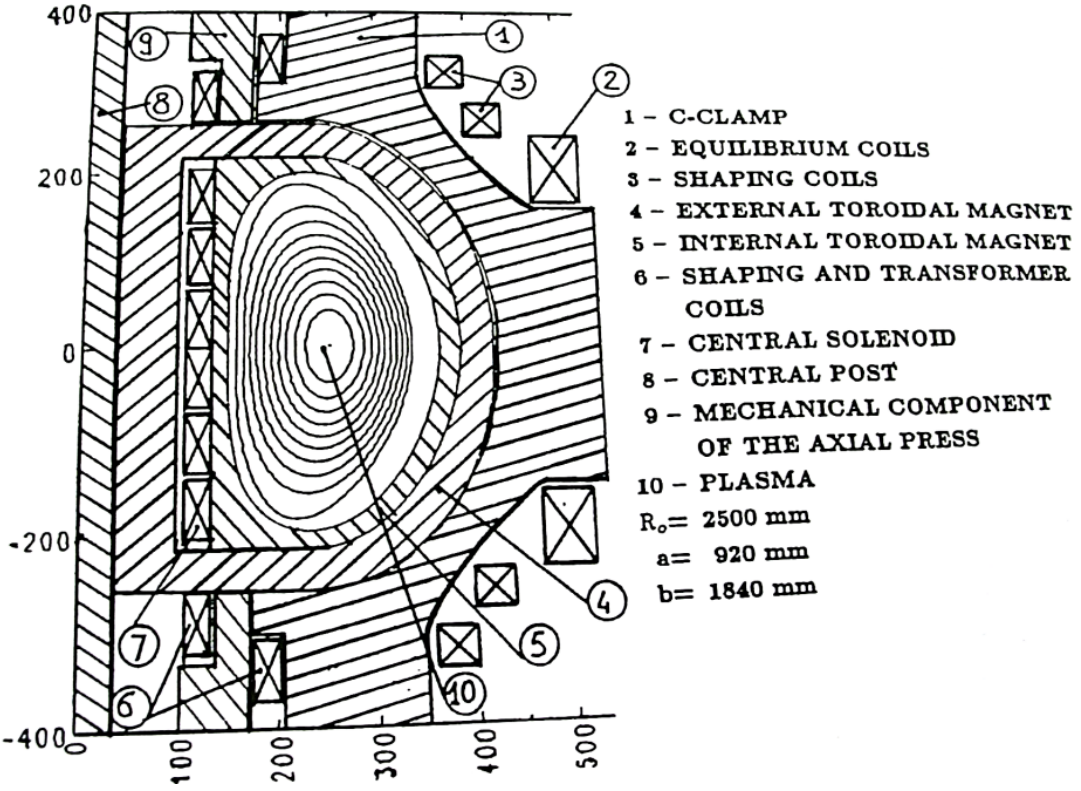


Figure 4.2: Vertical cross section of the CANDOR machine.

This configuration would allow the reduction of the current density in each coil, while reaching the necessary parameters to sustain the reaction. Assuming the same operating temperature of IGNITOR, CANDOR should be able to maintain the plasma for longer period of times, up to 16 s.

4.1 The "Coppi's match" and beyond

In CANDOR an initial operation with Deuterium and Tritium would acts as a trigger. the resulting alpha power heating, combined with additional ICRF (Ion Cyclotron Radio Frequency) heating take the plasma to the necessary temperature conditions [67]. In this case the initial mix would be 50%D and 50%T, but then T would not be reinserted in the cycle, but progressively replaced with He3.
Prof. B. Coppi [56] described the steps necessary to reach DHe3 burning conditions:

- Ohmic heating of DT plasma with the highest current that is compatible with the

stability of the plasma column;

- Ion cyclotron heating, with He3 as a minority species, up to ignition conditions for the adopted DT mixture;
- DT burning with raising peak plasma temperature. During this phase He3 is injected into the plasma chamber while Tritium is depleted;
- DHe3 burning when the peak temperature achieves the maximum values that are compatible with the macroscopic equilibrium and stability conditions of the plasma column. In this phase Tritium should be almost completely burned out.

This method requires the use of Tritium nonetheless, bringing with it the complications described in the previous chapters. In principle the use of Tritium could be entirely avoided by adopting suitable external heating techniques, such as Ion Cyclotron heating, but it has been estimated that it would be uneconomic for this reactor design.
Since CANDOR was designed to reach ignition with the technology available at the end of the 1990s, the recent developments in high-temperature superconductors and additional heating techniques can reduce significantly the use of the "Coppi's match" and further reduce the use of Tritium and the production of neutrons.
In a pure DHe3 mixture (50% D and 50% He) the power coming from neutrons has been estimated to be at about 1% of the total, but this fraction could be further reduced by injecting Deuterium atoms with spin polarized in the direction of the magnetic field [56]. If the polarization can be maintained during the operation, the DD reactivity is almost negligible. If the spin of He3 nuclei are parallel to the Deuterium atoms, the reaction rate between Deuterium and Helium-3 could be further increased by a factor 3/2.

Chapter 5

Analysis of neutron-induced radioactivity in a compact fusion reactor

The public acceptance of fusion is inextricably linked to the promise of a cleaner energy source. As a consequence it is necessary to reduce as much as possible the radioactive inventory of the fuel cycle. The use of advanced fuels that lower the radioactive fuel inventory (i.e. Tritium) as well as material activation is an optimal choice to obtain this result.
The aim of this chapter is to briefly present the strategies currently employed in the fusion field to minimise the nuclear waste output and analyse the possibility of further reducing it by the use of advanced fuels, such as DHe3. An analysis of the neutron-induced activation of the first structural layer of the ARC reactor (STR 1 in figure 2.6b) will then be performed, using FISPACT-II, and compared with the activation produced by a lower neutron flux. Particular attention will be devoted to the possible operation with DHe3 fuel mixture.

5.1 The zero-waste option

Neutron-induced activation occurs when atomic nuclei capture free neutrons, increasing their mass number and entering excited states. This causes them to become unstable and emit radiation: being gamma, beta, alpha or through fission, that is not of interest in this context. Some nuclei require more than one added neutron to become radioactive (for example Hydrogen, needing two neutrons to become Tritium). In addition all in-vessel components of fusion reactors will be contaminated with tritium as well as becoming activated.
In the case of a DT fusion plant, alongside the optimisation of the Tritium fuel cycle, low-activation materials are a widely studied solution for the structural materials that suffer the largest neutronic load. The use of low activation materials for the first wall

and structural components is crucial in order to attain safety, environmental sound and economical advantages in fusion power plants. Such materials are composed of a mix of low-activation elements (Fe,Cr,V,Ti,W,Si,C,Ta); the most promising at the moment are: reduced activation ferritic/martensitic (RAFM) steels, vanadium alloys, tungsten-based materials and composite materials such as SiC/SiC. They all exhibit reduced activation, but present specific issues related to radiation damage or compatibility (for example corrosion problems arising from the use of incompatible coolants). The use of such materials is therefore only a part of the solution: in order to obtain the minimum possible amount of nuclear waste, it is necessary to reduce the neutron flux in the first place.
It is of particular interest to investigate solutions that can minimise the use of final repositories. The strategy proposed in [43] is based on two principles:

- Conditional recycling: the re-use of the activated materials from in-vessel zones in new reactors.
- Clearance, the declassification to non-active waste of activated materials from ex-vessel zones.

The IAEA Clearance Limits will be considered in the context of this study.If the analysed material is composed of multiple nuclides, the Clearance Index $CI = I_c$ can be computed as:

$$I_c = \sum_{i=1}^{z} \frac{A_i}{L_i}$$

Where A_i is the specific activity and L_i is the clearance level of each of the z nuclides. The material can be cleared if $I_c \leq 1$.
However recycling is not only linked to a radiological evaluation of the material, but also to the economical feasibility of the process in the first place. Nonetheless a material classified as "recyclable" from the radiological viewpoint could be disposed as a Low Level Waste (LLW), reducing the hazard in transport and lowering the costs and environmental impact of disposal.
For this last reason it seems plausible that a zero-waste option for DT reactors will not be available [44]. The use of advanced fuel cycles, together with a suitable choice of materials, may be necessary to achieve a real "zero waste option".
Table 5.1 summarises different possible recycling routes and handling techniques [45] [46]. The routes define the processes to treat the activated (and Tritium contaminated) materials; three main routes have been defined: clearance, recycling in foundries and other techniques that still have to be defined/developed.
The clearance category is further subdivided in unconditional clearance, for which the CI must be lower than 1, and conditional clearance, that depends on local regulations. Conditional clearance allows the recycling or reuse of the material in a specified application and subject to continuing regulatory control until specific conditions are met to allow unconditional clearance .
For the recycle in foundries a limit of 1000 Bq/g is taken as reference, while for the other

Limit	< 10 μSv/h	< 2 mSv/h	< 2000 W/m^3
Handling	HOH	SHOH	RH
Routes	Clearance	Recycle in foundries	Process to define
Limit	CI<1	< 1000 Bq/g	< 2000 W/m^3

Table 5.1: EU management routes for fusion radioactive materials [46]

recycling possibilities, the limit is dictated by the decay heat [46].
In particular, the following recycling categories were proposed, based on handling [46]:

- HOH (Hands-On Handling). Contact dose rate (DR) <10 μSv/h.
- S-HOH (Shielded Hands-On Handling). Contact DR < 2 mSv/h.
- RH (Remote Handling). Contact DR >2mGy/h, it can be dealt with by remote handling equipment, without active cooling: decay heat is <2000 W/m3.
- ACM (Active Cooling Material). This requires active cooling and it is unlikely that any recycling operations can be performed until its decay heat decreases to levels not requiring active cooling, hence interim storage with cooling is the only option available.

The ultimate goal of this study is reaching the ambitious "zero-waste" goal. Recycling has to be an important option for those components closer to the plasma: these routes will be taken into full account too.

5.2 FISPACT II software setup

Fispact-II is an inventory code capable of performing modelling of activation, transmutations and depletion induced by neutron, proton, alpha, deuteron or gamma particles incident on matter [47]. For FISPACT-II, the main input is the composition of the analyzed material, the neutron flux and energy spectrum and the irradiation time. Results from FISPACT-II are the time behavior of specific activity and dose rate, as well as the Clearance index.
It is important to note that FISPACT-II calculates the buildup of Tritium during irradiation, but doesn't take into account the absorption of Tritium used as fuel or the one produced in the breeder. Therefore we assume that complete detritiation is performed at the end of life of the component.
For what concerns the material, Inconel-718 will be considered, a Ni-based alloy and, as such, prone to nuclear activation. Its density at room temperature is 8.19 g/cm3. Its element composition is shown in table 5.2.
This material composes most of the Vacuum Vessel of ARC [1]. The most critical region of the VV for what concerns neutron-induced radioactivity, is therefore the first structural layer (STR1), shown in figure 2.6b. This layer has a total volume of 3,529,943.32 cm^3.

Element composition	Al	C	Co	Cr	Cu	Fe	Mo	Ti	Nb	Ni
Weight %	0.52	0.021	0.11	19.06	0-02	18.15	3.04	0.93	5.08	53.0

Table 5.2: Inconel-718 chemical composition by $wt\%$ [48]

The neutronic spectrum on the first structural layer (STR1) was obtained with Monte Carlo simulations using OpenMC; the total flux value is: $3.69660343e+14$ $\frac{neutrons}{cm^2 \cdot s}$.

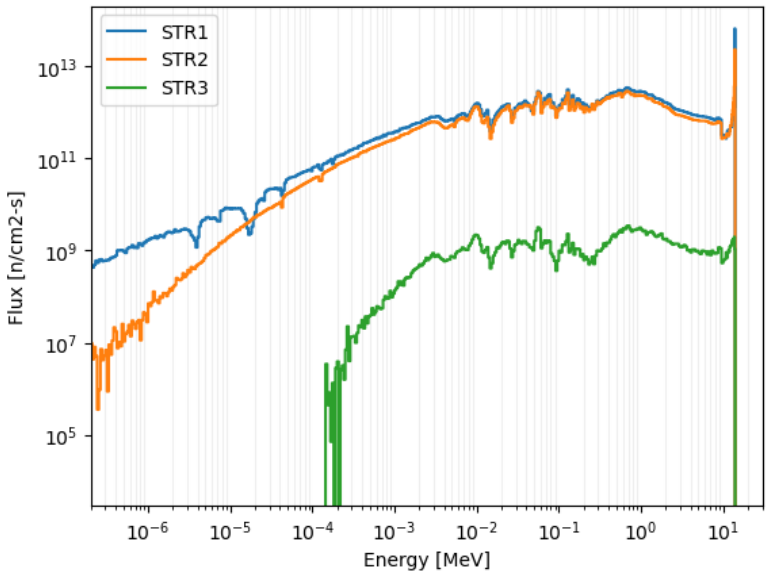


Figure 5.1: Neutron spectra on different structural layers of ARC. First layer (STR 1) in blue, second layer (STR 2) in orange and third layer (STR 3) in green.

All subsequent calculations were performed with the TENDL-2017 nuclear data library, developed at PSI and IAEA. The data post-processing was performed using MatLab.

5.3 ARC materials activation

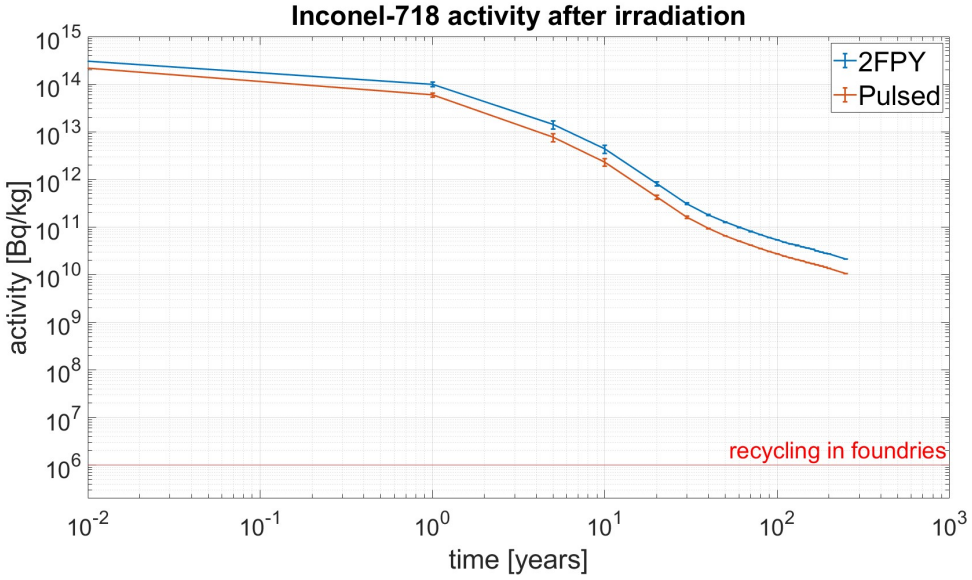
The expected lifetime of the VV of ARC is 2 years [1], due to the damage induced by the neutron flux. A first calculation was done considering a worst case scenario: two full-power years (FPY) of operation. A second calculation considering a low fluence pulsed operation with 6 months long discharges alternated with switched-off periods of 6 months, for a total of 1 year in operation and 1 year dwelling time. It must be noted that such operation would halve the expected lifetime of the component, with a corresponding increase in costs. Such operation has been considered only to quantify the reduction in activation with irradiation time.
The annual dose limit for workers in the nuclear field in the US is 20 mSv/y, this, assuming 8 working hours per day and 5 days per week (constant contact with the

material for the entire working time), equates to a dose rate of 1e-5 Sv/h. This value was then chosen as reference for recycling in the nuclear industry. A dose rate of 1e-6 Sv/h was chosen as reference for recycling in other industries, corresponding to the 1 mSv/y regulatory limit for non-nuclear workers, assuming once again constant contact time during a working week. A dose rate of 1.14e-7 Sv/h was also added to compare the results with the natural background average dose (1 mSv/y), assuming a contact during all the hours of a week (24/7).
It is important to specify these limits are useful as recycling and clearance criteria rather than dismissal or disposal. Other parameters and limits (such as Low/medium/high level waste, shallow land burial limit,etc.) have not been taken into account for this study.
It is also important to note that a difficulties may arise in the recycling of complex components, made of different constituents. In the case of this study the STR1 layer of ARC has been considered completely detritiated and composed entirely of a single metal.
The time usually taken as reference to reach these limits in the fusion community are [49] [50]:

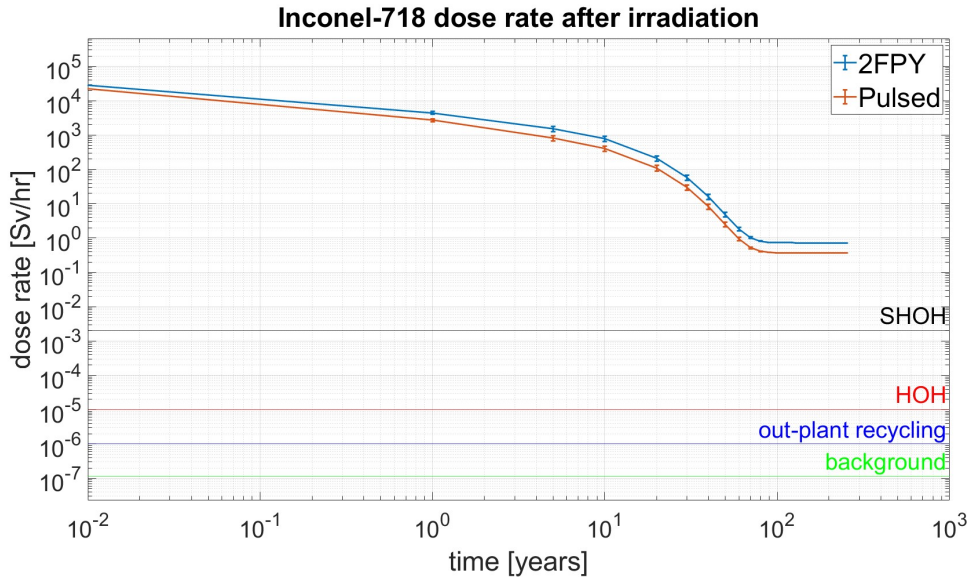
- for "in-plant recycling": 100 years, 10 years is the goal for ARC.
- for "Out-plant recycling": 100 years, 50 years is the goal for ARC.
- for the Clearance index: 100 years, 100 years is the goal for ARC.

5.3.1 Results

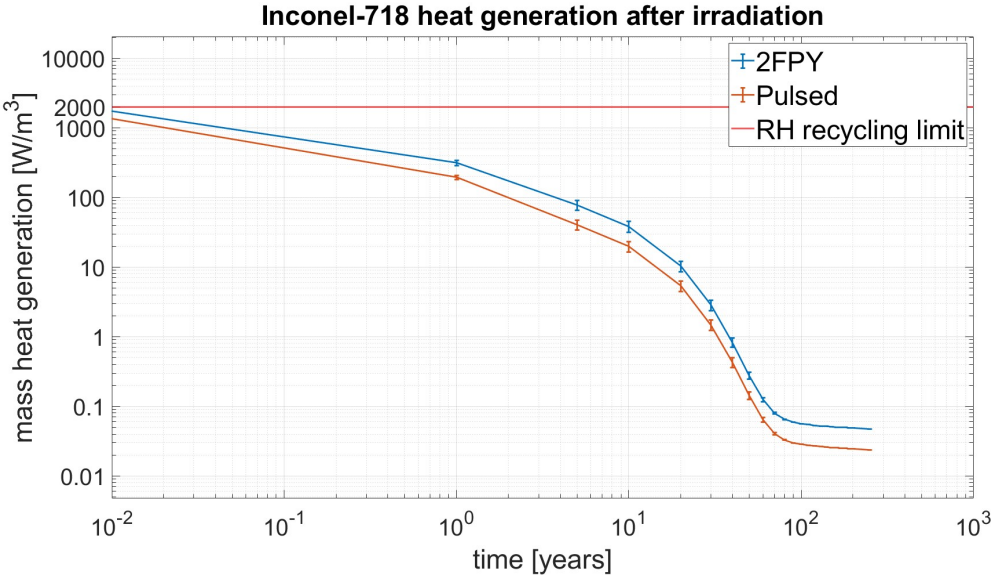
Figure 5.2b shows the result for 2 FPY of operation, compared with the limits previously described. Even if Inconel-718 is a good structural material thanks to its great properties at high temperature [48], it shows values of activation far higher than the prescribed limits for recycling, This is mainly due to the presence of high-activation elements (Nickel, Molybdenum and Niobium) in its composition. The 2 FPY operation shows a long-term contact dose rate at around 1 Sv/h. Even considering a lower neutron exposure, as in the pulsed operation, the long-term contact dose rate remains far above the limit for in-plant recycling.
The clearance index after 100 years for the 2 FPY operation is 2.02E+07.
The results obtained with a pulsed operation do not differ much from the previous case. The dose rate remains at around 400 mSv/h after 100 years, again far above the limits for recycling. The clearance index in this case is 1.02E+07, still far above the desired value of 1.
Such levels of radioactivity may preclude a "zero waste option" for this type of material, since higher levels of radioactivity make recycling more complex and ultimately uneconomic; nonetheless it could be possible to treat it with remote handling (RH), since it does not present high heat generation, as shown in figure 5.2c. This means recycling the first structural layer of Inconel-718 would still be possible, even though more difficult and certainly more expensive.



(a) Specific activity [Bq/kg] during cool-down of the 2 FPY configuration (blue) and the pulsed configuration (orange). The limit for recycling in foundries is shown (red).



(b) Dose rate [Sv/h] during cool-down of the 2 FPY configuration (blue) and the pulsed configuration (orange). Limits for in-plant recycling (black and red), out-plant recycling (blue) and background radiation (green) are shown.



(c) Specific heat generation [W/m^3] during cool-down of the 2 FPY configuration and the pulsed configuration. The limit for RH is shown in red.

Figure 5.2: Plots concerning the activation of ARC first structural layer in Inconel-718 during cooling time.

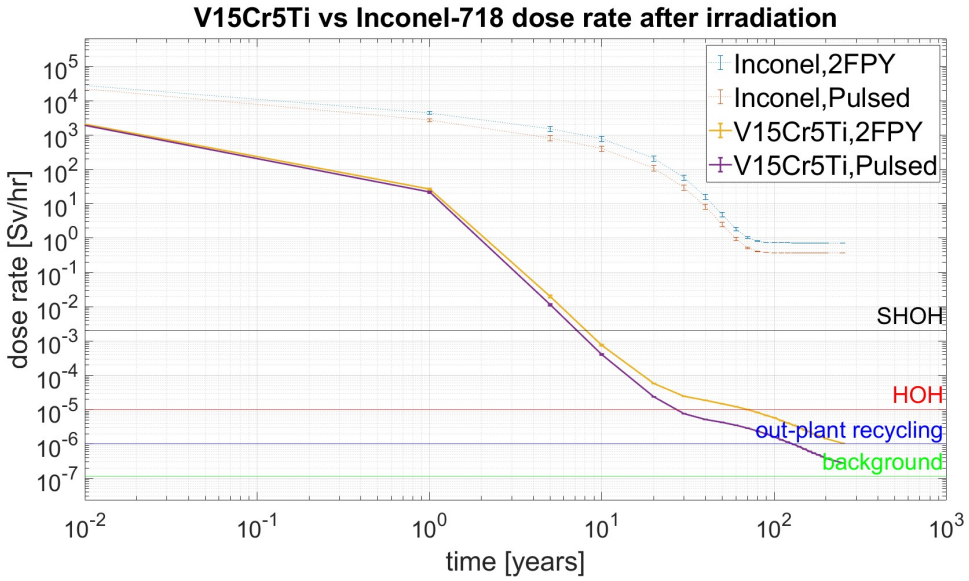
Similar studies [26] [51] show comparable results for what concerns the first structural layer of Inconel-718 and propose the use of alternative reduced activation materials, in particular Vanadium-based alloys. Vanadium alloys can become an optimal choice for materials exposed to neutron fluxes; in recent years a great progress has been made especially for what concerns fabrication and research on corrosion damage, two crucial aspects for the implementation in a fusion reactor such as ARC. However, critical issues regarding high-temperature operation, low-temperature embrittlement of the material remain to be solved [52].
Vanadium is resistant to activation, but some of the impurities may generate long-lived radionuclides, such as Nitrogen transmutation in C-14; in any case the concentration of such impurities should be several orders of magnitudes lower than the main elements in the alloy. In order to make a more realistic comparison with Inconel-718, impurities will be considered inside the chosen Vanadium alloy. One material that can be taken into consideration is V-15Cr-5Ti: the composition of this alloy is shown in table 5.3. The density of the alloy, necessary for the calculations in FISPACT has been taken from [53].

Element composition	V	Cr	Ti	N	O	C	Si	Fe
Weight %	80.39	14.5	5.0	0.0096	0.033	0.012	30.04	0.02

Table 5.3: V-15Cr-5Ti chemical composition by $wt\%$ [54]

The resulting dose rates for a 2 FPY operation and pulsed operation are shown in figure

5.3a. compared with Inconel, Vanadium alloy reaches the HOH recycling limit after 100 years and, in the case of pulsed operation, can be even considered for out-plant recycling. Despite the far better performance under the recycling point of view, its CI after 100 years remains high, being 3.64E+3 in the case of pulsed operation.



(a) Dose rate [Sv/h] during cool-down of the 2 FPY configuration and the pulsed configuration for V15Cr5Ti and Inconel-718. Limits for in-plant recycling (black and red), out-plant recycling (blue) and background radiation (green) are shown.

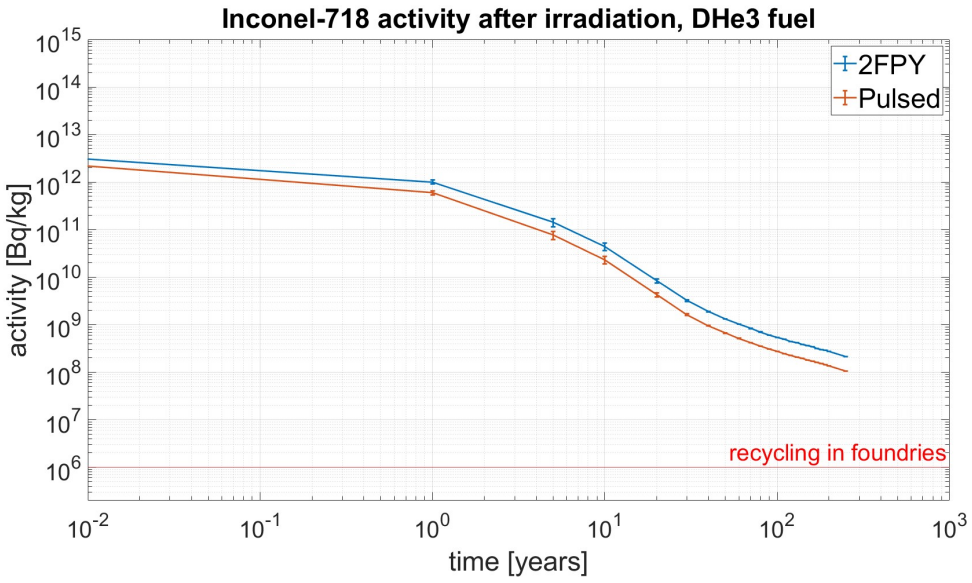
Figure 5.3: Comparison of the dose rate during cooldown time of Inconel-718 and V15Cr5Ti.

5.4 Reduced activation with advanced fuels

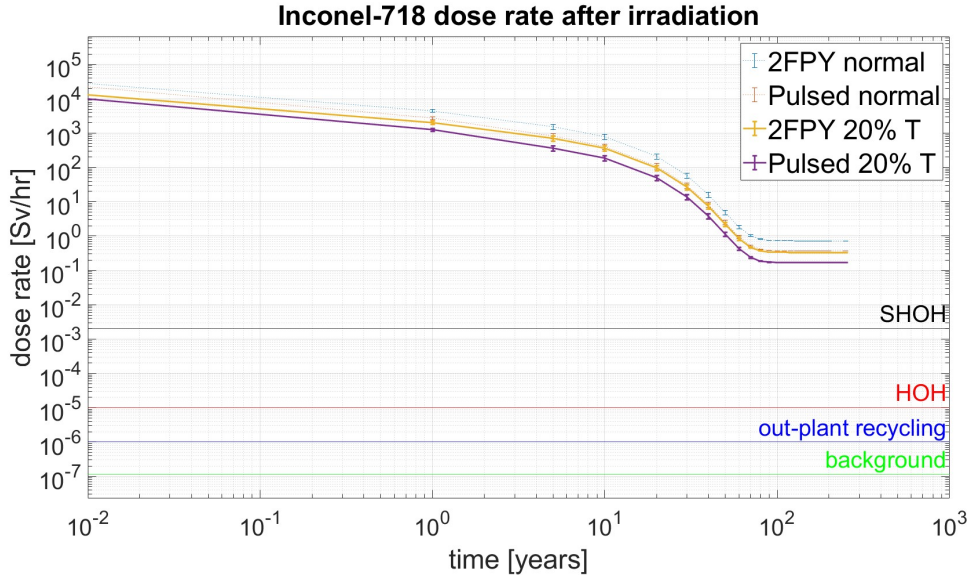
A possible path for the reduction of materials activation could be the choice of a different fuel mixture. The final aim of this analysis is to find a possible solution for the achievement of zero-waste without the change of structural materials, for this reason only Inconel-718 will be considered.

5.4.1 Tritium-poor fuels

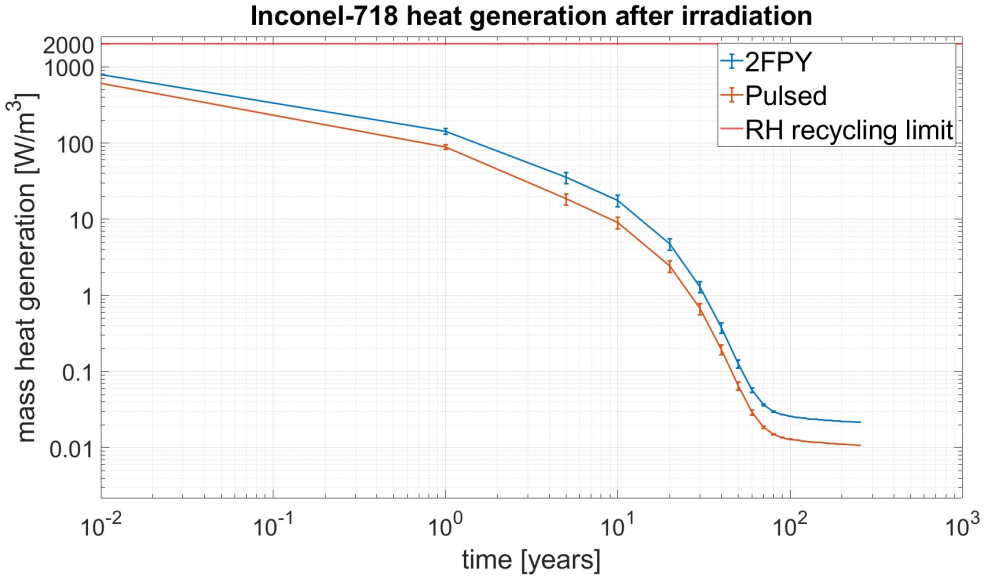
A first, more practical, option is the use of Tritium-poor fuels. A pure DD fusion operation is less efficient and generally more difficult to maintain with respect to a DT mixture. S. Zheng et al. [6] show that with a reduction of the Tritium content from 50% to 20%, the fusion power is almost halved, but the total neutron yield is reduced by a factor of about 2.2, accounting for the summed contributions of DD and DT reactions. A further reduction of the Tritium fraction to 10% yields a neutron production by a factor of about 4.4.
A 80% D and 20% T reduces the activation of the first structural layer of Inconel-718 of a factor close to 2.2 (figure 5.4). This approach is not sufficient to reach the HOH limit. It is therefore necessary a further reduction in neutron flux if Inconel-718 wants to be kept as structural material and an easy recycling route is the final goal. The high activity implies the need for RH recycling, that can be performed since the heat generation is lower than the 2 kW/m^3 limit.



(a) Specific activity [Bq/kg] during cool-down of the 2 FPY configuration (blue) and the pulsed configuration (orange) for operation with 80% D and 20% T.



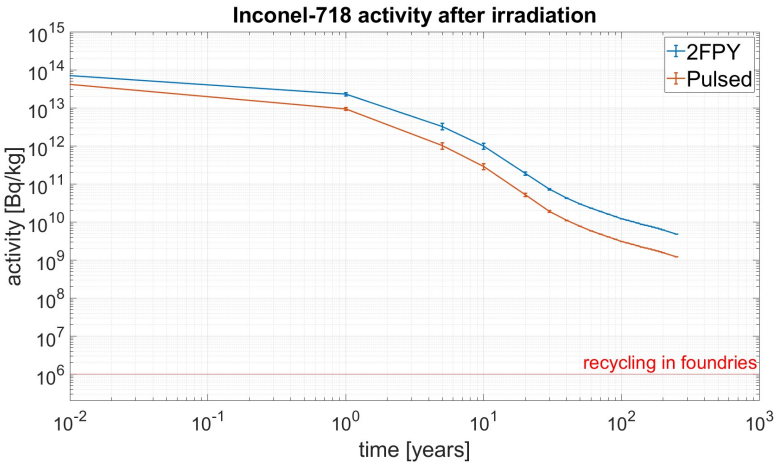
(b) Dose rate [Sv/h] during cool-down of the 2 FPY configuration (yellow) and the pulsed configuration (purple) for operation with 80% D and 20% T. Limits for in-plant recycling (black and red), out-plant recycling (blue) and background radiation (green) are shown, as well as dose rate values for 50D50T operation (dotted lines).



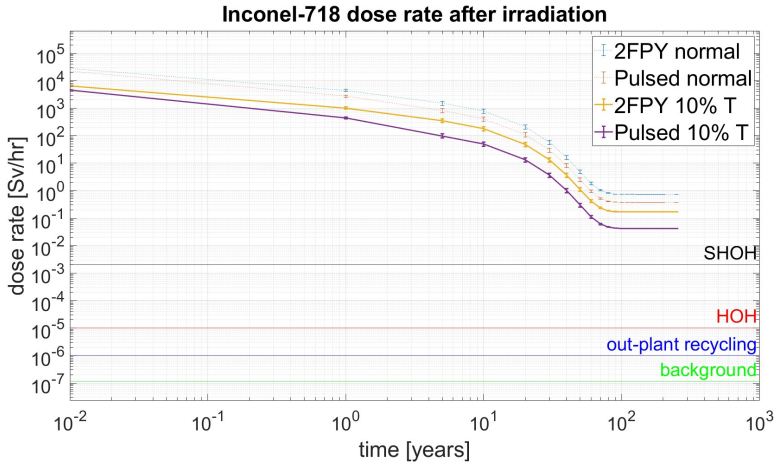
(c) Specific heat generation [W/m^3] during cool-down of the 2 FPY configuration (blue) and the pulsed configuration (orange) for operation with 80% D and 20% T. The limit for RH recycling is shown in red.

Figure 5.4: Specific activity, dose rate and specific heat generation of ARC first structural layer in Inconel-718 for operation with 80% D and 20% T.

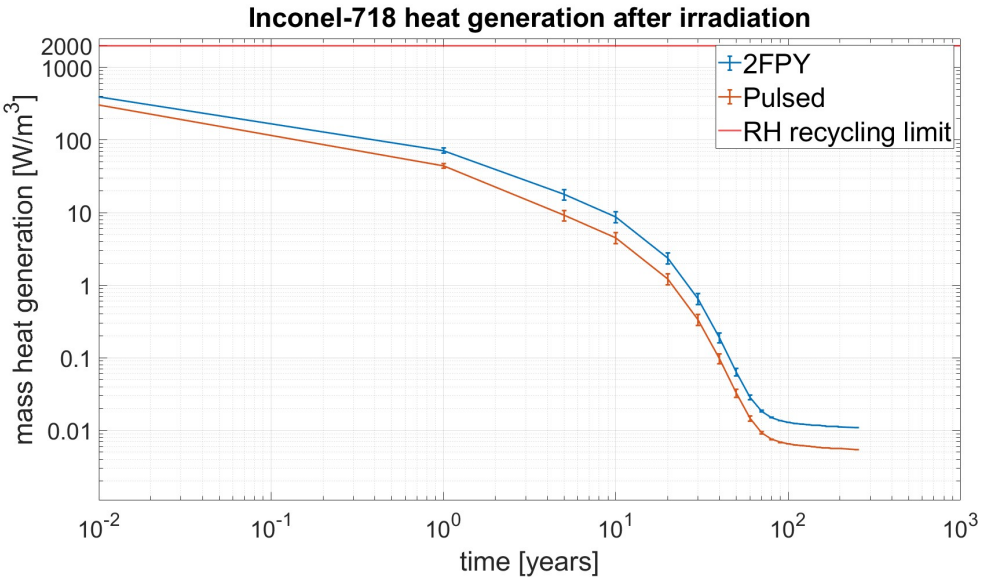
A similar calculation has been carried out with a further reduction of the Tritium content to 10%. In this case the reduction in power becomes about 1/4 and the neutron production is decreased by a factor ~ 4.4, as shown in figure 5.5. The high dose rate and activity preclude the adoption of HOH recycling techniques and RH becomes necessary.



(a) Specific activity [Bq/kg] during cool-down of the 2 FPY configuration (blue) and the pulsed configuration (orange) for operation with 90% D and 10% T.



(b) Dose rate [Sv/h] during cool-down of the 2 FPY configuration (blue) and the pulsed configuration (orange) for operation with 90% D and 10% T. Limits for in-plant recycling (black and red), out-plant recycling (blue) and background radiation (green) are shown.

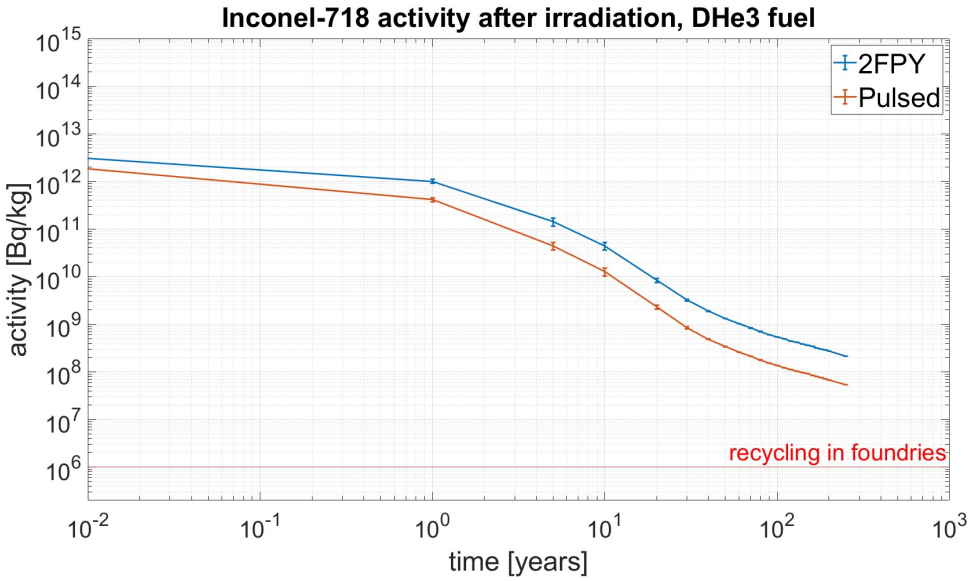


(c) Specific heat generation [W/m^3] during cool-down of the 2 FPY configuration (blue) and the pulsed configuration (orange) for operation with 90% D and 10% T. The limit for RH recycling is shown in red.

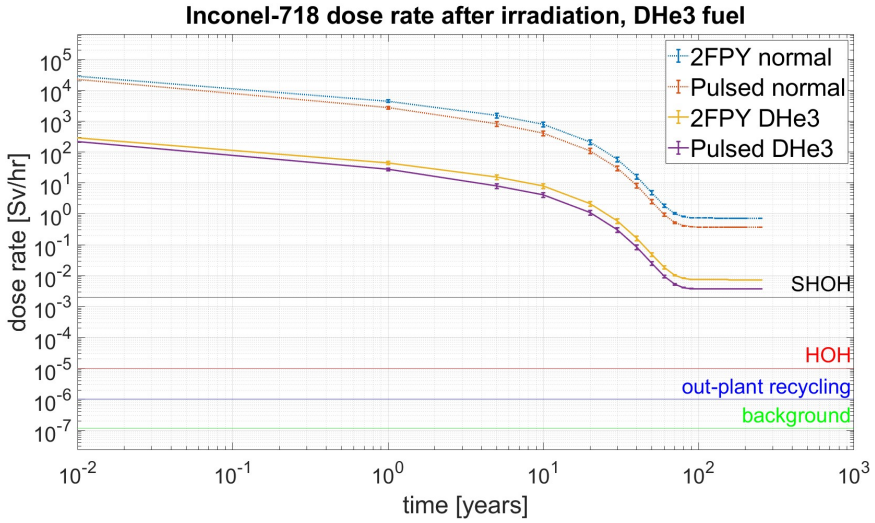
Figure 5.5: Specific activity, dose rate and specific heat generation of ARC first structural layer in Inconel-718 for operation with 90% D and 10% T.

5.4.2 Deuterium-Helium3

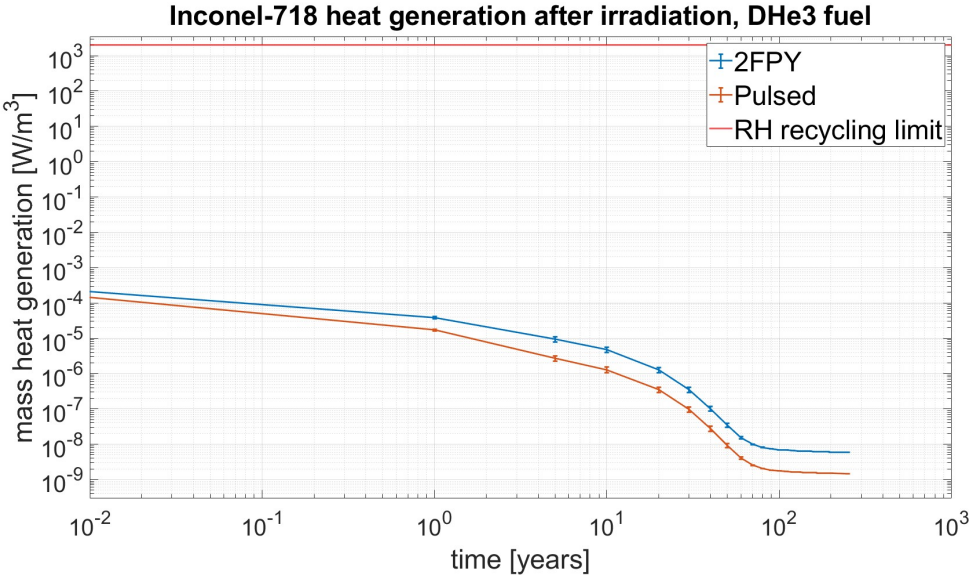
It is interesting to calculate the activation coming from a possible operation of ARC with a D-He3 fuel mixture.
Normally a DT reaction liberates $\sim 80\%$ of its energy via the release of 14 MeV neutrons [55], while in a pure 50-50 DHe3 mixture, the fraction of power by neutrons is estimated to be $\sim 1\%$[56]. For an initial estimation, a reduction of two orders of magnitude of the neutron flux can be taken into account for a 50% Deuterium - 50% Tritium operation in ARC with the same power output. It must be noted that the percentage of power transported by neutrons does not directly translate to the neutron flux on the walls, and a more complete study would be necessary in order to provide a correct value. Furthermore this estimation doesn't take into account the different neutron spectrum produced by DHe3 reactions, nor the use of Tritium to startup the reaction ("Coppi's match") that would increase the production of high-energy neutrons.
The considered flux for the subsequent calculations is therefore $3.69660343e{+}12\ \frac{neutrons}{cm^2 \cdot s}$.
The results for a 2FPY operation show a considerable reduction of the dose rate, of approximately two orders of magnitude, but still not enough to reach the limit forHOH recycling. In this case the clearance index would be 2.05E+05.
With pulsed operation it would be possible to get closer the SHOH recycling limit in 100 years, but the activity (figure 5.6a) would still be too high for recycling in foundries and would need the use of special techniques.



(a) Specific activity [Bq/kg] during cool-down of the 2 FPY configuration (blue) and the pulsed configuration (orange), using DHe3 fuel.



(b) Dose rate [Sv/h] during cool-down of the 2 FPY configuration (yellow) and the pulsed configuration (purple), using DHe3 fuel, compared with normal operation (dotted lines). Limits for in-plant recycling (black and red), out-plant recycling (blue) and background radiation (green) are shown.



(c) Specific heat generation [W/m^3] during cool-down of the 2 FPY configuration (blue) and the pulsed configuration (orange) for operation with DHe3. The limit for RH recycling is shown in red.

Figure 5.6: Specific activity, dose rate and specific heat generation of ARC first structural layer in Inconel-718 for operation with DHe3

5.4.3 DHe3 operation, using Vanadium alloy

The calculations performed in the previous sections show that the activation of the first structural layer of ARC is significant, even with the use of advanced fuels. In the discussed cases it is always required the use of Rh techniques and the limit for the CI is never met.
This results would make extremely difficult to reach a feasible and economic "zero-waste" scenario. For this reason it is interesting to calculate the performance of a combined solution, using a Vanadium alloy structure with advanced fuels.
V15Cr5Ti will be considered as material (refer to section 5.3.1), along with a 50D-50He3 fuel mixture.
The combination of the two leads to a substantial reduction of the activation. After about 10 years of cooldown the material respects the criteria for HOH, and after about 25 years the resulting dose is comparable with the background radiation. Unfortunately the clearance index remains higher than unity, being 72.7 after 100 years in the case of 2 FPY and 36.3 for pulsed operation. This is mainly due to the presence of Tritium and C14, product of the activation and decay of Nitrogen impurities. The reduction of this element in the Vanadium alloy composition could significantly reduce the clearance index. An additional calculation was performed, eliminating the impurities inside the Vanadium alloy (V=80.5%, Cr=14.5 % and Ti=5%). In the case of 2 FPY it would be 0.447, while for pulsed operation it would be 0.223.
Only in this case it would be possible to recycle the material outside the nuclear industry, considering a cooldown time of only 100 years.
Further studies would be necessary to assess the economical aspects of this approach and consider the possibility of producing materials with lower levels of impurities.

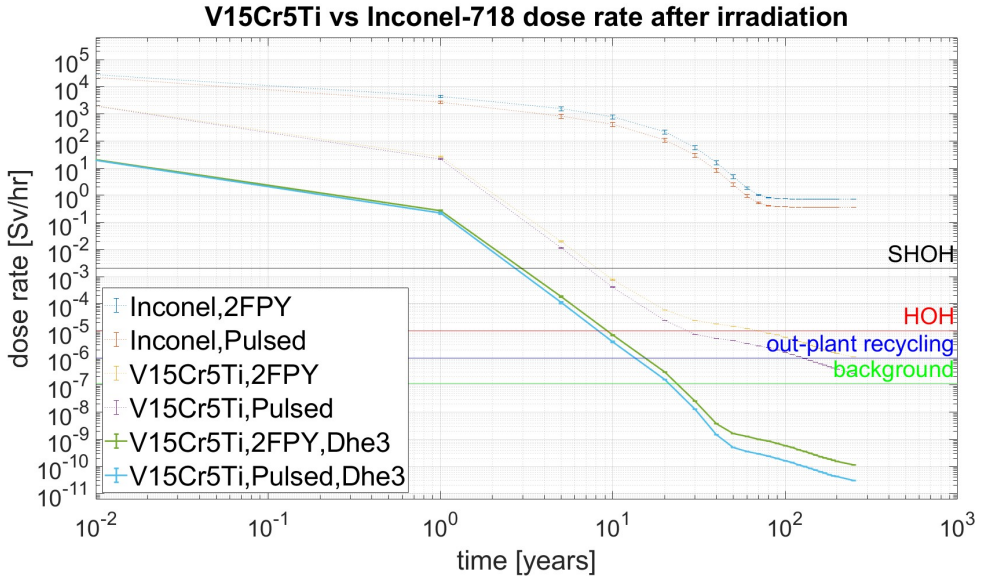


Figure 5.7: Dose rate [Sv/h] during cool-down of the 2 FPY configuration (green) and the pulsed configuration (cyan), using DHe3 fuel and V15Cr5Ti as structural material, compared with normal operation and normal operation with Vanadium alloy (dotted lines). Limits for in-plant recycling (black and red), out-plant recycling (blue) and background radiation (bright green) are shown.

Chapter 6

Conclusions

The recent advancements in HTS magnets can lead the path to the development of more compact and efficient tokamaks, capable of operating with higher plasma parameters. This could lead to the implementation of advanced fuel cycles that can help getting closer to the recycling and clearance criteria aspired in fusion, provided that the issues linked with power production performance are overcome.
One of the most interesting and promising examples is ARC, which design greatly differs from those of ITER and DEMO. The smaller overall dimensions and the position of the vacuum vessel closer to the chamber makes it subject to higher neutron fluxes, leading not only to more neutron-induced damages, but also to activation.
The activation analysis of the first structural layer of Inconel-718, currently chosen for ARC mainly due to its good mechanical properties at high temperatures, show that a prolonged exposure to high-energy neutrons lead to an excessive level of radioactivity even decades after shutdown. A dose of about 1 Sv/h has been estimated after 100 years of cool-down period. This would preclude traditional recycling techniques and probably result in the geneartion of long-lived waste.
The use of low activation materials seems a good candidate for the reduction of neutron-induced radioactivity, Vanadium alloys (V15Cr5Ti in particular) can reduce the dose rate after 100 years of cool-down of several orders of magnitudes, as it has been calculated in section 5.3.1 and in other similar studies [26] [51]; but it doesn't seem sufficient to reach the ultimate goal of an easy and economically feasible "zero waste" option described in chapter 5.1. Furthermore the presence of Tritium inside the fuel cycle, and the necessity to breed it inside the blanket, make it an intrinsic radioactive hazard.
The ambitious "zero waste option" [44] aims to reach the recycling and clearance criteria for all materials employed in a fusion reactor, and the only plausible way to do it seems to be the combined usage of reduced-activation materials and advanced fuel cycles. Furthermore the reduction of Tritium may have beneficial effects not only from the radiological point of view, but possibly economical too [31] [32].
In section 5.4 calculations have been performed on the same structural layer of ARC, but with different fuel mixtures: low Tritium fuels (80D-20T, 90D-10T) and Deuterium-Helium3. It has been estimated that this could lead to the reduction of the neutron flux

of a factor ~ 2.2 and ~ 4.4 in the first cases and of about two orders of magnitudes in the second case. Further studies would be necessary to take into account the different energy spectra of these two reactions, that may lead to a further, albeit minor, reduction to the activation.
Ultimately the complete clearance of materials employed in ARC would be extremely difficult to reach, only the combined effect of reduced neutron flux with 50D-50He3 fuel and reduced-activation with highly pure Vanadium alloys (V15Cr5Ti, with no impurities) would make it possible. Nonetheless the results obtained with advanced fuels and Inconel-718 show an appreciable reduction in activation that could lead to a simpler recycling with respect to the original configuration.
Further development of compact ignition tokamaks employing advanced fuels may help significantly reduce wastes related to nuclear fusion technology and ultimately reach the goal of "zero-waste".

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