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ARC reactor: Preliminary evaluation of tritium storage technologies and related safety issues

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Abstract

The fusion reactor ARC proposed by MIT is a concept of high-magnetic field reactor, using high temperature superconductors (HTS), with an innovative liquid breeding blanket (BB) made of a molten salt (FLiBe). The plasma chamber and the vacuum vessel are contained in the liquid blanket and the FLiBe is used also for the cooling of the vacuum vessel and of the divertor. The lithium (isotopically enriched with 90% 6Li) contained in the molten salt has the aim of producing tritium after the interaction with the neutrons released by the fusion reactions. FLiBe is then extracted from the breeding blanket and sent to a heat exchanger to finally produce electricity with a traditional thermodynamic cycle. The presence of high-magnetic field allows to reduce the dimensions of the reactor, with possible benefits on the capital cost.

The aim of this work is to analyze the tritium breeding ratio to achieve self-sustaining fusion reactors and study the excess tritium production that need to be stored in particular structures to be later sold for new fusion reactors that require fuel to start-up.

Depending on the design parameters of the reactor, the amount of tritium produced and stored can vary significantly. At the same time, tritium is a serious hazard for a fusion power plant. The smaller the tritium inventory in the whole plant, the lower the risk related to tritium accidental releases. Hence, a trade-off between economics and safety must be found.

After calculating the quantity of tritium produced, it will be necessary to analyze the two main storage technologies currently on the market, namely the Zr-Co beds and those of U, to try to decide which is the best one to adopt for the specific case of the analysis.

Through the case study of a possible accidental release of tritium from the containment system in the hypothetical site of realization of the ARC, first of all a technical analysis of the possible environmental impact that would be caused was carried out and then with the help of two specific software, resrad and hotspot, the specific contamination of the surrounding area has been entered with all the specifics of the case and of the place.

Up to the final goal of calculating the dose received by the population surrounding the affected area.

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Chapter 1

Introduction

1.1 General introduction

Fusion power has the promise of offering clean, safe, widely available and abundant energy for future generations. It may play an important role in the future energy mix. Whether this really is going to happen depends on the technological successes that will be achieved with future fusion experiments.
However, technological successes are not sufficient. Fusion as a technology will have to fit in future energy systems and will have to be accepted by our future society. In the possible future energy scenario and future storylines, fusion must compete an important role and makes a comparison with other existing and future energy technology.
Fusion also has a great advantage on its side, which is that it is a potentially unlimited energy source. This point in favor should also fill a defect in the other nuclear source, fission. Fission presents the need to depend on the limited uranium reserves on our planet. Leaving aside this last aspect, fusion is certainly one of the energy sources on which more is being invested for the future and on which many hopes are placed.
The nuclear fusion is a particular reaction in which two or more atomic nuclei are combined to form one or more different atomic nuclei and subatomic particles and in the context of power generation, fusion energy is expected to play a serious role in the long term. A nuclear fusion reactor relies on the fusion reactions that occur within the plasma, which must be limited to avoid interactions with surrounding materials. The main challenge to get a working fusion reactor is to be able to confine the plasma. To date, a widely used technology is the American one of inertial confinement, but which still poses proliferation problems or that of magnetic confinement has been developed, which is the most common type of confinement used in types of reactors such as ITER, DEMO, and is that precisely taken into account in ARC. When it comes to ITER and DEMO, these are certainly well known as they are projects in communion between different European and non-European nations. However, in the field of fusion there are also smaller natures such as that of ARC, a new generation project with reduced powers compared to giants such as DEMO, but entirely developed by a company, specifically the Massachusetts Institute of Technology (MIT).
Of course, since the process is based on nuclear reactions, radioactive risks are associated with it, but on the side of fusion reactors there is the non-production of CO2 during its operational life. However, unlike fission, it is related to lower levels of radioactivity. In fact, the only products of a fusion reaction, just like the most studied

deuterium-tritium reaction, are alpha particles and neutrons. This suggests that the only source of radioactivity is the activation of structural materials after interacting with neutrons. Of course, there is also the presence of tritium which can be dangerous, but its half-life is 12.3 years, so it is not a large-scale problem, although it will be possible to see how fundamental the study of its confinement is for the start-up of future fusion reactors. Most of the radioactive material from a fusion reactor is low and medium level and almost no long-lasting high-level waste is produced, furthermore they are not subject to chain reactions like fission reactors and in so doing they do not suffer from criticality avoiding accidents associated with power runaway. In this sense, a fusion reactor is inherently safe. In fact, when the plasma becomes uncontrollable, the fusion reactions stop automatically and the residual heat derives mainly from the decay of the activated materials inside the structural components, and not from the decay of the fission products; therefore, the removal of residual heat is also easier [1].

The fusion reactions are characterized by a larger density of energy production than the fission reactions because the fuel is formed by light isotopes such as deuterium and tritium, it follows that the mass of fuel required during the life of the reactor is considerably lower than the quantities of uranium [2].

Tritium is not found in nature due to its short half-life. Therefore, the amount of tritium required for start-up and the stationary operating phase must be limited.

To date, the limited quantities of tritium available are produced in fission reactors, in particular in CANDU reactors, but they are not sufficient for the production of fusion energy on a large scale; and it is for this reason that the long-term large amount of tritium will have to come from the fusion reactors themselves. It's also trying to develop other artificial techniques for the production of tritium and to describe this phenomenon in physics there is a precise quantity called tritium breeding ratio (TBR). Reactors for energy production are not only made from reproductive materials, but also from structural ones and if neutrons are absorbed within the latter and do not interact with lithium, they are lost, meaning that they do not contribute to the production of tritium. For these problems the TBR is actually smaller than 1. But a certain number of produced tritium atoms can roam inside the blanket extraction systems and for this reason, it is not only important to achieve a TBR = 1, but it is It is also essential to have a TBR> 1 to recover the tritium that will be lost inside the reactor. A possible solution to obtain a TBR> 1 is to use some materials that have the function of neutron multipliers, such as beryllium, to compensate the neutrons absorbed within the structural components, and to integrate lithium with its 6Li isotope; because its tritium production cross section is better than the 7Li which is characterized by a threshold behavior. Furthermore, having a TBR> 1 is also necessary to provide the tritium needed for starting another reactor. It is therefore easy to deduce how the calculations relating to the TBR and the obtaining of a value as high as possible are fundamental studies [3].

Similarly, as in fission reactors, fluids are also used in fusion reactors during the electricity production process but if in general, the fluid used in fission reactors is water, used both as a coolant and as a moderator, in fusion ones can only be used as a coolant; so it can be deduced that water may not be the best choice because it cannot be use both as coolant and as moderator. A possible choice, which should also be the one adopted by ARC is FliBe, a molten salt composed of fluorine, beryllium and lithium. The FliBe also allows to replace the solid blanket with a liquid one.

Differently from tritium, for deuterium there are no particular problems related to the amount available, since it is estimated that there are about 4.5×10^{-13} t of deuterium in seawater [1].

In addition to the liquid blanket, the Affordable, Robust, Compact (ARC) fusion reactor, developed at MIT, also brings with it other innovations, one of that is the use of high temperature thermal superconducting (HTS) magnet confinement materials, allowing to realize higher magnetic fields and to scale the reactor quantity while maintaining the fusion power almost like bigger projects like ITER. In fact, one of the biggest challenges for ARC is to achieve the density value equal to ITER but in a smaller volume, thus giving the manufacturer the possibility to reduce production costs, but also the time required for construction. By the way, the electric power of ARC (200MWe) is much smaller than other fusion projects like ARIES (1000MWe), but it is well known that this does not necessarily represent a disadvantage. The future, in fact, is increasingly oriented towards the construction of a larger number of reactors, but with smaller dimensions to facilitate their control. To date, the greatest technological and physical challenge is to try to develop materials capable of withstanding the large heat fluxes produced by fusion reactions within the plasma chamber, especially within the divertor region. Some possible solutions could be the use of liquid rather than solid metal deviators, to intervene on the plasma physics in order to obtain the plasma detachment or to design advanced, double-null, long-legged divertor. The latter solution is proposed as part of the ARC design. Another problem of the numerous fusion reactor designs (such as ITER and DEMO) is that the plasma confinement and therefore the generation of the plasma current necessary for the confinement, are obtained inductively with the transformer principle. This suggests that the production of the plant cannot be continuous but exhibits a pulsed behavior, which is never optimal as a physical principle in the field of energy production since it limits the likelihood of using fusion reactors as base load power plants. It is certainly an ambitious project, but it could lead to improving or even solving two of the biggest current problems for fusion reactors.

1.2 Aim of the work

The description above shows how many technological challenges still need to be faced in order to make fusion a marketable energy source, but others have already been overcome. All this makes merging a great challenge, but full of satisfactions for the global energy future.

The main important aspects to consider in the framework of the modelling are: the plasma physics, the divertor heat flux challenge, the neutronics and the magnet design, but also in the framework of the safety in which the main important aspects are the activation of the material, the security of the workers and the treatment of all the products.

This work on ARC is focused on one or the most important product of the fusion reaction, the Tritium.

First, tritium production , the quantity burned, the one produced to finally arrive to the surplus of tritium, were evaluated.

Since the reactor is still in a conceptual design phase, it is necessary to proceed to a parameterization so that it is possible to analyze the different results that can be obtained by varying the structural parameters to understand the feasibility range.

Is also important to understand what the right maintenance time for the material in the plant is before sending it to future reactors, strictly correlated to the concept of the doubling time, which must not be too short since the quantity obtained would not be sufficient and not too high as there would be a risk of a high decay of the material due to its short half-life (12.3 y).

After that it is important to discover how to storage this quantity of Tritium in the best way of trade-off between security and economic costs.

Therefore, understand how many levels of containment necessary and what type of engineering are to adopt.

Also, for this point it will be necessary to use a trade-off strategy by exploiting the choices made by previous reactors such as ITER and DEMO as first.

Finally, the analysis highlights the aspect of the environmental impact of tritium caused by a possible accidental release looking at the categories most impacted by tritium. This analysis is supported by the use of a software called RESRAD.

Chapter 2

The ARC reactor

2.1 The design

The affordable, robust, compact (ARC) reactor is a new conceptual design supported the tokamak concept developed by Massachusetts Institute of Technology (MIT) researchers and is a conceptual tokamak design which will function as both a demonstration fusion power plant for energy generation and a fusion nuclear science facility (FNSF) for integrated materials and component irradiation testing during a D-T neutron field but with the aim of reducing the dimensions, cost and complexity of traditional designs of fusion reactors.

ARC is based on a deuterium-tritium plasma, because the more traditional designs, with a comparatively small fusion power (around 500 MW), just like the fusion power of ITER. The great challenge is increasing the magnetic flux; in many of the fusion reactor designs (for example ITER and DEMO) the magnetic flux is generated by superconductive materials that lose superconductive properties when the generated magnetic flux is just too high. For this reason, if the aim is to extend the fusion power increasing the magnetic flux, the simplest solution is to use heat superconductive materials like REBCO tapes, characterized by high critical current density at high magnetic fields.

The mayor disadvantages of the quality fusion reactors are associated with high cost and long construction time and thus the implications of possible failures during the event of the planning might be dramatic. But with the thought of reducing at 200 MWe of output power, the ARC reactor set its position on a lower risk curve having the ability to become a legitimate alternative within the market. Going a bit more into the detail, ARC may be a high magnetic flux design using REBCO (rare-earth barium copper oxide) heat superconducting (HTS) tapes and inboard lower hybrid current drive (LHCD) to get the plasma current during a non-inductive way. The other main issues as said before concern the capital cost and the related financial effort, that is often an obstacle for the realization of reactor themselves, but ARC tries to reduce the capital cost in the most intuitive way: minimizing the reactor size. Pay attention, reducing the size means that the coils are nearer to the source of neutrons, with possible consequences on the shielding; if the size of ARC is smaller and the power density [MW/m^3] is bigger, consequently the flux at the border increases. This is a problem that is important to take into account from the safety point of view. In this way, without compromising the performances of material, the capital cost is reduced thanks to the smaller volume [4].

Using HTS, a lower amount of superconducting materials is required so there is more space for structural materials. The main drawback of HTS is that up to now they are underdeveloped compared to coldness superconductive materials and there are still many issues associated with cost, mechanics, quench and anisotropy. However, the possible advantages are so attractive that research in HTS is currently a main topic within the fusion field. during this sense, ARC is going to be a crucial test bench. Another advantage with REBCO tapes is that the possibility to use resistive joints within the coils (thanks to the upper temperature, which may enable also the utilization of various coolant like liquid hydrogen or liquid neon rather than liquid helium proposed for LTS magnets in ITER and DEMO), in order that the toroidal field coils are often separated in two pieces with dramatic consequences on the maintainability, enabling to simply replace temporary components just like the blanket tank, the vessel, the auxiliary and poloidal field coils. Modular maintainability and replacement also are allowed by the improved thermal properties obtained because of the upper operating temperature window, guaranteed using REBCO tapes, around 20 ÷ 30 K. Maintainability and construction simplicity are two key aspect within the design of ARC and may speed up its development on the economic scale. Finally, REBCO tapes also are easier to fabricate than LTS like Nb3Sn. According to the planning of ARC, the plasma energy gain is larger than 10, which is that the design value of ITER it is obtained during a smaller volume with a serious radius of 3.3 m instead of 6.2 m for ITER, and a minor radius of 1.1 m, thanks to the presence of heat superconducting REBCO tapes which allows to succeed the peak magnetic flux on the coil. A parameter almost like ITER is that the fusion power, around 500MW in steady state, is produced during a volume eight times smaller than the one among ITER, with possible consequences on the plasma power exhaust. Other significant parameters for ARC are the plasma current of 7.8 MA, the typical plasma temperature of 14 keV, the typical plasma density of 1.3×10^{-20} neutrons/m^3. The previous parameters are almost like other reactors with an equivalent fusion power like ITER, the large difference of ARC is said to its smaller volume and better magnetic flux [4].

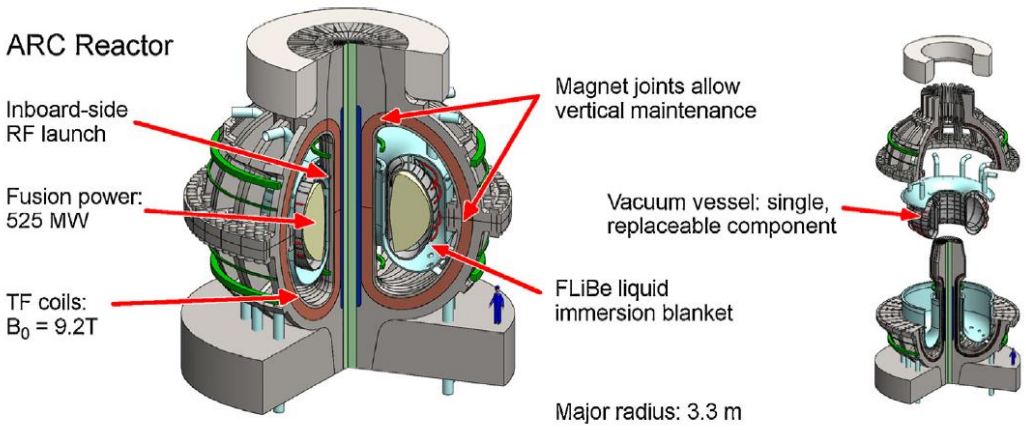

Figure 1 design and dimension of ARC. [3]

2.2 The divertor

ARC is based on a divertor configuration. The presence of a divertor is vital for the performances of the plasma, for instance it contributes to the steadiness and reduces the quantity of impurities inside the plasma itself, avoiding plasma-wall interactions. The main challenge of a divertor configuration is that the flux that has got to be exhausted by the divertor plates is very large. Considering a fusion power of 525 MW, around 105 MW are associated to charged alpha particles which are entrained by the magnetic flux and must be exhausted by the divertor plates. The surface where this power must be exhausted is said to the scrape-off layer thickness. The results of this example may be a heat flux above 10 MW/m^2, which up to now is a very big quantity that necessarily need improvement, in the point of view of the materials, to be sustainable [3].

A key aspect of the divertor design is that the design of the target plates. They are almost like ITER's targets, with a plasma-facing surface made from tungsten, embedded cooling channels and a structural substrate made from another material. The coolant inside the cooling channels is not high-pressure water but heat single-phase FLiBe.

The divertor also benefits of the vertical maintenance scheme of ARC that permits to exchange the whole assembly and reduces the necessity of remote maintenance inside the vacuum vessel. Thanks to the actual design of the long divertor leg, the estimated heat flux on the divertor surface is 1.4 MW/m2 albeit the planning specification is 12 MW/m2 thanks to the uncertainty on the warmth flux and to possess a bigger margin of safety [4].

A possible solution is to style advanced divertor geometries with extended volumes for the divertor and extra poloidal field nulls. These sorts of advanced divertors generally need a particularly precise control performed by the poloidal coils. In many reactors (like ITER and DEMO) the TF coils are not demountable then the PF coils must be placed outside for practical consideration. during this way the PF coils are far away from the plasma and so as to regulate the form of the plasma they have to hold huge currents and, as a consequence, to sustain huge Lorentz forces. within the case of ARC, the TF coils are demountable, therefore the PF coils are often put inside them nearer to the plasma, with the likelihood to scale back the present within the coils. The presence of the thick FLiBe blanket reduces the flux at the poloidal coils, which otherwise would be subjected to too high neutron fluxes since they are almost the neutron source.

Remember always that the use of FLiBe introduces problems of corrosion within the divertor, so a trade-off between the ultimate thickness associated with the warmth deposition and therefore the minimum thickness associated with corrosion issues must be found.

2.3 The vacuum vessel

The plasma chamber of ARC is put inside a single-piece and replaceable double wall vacuum vessel made on Inconel, in particular with the 718, a nichel-based alloy, chosen thanks to its great properties like high strength and corrosion resistance at elevated temperature, but like all the material in the nuclear field it has also some disadvantages like the extremely susceptible activation due to the presence of nichel. Therefore, Inconel 718 is perhaps not the fabric which will be utilized in the ultimate design, but other materials like Eurofer97 and other are going to be tested. One of the most peculiar characteristics of ARC is that the design is in a continuous evolution and this great advantage because can be the "home" of new ideas and technology. The vacuum vessel is based on a region with high thermo-mechanical loads and neutron fluxes and may be subjected to plasma disruptions, so it is an independent component which is subject to the replacement but without consequences on the other permanent components. Remember always that this is a plant characterized by cyclic thermal loads and neutron fluxes, which may affect the performances and therefore the integrity of the vacuum vessel. Therefore, it is important to evaluate the lifetime of this component. The structure of the double vessel contains a particular channel called cooling channel where the single-phase FLiBe molten salt flows for the active cooling of the vessel itself and also for one other fundamental aspect: the breeding of tritium. The other works that do the vessel are the shielding of the neutron, the protection of the entire physical structure and the communication with the external world with the vacuum ports [3].

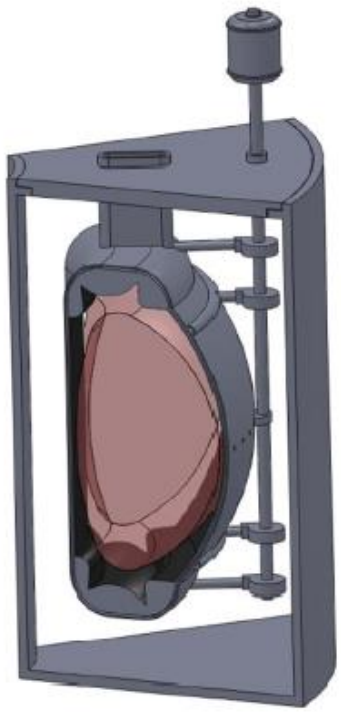

Figure 2 Double-walled vacuum vessel design with FLiBe coolant channel. [3]

2.4 The FliBe blanket

The blanket inside ARC is totally composed of continuously recycled liquid FLiBe which plays four very important role that are: the neutron moderator, the breeder, the shield and the neutron multiplier. But these are not the totality of the role play by the blanket because, the FLiBe works also as tritium carrier, with the tritium extracted from the liquid FLiBe after it flows out of the blanket tank and this is an important aspect that it's need to take into account in our purpose. This is a crucial innovation since just one material is employed for so many different functions. Externally the component that is a single-piece low-pressure blanket tank, is made of Inconel 718, which acts as primary nuclear containment boundary.
The presence of an outsized blanket tank is additionally a plus because simplifies the geometry of the reactor and prevents the problems associated with cyclic loads, since it eliminates an enormous amount of structural and solid materials which might be suffering from fatigue and reduces the quantity of radioactive material too.
The FLiBe may be a high-temperature single phase fluid (between 732 ÷ 1700 K), in order that it is also an efficient thermal reservoir, minimizing the problems of safety associated with two-phase operating fluids like in the typical fusion reactors. The neutrons produced by the D-T reaction undergo the skinny double vacuum vessel and reach the blanket where they deposit their energy that is then extracted from the blanket and converted in electrical power. FLiBe is not only the foremost suitable liquid salt for the moderation, breeding and shielding, but it is up to now for the purpose of ARC seems to be the best choice possible due to the lower magnetic flux in contrast to DEMO, and thanks to the use of this molten salt there is an extremely reduction of the MHD effects within the flow. Having low MHD effects, means the likelihood to realize sufficient fluid flow with low pressure drop and pumping power. The shielding effect is also more relevant using the thick FLiBe blanket and remember that the role of neutron shielding inside a little fusion reactor is perhaps even more important than large scale reactors and it is ready to affect the operational lifetime of the superconducting coils. The result is that the upkeep of the coils is reduced also because the cost of superconductive materials just in case of substitution and it must always be borne in mind that the economic aspect is of great importance in a small reactor [4].

2.5 The magnets design

The design of the ARC magnet system considers REBCO tapes consist in four set of different coils:

- toroidal fields,
- central solenoid and
- poloidal field coils and
- auxiliary coils

TF coils and PF coils are steady state superconducting magnets for stability, shaping and startup of the plasma. The conductor utilized to produce the magnetic flux is usually composed of copper and Hastelloy, with a skinny layer of REBCO superconductor operated around 20 K and during this seems that it is "sub-cooled", because its operating temperature is far less than the critical one (80 K). However, having a better temperature margin as during this case might be a problem just in case of quench, with an outsized amount of energy concentrated at the situation of the quench itself without energy propagation within the coil.

On the opposite hand, with a better temperature margin it is harder to possess a quench. because of the upper temperature allowed by the utilization of HTS, the magnet cooling in ARC is performed by liquid hydrogen that is pressurized between 5 ÷10 bar to extend the liquid temperature range and thanks to its abundance and low cost compared to liquid helium. The cooling circuit of TF coil is found inside the copper stabilizer, while the joints are cooled by channels inside their honey-comb structure and therefore the two circuits are independent. Other benefits associated to the upper temperature are the reduced thermodynamic cost of the cooling and therefore the enhanced thermal stability because of the upper heat capacity.

The 18 toroidal field coils have the standard D shape structure, demonstrated to be the simplest to attenuate the mechanical stress, made from cryogenic 316 LN chrome steel to support the large Lorentz forces. The central solenoid is especially used for inductive startup of the plasma current and for off-normal plasma current control, while in steady state operation the plasma current is generated non-inductively by lower hybrid current drive (LHCD).

The PF coils are wont to generate the X-points inside the plasma and are located outside the blanket tank but inside the TF coils and they are shielded both by the FLiBe blanket and by neutron shields made from zirconium hydride. There is also a group of poloidal coils for outboard plasma shaping and equilibrium fields outside TF coils.

The auxiliary coils are located inside the blanket and near the plasma because their aim is to regulate in real-time the form of the plasma, helping to avoid disruptions. They are subjected to intense flux, for this reason they are made by copper and not by superconducting materials. The auxiliary coils inside the blanket are cooled with FLiBe [4] [3].

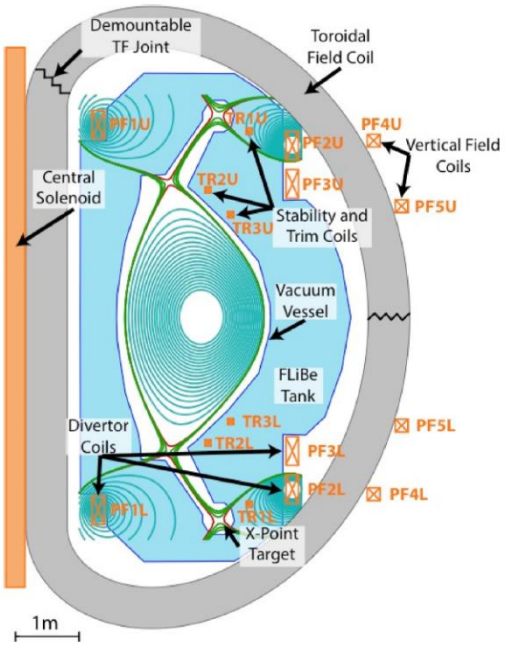

Figure 3 Magnetic equilibrium, coils set, vacuum vessel and FliBe tank geometry of ARC. [4]

Chapter 3

The Tritium production

3.1 Evaluation of tritium production

To evaluate the tritium production in a fusion reactor there are different possibilities but the most useful one is starting from the fusion power of the reactor and calculate the fusion frequency using the energy for fusion tritium atom that is a standard value as presented in this formula: [2]

$$Fusion\ frequency\ (f_f) = \frac{P}{E_f}$$

where *P* is the Fusion Power and E_f is the energy for fusion tritium atom.

After that is fundamental to discover two important values that are the tritium burned in the plasma, called T_b, and the tritium produced in the blanket, called T_p:

$$T_b = M * m_p * f_f * s_{day}$$

Where M is the atomic weight, m_p is the mass of proton, f_f is the fusion frequency and s_{day} are the second in a day.

$$T_p = T_b * TBR$$

Once these formulas have been obtained, it is possible to obtain the value of the tritium surplus, called T_s, by difference between production and burnt:

$$T_s = T_p - T_b$$

Finally, it is fundamental to take under control the decay during time of the tritium, calculate the coefficient of half-life starting from the general value of 12.3 years and using the next formula, the results came from:

$$Tritium\ with\ decay = Tritium\ without\ decay * \left(\frac{1}{2}\right)^C$$

In table 1, the design parameters of ARC to carry out the analysis here described are reported.

Fusion power of ARC	*525 MW*
Energy for fusion tritium atom	*2.82*e-12 J*
Atomic weight (M)	*3*

Mass of proton m_p	*1.67*e-27 Kg*
Second in a day s_{day}	*86400 s*
TBR	*1.084*
Full power year	*356 days*
Coefficient of half-life (C)	*0.081*

Table 1 Technical parameters of ARC. [4]

Fusion Frequency	$1.861 * 10^{20}$ $[1/s]$
Tritium burned in the plasma (Tb)	29.414 [Kg/fpy]
Tritium produced in the blanket (Tp)	31.9 $[Kg/fpy]$
Tritium surplus without decay	2.486 $[Kg/fpy]$
Tritium surplus with decay	2.35 $[Kg/fpy]$

Table 2 Results calculation for ARC.

3.2 Parameterization of TBR and Availability Factor

Two quantities are really crucial for proper analysis: TBR and AF.
The TBR is defined as the ratio of the rate of tritium production in the system to the rate of tritium burned in plasma. The feasibility of tritium breeding depends on both basic physics and engineering issues.
The AF of power plant is the amount of time that it is able to produce electricity over a certain period, divided by the amount of the time in the period. Occasions where only partial capacity is available may or may not be deducted.
Starting from the values obtained in the previous calculations, because now the project of ARC is in the design phase, to evaluate better the different construction possibilities for MIT it is fundamental to parametrize the two parameters described before. This value determines the amount of tritium required in order to have a self-sufficient fuel cycle and to compensate for tritium losses by radioactive decay during time between production and use, and during fusion system shutdown, to supply tritium inventory for start-up of other reactors and to provide a "reserve" inventory necessary for continued reactor operation under certain conditions, like failures. Furthermore, low initial inventory of tritium reduces the risk associated with high amounts of radioactive material [2].
It is also necessary to take into account that to arrive at the standard value of TBR during the past has be done additional detailed neutronics calculations of the impact of various materials, design options, and physics and technology choices on the achievable TBR [1].
From these results has been started many calculations of tritium breeding by many authors and all of these are associated with uncertainties. Uncertainties in calculating the achievable TBR are in three areas:

1. System definition: the achievable TBR depends on many system parameters and design considerations that are not yet well defined (e.g. the amount and configuration of structure, the required FW thickness, the using separate coolant and/or neutron multiplier, the need for electric insulator, the chamber penetrations, the absorbing materials in plasma-stabilizing shells, the choice of divertor, and the plasma heating). By the way up to now, no blanket has been built or tested yet. For instance, note that up to 30% reduction in TBR would result from using 20% of the blanket volume for structure.
2. Modeling and calculation method. There have been major advances in neutron transport calculation methods and codes since the early 1970s. A lot of new Monte Carlo and discrete ordinates (Sn) codes have been developed based on significant improvements in the methods and utilizing the tremendous progress in speed, storage capacity, and faster data handling of modern computers. So, during the years all is in evolution and progression so it is fundamental to keep up with the

times to reduce as much as possible the error and the uncertainties. For instance, neglecting heterogeneity effects results in errors up to ∼10% in the predicted TBR.

3. Nuclear data. The uncertainties in the achievable TBR associated with nuclear data are primarily due to uncertainties in the measured cross sections, energy and angle distributions of secondary neutrons. Another uncertainty arises from processing the cross sections into multi-group data libraries. However, this uncertainty can be greatly reduced by relying on continuous energy cross section data or using a fine energy group structure. Many cross-section sensitivity/uncertainty analyses have been performed to provide an estimate of the uncertainty in the calculated TBR in different blanket concepts and where values in the range of 2%–6% were found. Then there is uncertainty of ∼5% in measuring the tritium production rate in the mockup assemblies. And last but not least there is a 10% uncertainty of the nuclear data of the blanket [1].

From this it is immediately understood how a parametrization of the TBR is necessary and in ARC the choice of the range analyzed is:

$1 \leq TBR \leq 1.12$

Since the effectively operational time during the year of arc is not yet known obviously for the AF the uncertainties are lower but not zero, for this reason it is very important to choose a range for the AF, in such a way as to be able to evaluate all possible variations, that is: [5]

$50\% \leq AF \leq 75\%$

Starting from these two ranges at fixed value of TBR, they were consider in particular four case (1.005, 1.04, 1.08, 1.12), with a discretization of 1% each time of the AF they have been discovered the values of tritium surplus at 1 year with and without decay, then they have been plotted these results in a graph to better highlight the difference between the four cases [6].

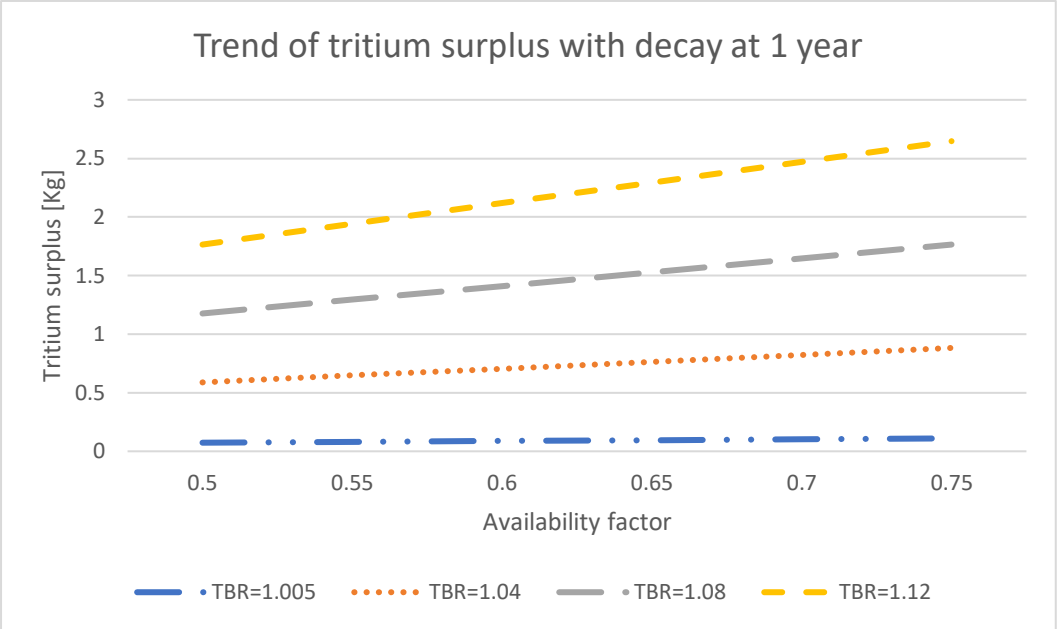

Figure 4 graphical comparison of the four cases.

- Case 1 (TBR=1.005) worst-case scenario in which the value of TBR is the lowest possible. From the result highlighted in the graph it is immediately easy to understand that the quantity of tritium surplus is insufficient to sustain the reactor but also to produce a new tritium inventory for future reactors. So, this option is forced to be discarded.
  It is immediately easy to see how the tritium surplus even if adopting the maximus AF does not overcome the 0.1 Kg/year, totally insufficient for our purpose!

- Case 2 (TBR=1.04) pessimistic scenario with uncertainties in which the value of TBR is very low and the value of uncertainties are the highest. From the result highlight in the graph is possible to derive that the quantity of tritium that the reactor produces in not so much, is sufficiently to sustain the reaction due to the losses but it is very difficult to achieve the goal of the doubling time production of a new tritium inventory is a range of time consistent with the market demand. So, in terms of feasibility this option is forced to be discarded.
  It is immediately easy to see how the tritium surplus even if adopting the maximus AF it is only able to sustain the continuity of the reactions, but it does not overcome the 0.8 Kg/year, and also in this case it is totally insufficient for our purpose!

- Case 3 (TBR=1.08) optimistic scenario without progress in the actual TBR in which the value of TBR is consistent with the standard of theory and the value of uncertainties are in an acceptable range. From the result highlight in the graph is possible to derive that the quantity of tritium that the reactor produces is sufficiently to sustain the reaction due to the losses and if the system of storage works with high value of efficiency, the goal of the doubling time production of a new tritium inventory is a range of time consistent with the market demand, is achievable. So, in terms of feasibility this option can be taken into consideration.

  It is immediately easy to see how the tritium surplus even if adopting the minimum AF it is, at first, able to sustain the continuity of the reactions but also with all the AF overcome the 1 Kg/year, with pick over the 1.6 Kg/year; so this is the first case analyzed in which the goal has been achieved in a congruent amount of time!

- Case 4 (TBR=1.12) optimistic scenario in which the value of TBR is higher than the standard of theory and the value of uncertainties are reduced to the lowest possible values. From the result highlight in the graph is possible to derive that the quantity of tritium that the reactor produces is sufficiently to sustain the reaction due to the losses and if the system of storage works with high value of efficiency, the goal of the doubling time production of a new tritium inventory is a range of time consistent with the market demand, is achievable in a very "short" time. So, in terms of feasibility, this option is obvious that is the best possible option in the four which have been analyzed.

  It is immediately easy to see how the tritium surplus even if adopting the minimum AF it is, at first, able to sustain the continuity of the reactions but also with all the AF overcome the 1.5 Kg/year, with pick at the order of 2.5 Kg/year; so this is obviously the best case analyzed that would guarantee to reach the goal in a relatively short time, provided that the physical and structural conditions can be recreated to reach these numbers.

Chapter 4

The tritium storage

4.1 The storage beds

Tritium storage beds are a technology developed for the first time in the 1970s to meet the first needs to store hydrogen isotopes in order to study them and use them in future applications. In particular, the importance of tritium was immediately discovered thanks to the first in-depth studies on fusion. From there, the need to improve these technologies to ensure ever greater storage efficiency.
The main parameters to be taken into consideration when developing these techniques are certainly temperature and pressure. These values must fall within precise ranges, which differ between the various types of beds, which will be explained in the next chapter.
Storage beds are part of the SDS components and the most important characteristics of the SDS is to play a role of getter bed, which can be absorbing, storing and desorbing of hydrogen isotopes in the fusion fuel cycle at need at any time. The storage bed, however, is intermingled with various additive and subsidiary devices to actualize and to act as a safety device of tritium storage. Most of SDS component is to be prepared by glovebox system, because of the double confinement of tritium high concentration stream. Thus, SDS glovebox has lot of getter beds, tank, and transfer pump in it. Here fire sector concept is applied as a certain amount of tritium in one tritium confinement barrier. The tritium building is the second barrier, but the glovebox services the intermediate tritium-free area in the operation space.
The schematic feature of the SDS mock-up bed is composed of two containers: primary vessel and outer jacket. Primary vessel is the inner vessel that contains hydride getter material which can absorb and desorb the hydrogen isotopes, inlet/outlet gas lines, filter part, coil heater, cartridge heater for similarity of tritium decay, thermal sensors and so on. Outer jacket is the outer vessel that protects the primary vessel. Outer jacket is to be operating under vacuum to mitigate the heat loss when the primary vessel is heated. On the contrary it fulfills He gas when cooling is necessary so that the heat can be dissipated to the atmosphere by radiation and convection. In in-bed calorimetry application, it is recommendable that there is no convection heat loss through the outer jacket and any conduction loss with thermocouples and electric tubing which are penetrated, directly connected to the outer jacket and/or top flange of the primary vessel, except inlet/outlet gas lines. To overcome these two heat loss problems in in-bed calorimetry operation, the reflector and the feed-through can be equipped between spaces with the outer

jacket and the primary vessel to prevent the radiative and conductive heat loss, respectively.

Having provided the first basic characteristics, it is now important to analyze what are the current types of storage beds that can be used in ARC.

4.2 Types of storage beds

After analyzing the tritium production values within ARC, it is essential to study the possible methods of storing these important quantities of tritium. The main characteristics that a good storage method must possess are:

- medium-long term safety and resistance.
- ease of construction and maintenance.
- low costs.

To date, the greatest experience in this field certainly derives from the CANDU fission reactors which are also the main source of tritium production for first generation fusion reactors [7]. But fusion reactors themselves are also advancing in storage technologies. In fact, it is possible to state that in the current situation the two main technologies taken into consideration and on which the greatest efforts and funds are used are:

1) Uranium beds
2) Zirconium-Cobalt beds

They are two very different types in some respects but also shared by some physical principles. Tracing the history of these two types, certainly the first to have been developed is that with uranium beds due to the basic material that was well known thanks to fission principles, advantages and disadvantages. Later to overcome some problems of this type, the one with zirconium-cobalt beds was developed, new characteristics are being discovered year after year, but which has already been used in some types of fusion reactors. In fact, the future is turning more and more towards this type of storage.

Analyzing both types, the goal will be to arrive at the formulation of which will be the most suitable for ARC.

To do this it is obviously necessary to analyze the pros and cons of the two technologies. The necessary information will be obtained from various sources that vary from the purely scientific field of articles dealing with the two themes, but also to the public information from the actual producers of these two technologies [8].

In fact, it must be stated that these two methods are not only used in the nuclear field but basically have the function of storing all the hydrogen isotopes in the various technological areas [8].

In a fusion reactor that burns tritium, the greatest amount will be in the fuel storage system. There are always various safety problems associated with storage systems. One of the main problems of such systems relates to the risks associated with accidental exposure of the storage bed to the air. Uranium, the metal that was most often used in the past, is known to be pyrophoric in the air, particularly after it has been activated, thus forming hydride, deuteride or tritide. When exposed to air at

ambient or moderately elevated temperatures, a rapid chemical reaction occurs. This result in the transformation of the tritium found on the bed, converting it into the most dangerous form of oxide (HTO). Additional problems that can occur concern the bed itself, including the possible breakage and consequent dispersion of tritium materials and uranium into the surrounding environment. These types of accidents fall within the case of rupture or dispersion caused by accidental events, such as accidental opening of the container or the rupture of the container by breakage fragments of other components in the vicinity. All this must therefore be taken into consideration in the safety analysis. Furthermore, such analyzes, in order to be as precise as possible, require quite detailed information on the chemical response exhibited by these reactive metals when exposed to air or other atmospheres [9].
The Tritium Storage and Delivery System (SDS), one of the major components of the tritium fuel cycle, is designed to store and supply the T & D in the plant. Based on the SDS of ITER and DEMO also the system of ARC necessity of:

1. check the performance of processes,
2. load-in and load-out of tritium from transport containers to SDS circuits
3. storage of tritium and deuterium
4. test the performance of tritium compatible pumps,
5. monitor gas delivery from tank to tank and from bed simulator to tank,
6. control the storage and supply of hydrogen using the hydrogen storage bed,
7. allow bed calorimetry using the heater to simulate tritium storage and cooling capacity of the container bed and
8. supply the required hydrogen mixtures and noble gases (He, Ne, Ar, N2, H2, He + O2) for the operation and control of the tanks.

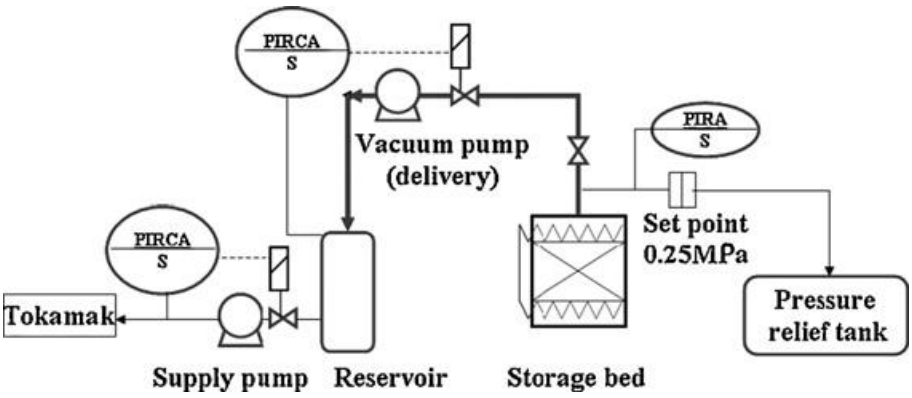

Figure 5 SDS pump. [10]

Having studied the technological and structural characteristics of ARC in the first part, the two technologies will now be analyzed in detail in order to better understand what the strengths and weaknesses of both types of beds are.

4.3 The Uranium beds

Due to its high mobility in metals, tritium can diffuse through the heat exchanger, through the first wall, and through structural materials in tritium handling systems. Therefore any released tritium must be completely recovered both for safety reasons and to prevent environmental contamination. The tritium component coming from the exhaust must also be recovered and separated from helium and other impurities to carry out a complete and correct recycling.

In the past, most studies on tritium recovery and storage have been performed using uranium beds. The fine uranium powder readily reacts with isotopes of hydrogen gas to form hydrides at room temperature and these hydrides readily decompose into uranium and hydrogen gas at approximately 720 K and a pressure of 1 atm. The gaseous tritium recovered in this way is much less dangerous to handle.

The Tritium Storage & Delivery System (SDS) in this aspect has important functions such as the storage and delivery of T, inventory accounting, tracking, and the operation of a metal hydride bed. Depleted uranium (DU) was chosen as a storage material to try to make up for the shortcomings of zirconium-cobalt (ZrCo) in terms of material properties, limiting its application.

The external shape of the bed consists of a primary and a secondary vessel (208 mm-D × 612 mm-L × 2 mm-t). Four cylindrical and disc thermal reflectors were placed between the two vessels to reduce radiant heat from the primary vessel. The main tank contains approximately 1.893 kg of DU (U235 / U = 0.22% by weight), which theoretically should store 23.598 grams of hydrogen [11].

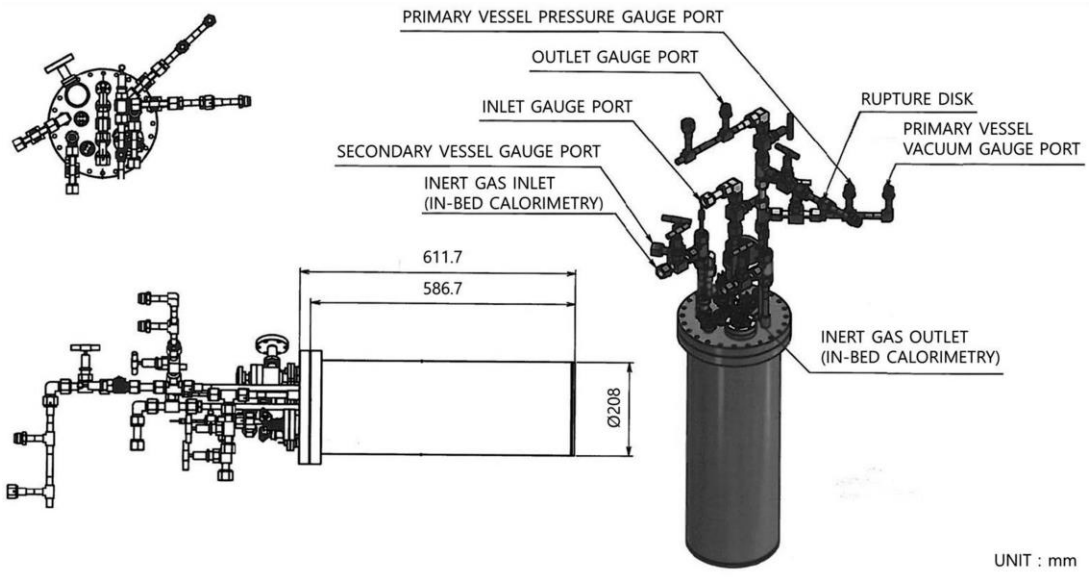

Figure 6 Outer shape of a DU bed. [12]

The DU bed is basically composed of a primary vessel and an external lining for double confinement. The main tank is 316L stainless steel and contains DU hydride

and is equipped with an electric sheath heater, gas inlet / outlet tube, sintered metal filter, thermocouples and other components. The 304 stainless steel outer casing has a conflict flange with gas ports, electrical feedthroughs, and connected gas pipes. Between the primary tank and the outer shell, multiple layers of thermal reflectors are installed to minimize heat loss from the primary tank. First, the bed was installed in the horizontal direction, as shown in Fig. 6. The developers of this experiment chose this installation for effective and uniform heating of the metal hydride. The expansion of the volume of depleted uranium by hydrating is more than 75%, and the volume of the buffer is needed to release the gas from the metal hydride. Therefore, the internal primary vessel requires an expansion volume. However, this extra volume in the primary tank is unfavorable from the point of view of rapid and uniform heating / cooling of the DU powder due to a necessarily increasing thermal inertia and consequently the thermal contact between the DU powder and the primary tank is reduced. To compensate for this, a copper metal foam is installed in the main tank. The pulverized metal hydride is distributed in the open cells of the copper foam and the skeleton of the foam transfers the heat more effectively. The metal foam works as a component that improves heat transfer and is helpful in mitigating the migration of dust in the bed. Making a comparison between the metal fin and the metal foam as a component to improve the heat transfer in the metal hydride bed, it was shown that the metal foam is more effective for heat transfer and absorption / desorption performance [13].

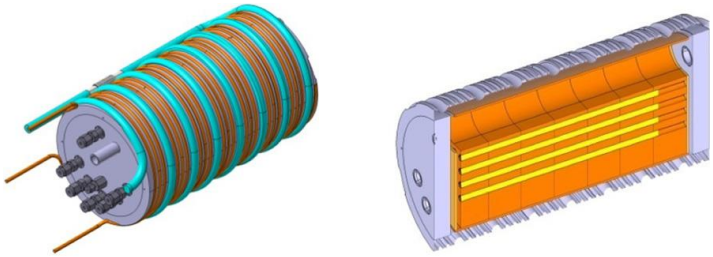

Figure 7 Major design features of current DU bed (a) Primary vessel brazed with electrical heaters and cooling channels (b) DU rods(yellow) loaded into holes in copper foam(orange). [13]

Second, the safe handling of depleted uranium during manufacturing was considered. Due to the pyrophoric characteristics, the handling requires an inert gas condition with the addition of a dedicated zone, which increases manufacturing costs and times.

Focusing on these risks, multiple DU rods were prepared by the injection molding method and inserted into the holes on the copper metal foam (Fig. 7 (b)). The holes on the copper foam were distributed considering the uniform distribution of DU powder in the metal foam cells. Furthermore, DU is eutectic with the iron of stainless steel at a temperature of approximately 630 °C. Consequently, a eutectic barrier is required within the primary vessel. The copper shell and lid were attached to the

inner surface of the main vessel. The electrical sheath heaters and cooling channels were vacuum brazed into the groove of the outer surface of the primary vessel for improved heating efficiency (Fig. 7 (a)). Cartridge heaters were inserted into the metal foam to simulate the decay heat of tritium. The components for bed calorimetry by calorimetric measurement were also installed [13].

Permeation of tritium through structural materials is a key issue in many activities related to the handling of tritium. The loss of tritium due to the high permeation rate through most metals causes radiological risks not to mention the economic side: tritium is an expensive material, and it is for this reason that the few quantities produced must be stored avoiding even the most minimal loss. Ceramic coatings are known to provide an attractive solution for lowering tritium permeation in structural materials. However, it should also be mentioned that the effectiveness of the permeation barrier largely depends on the integrity of the coating. Even small cracks (0.001% of the surface of the coating) have a significant influence on the permeation reduction factor of the protective coating from 1000 to 10 [11]. This is why cracks and pores should be avoided in order to obtain effective barriers to tritium permeation. For the application of alumina coatings as tritium permeation barriers it is important to have reliable technology for building dense, crack-free coatings with good adhesion to substrates. This turns out to be a very difficult technological challenge. For the safe operation of a tritium storage and delivery system (SDS), a metal hydride bed is essential for the recovery and delivery of the gaseous isotope of hydrogen. The main function of the bed is to safely store tritium as a form of metal hydride, quickly absorb emergency tritium gas absorption, and rapidly desorb to meet the fuel requirement and account for tritium at the site by measuring calorimetric of the heat of decomposition [12].

Depleted uranium (DU) does not present disproportion problems that are easy to find in metal alloy hydrides. Furthermore, the decomposition pressure is flat for almost all of the decomposition. It has high pressure delivery potential resulting from high equilibrium pressure at elevated temperature. For these reasons, depleted uranium is one of the strong candidates to be a tritium storage material, despite its radioactivity and handling limitation. Furthermore, depleted uranium is pyrophoric, even when it is not in the form of a fine powder. Hence, the manufacture of the DU bed requires special care and a dedicated environment. Previous studies on uranium bed applications for tritium service or storage have shown different hydrogen discharge results with or without pump operation. However, DU bed process performance validation for service in large fusion plants such as ITER must be done. In addition, design development is required that considers the production of a large number of DU beds. Following the design and construction of several experimental uranium beds, some important results have been achieved, through two main objectives. The first goal is to establish a design and manufacturing procedure that safely handles uranium. The second objective is to perform experiments on absorption/desorption (as visible in Fig. 8), heating/cooling and bed calorimetry [14].

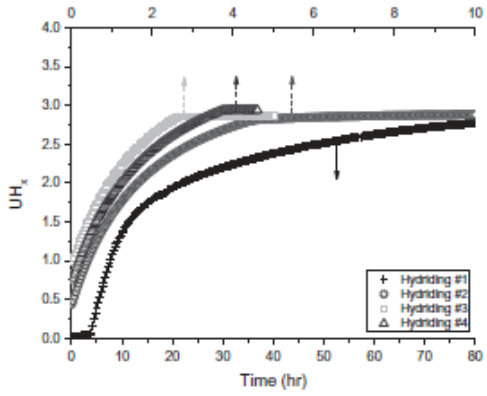

Figure 8 The ratio of hydrogen absorption in the storage material over time. [14]

Generally, the time to maintain the critical average delivery rate of 20 Pa m^3/s is too short and the amount of hydrogen delivery was too low. In view of this, depleted uranium (DU) has been adopted as the hydrogen storage material for the bed due to its wide equilibrium pressure range, as visible in Fig. 9, although it is a nuclear material. It is obvious that the DU bed was able to achieve the hydrogen recovery target of 17.5 mol, of which 92% hydrogen amount could be recovered in 30 min with an average recovery rate of 20 Pa m3 / s. It was found that the amount of hydrogen delivery and the average delivery rate increased with increasing delivery temperature. As time passes at a certain temperature, the delivery rate decreases but the delivery amount of hydrogen steadily increases. By maintaining the average delivery speed target of 20 Pa m3 / s, the maximum delivery quantity and the longest delivery time could be achieved at 450 ° C. At this temperature, the DU bed delivers 16.5 mol of H2 within 30.9 min, which could very well meet the requirement.

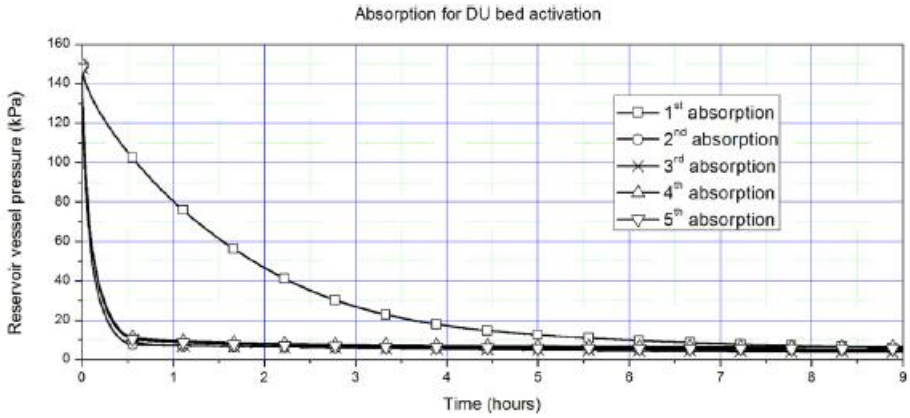

Figure 9 Absorption of hydrogen during DU bed activation – Pressure changes in vessel. [15]

Even investigating a large number of cycles it could be seen that the DU bed showed excellent performance in the measurements of the cyclic hydrogen delivery. The amount of hydrogen and the average delivery rate remain very stable. It was also found that the morphology of depleted uranium changes after cycles of recovery and

release of hydrogen. But at the same time, the considerable volume expansion of the DU causes the filters to deform, suggesting that the DU disk size is too large. It follows that it is better to distribute it homogeneously in the metal hydride layer of the bed to alleviate the expansion effect of DU [15].

Fortunately, during the various analyzes, neither the fracture of the filters, nor the DU dust in the empty space between the filters were highlighted in such a way as to be able to affirm that the filters are still able to filter the DU dust following a higher number of cycles. However, the results of computational analyzes foresee greater attention and greater consideration of the problem related to the expansion of storage materials before carrying out their fabrication and practical application, to avoid breakage accidents. Based on experiments the DU bed showed high hydrogen dispensing performance in terms of fast dispensing rate, high amount of hydrogen dispensing and excellent reversibility, indicating that the manufactured DU bed could meet the fast recovery and delivery requirements quite well. Data are reported in Table 1 [15].

Cycle	Absorbed amount (mole)	Absorbed amount (UH_x)	Time for absorption [min] ($UH_{2.0}$)	Time for absorption [min] ($UH_{2.7}$)	Time for absorption [min] ($UH_{2.8}$)
1	11.49	2.95	111	259	343
2	11.42	2.92	7	22	28
3	11.56	2.97	9	26	34
4	11.66	2.99	8	23	34
5	11.55	2.96	8	23	31

Table 3 activation absorption result. [15]

4.4 The Zr-Co beds

As a second material to be considered and analyzed there is the Zr-Co. An inter-metallic compound, composed of two materials, 51% zirconium and 49% cobalt, that was first studied as a possible substitute for uranium in the Tritium Process Laboratory (TPL) of the Japan Atomic Energy Agency (JAEA) about 30 years ago. The choice derives from the properties of the two individual materials that have already been tested in other operations and have provided excellent results also in the nuclear field [15].

The discovery of this alloy resulted from the study of different metal compositions that were to replace the more expensive and less abundant uranium used up to that time. Metals such as titanium, aluminum and others were tested to be bonded with zirconium which has already been analyzed for applications in the field of fission giving positive results. It was therefore tried to bond these materials together to obtain compounds with the desired characteristics, but nothing was to chance until at first it was decided to bond the zirconium with the nickel forming the intermetallic compound ZrNi. It was soon tested for the recovery and supply of hydrogen isotopes, but the generation of monohydride proved to be a disadvantage. So it was decided to replace nickels with cobalt obtaining the ZrCo compound which overcame the problem of monohydride accumulation and seemed suggest a possible applicability of this material for this purpose [9].

Initially the studies showed a simple substitution of uranium with Zr-Co directly in the reversible getter of tritium in the previous SDS systems and in doing so the first characteristics of this new compound were studied and tested.

The birth of this compound derives from the synthesis and subsequent fusion of an equimolecular mixture of Zr and Co. The melting point of ZrCo was immediately a point in favor because it was tested to be about 1380 ° C, more than 200 degrees higher than Uranium. The actual manufacturing process, however, has several further steps reproduced in a cyclic manner starting from the two materials melted several times until the desired compound and therefore the alloy is obtained. The ingot thus formed reacts readily with hydrogen. The powder sample used was previously prepared by temperature cycles under a hydrogen at 1 atm followed by a vacuum treatment above 450 °C. Subsequently, an X-ray analysis and the subsequent diffraction of the model provide the final data of the material so that it can satisfy the basic characteristics.

At this point the actual development and design phase of the Zr-Co beds comes into play.

One of the main features of this new type is the ability of Zr-Co to bind to the tritium atoms in a very short time so as to increase efficiency and decrease times, reducing losses due to the stagnation time of tritium in the beds.

During this use, the bed is regarded as a vacuum pump that evacuates the gas from the atmosphere obtaining negligible residual values compared to those obtained in the inventory.
In the bed container, the ZrCo activated powder is contained in sintered stainless-steel tubes to ensure a high contact surface between the ZrCo and the introduced tritium. This results in the use of large quantities of Zr-Co and therefore a high production is required. Sometimes a layer of impurities due to helium-3 can accumulate on the surface of the ZrCo thus blocking its maximum extraction functionality on the metal surface. In fact, this type of bed is dominated by large flows of gas that pass through each tube of the filter, and consequently if in a section the Zr-Co powder is inhibited, its performance in that section will therefore also be inhibited. These high flows have precisely the task of avoiding this so-called "cover" by impurities and specifically by helium. This configuration also provides effective "packing" of a large amount of metal powder into a small space without loss of pumping speed [9].
On the other hand, what can be a problem is the increase in the temperature of the metal during hydration which in turn causes an increase in the equilibrium pressure and a subsequent stop of the suction at a much higher pressure than expected. To avoid this, copper grains with Zr-Co are inserted and mixed inside the bed to absorb the reaction heat and improve its thermal conductivity, thus preventing an increase in temperature during the hydration process. In addition, better control of the bed temperature is also guaranteed in the heating process for dehydration.
Another very important element for the recovery of tritium in these beds are the heaters which are applied for regeneration and in so doing the column is contained in a vacuum jacket to confine the tritium that permeates from the primary envelope. Other characteristics that certainly make this alloy attractive are that it is a non-nuclear material unlike uranium and its low pyrophoricity.
However, attention must be paid to the storage capacity of a ZrCo bed which could be substantially reduced by a disproportionate reaction. ZrCo-based alloys suffer from the disproportion reaction and slow hydration or dehydration kinetic property. To date, the efforts on this alloy all go in this direction because if this obstacle could be overcome, it could be said with certainty that the Zr-Co alloy is the best to be adopted for the construction of storage beds. The analyzes on the disproportion mechanism of the ZrCo stated that the disproportion is closely related to the occupation sites of the H atom and the change in the crystal structure, which should be further confirmed by the advanced characterization and theoretical analysis which moreover has discovered that the effect of the isotope is strongly influenced by temperature, an aspect which already presented some problems and on which it will therefore be necessary to intervene. It is therefore necessary to investigate the stoichiometry of the reaction formed in the bed under the reference conditions of the previous studies carried out. The greatest disproportion occurs at high temperatures (~ 4000 °C) and high pressures (~ 100 kPa). The first possibility studied

to lower this rate could be to lower the temperature and pressure by decreasing the reaction time [15].

Storage beds will certainly suffer from multiple temperature cycles to supply and recover DT gas during the entire pulse plasma operation. In fact, studies have reported that the Zr-Co alloy rapidly loses its absorption-desorption capacity from 12 to 25 repetitive temperature cycles between rooms in the transition from low to high temperatures (400 to 600 °C). However, it's still talking about temperature and pressure conditions above those so far experienced and considered in reactors like ITER. Therefore, the effect that a high number of cycles can have on the disproportion remains an aspect to be taken into great consideration and on which it will be necessary to continue to study and verify the direct effects on the reactor. It is always important to make the transition from theory to practice, especially in an area such as that of fusion where many aspects are still uncertain and require a lot of experimentation.

To date, the major field trials have been carried out on ITER and from those it can be deduced that

- The reaction times of a disproportion of 10% under an equilibrium pressure at 350, 380, 400 and 415 °C are 67.76, 7.18, 0.39 and 0.23 days, respectively.
- Tests on temperature cycles have shown that up to 125 cycles there is no disproportion increase.
- The disproportion in the Zr-Co system was confirmed to be very sensitive to a temperature above ~ 3508 °C and a pressure above ~ 3.1 kPa.
- It is necessary to maintain a sufficiently high number of filtering area of the SDS bed (from 230 to 300 cm^2) so that it is sufficient to minimize the pressure drop across the filter during the discharge operation [16].

Maintaining a sort of parallelism between ITER and ARC, it is also possible to analyze what is the Zr-Co bed alternative for ITER through two figures, one in 3D and the other representing the cross section. In Fig. 10 it is possible to view the primary vessel in 316L stainless steel with dimensions 267 × 425 × 4 mm containing three trays and a secondary vessel in 304 stainless steel with dimensions 375 × 650 x 3.5 mm. Three cylindrical and disc thermal reflectors by 2 mm of 304L were placed in between the two vessels to reduce radiation heat from the primary vessel. Then there are thermocouples in strategic location to measure temperatures under different operational conditions. Then, Figure 11 shows a cross section and top views of a tray made by 316L. Each tray consists of a pair of cable heaters of 3kW each, decay heat simulation heaters (8 W/30 V), Helium flow pipe, three filter tubes for H_2 gas introduction, eight rectangular wire mesh bags loaded with ZrCo powder and filter plate [17].

Moving on to a more specific description of the components the bed is a double-walled cylinder containing a thin annular double layer of metal hydride. One side of the metal hydride layer is in contact with the bed wall. The other side of metal

hydride is adjacent to the annular filter, thus preventing the powdered metal hydride from passing into the gas inlet or outlet tubes of the experimental setup. The copper fins are inserted at regular intervals between the layers of metal hydride to avoid the aggregation of the latter and improve its thermal conductivity. Specifically, can be found 8 in the inner layer and 16 in the outer one. Between the double filter, the empty space exists to allow the gas to flow. The cooling fluid tunnels form an M-shaped ring which is pierced on the side of the bed. The heating instead takes place through a rod positioned in the central part of the bed that provides a power of 1700 W and a heating jacket on the side surface that provides a power of 5000 W during the entire recovery and delivery process. To measure the change in bed temperature during the experiments, the thermocouples are respectively located in the inner and outer layer of the metal hydride [17].

Figure 10 3D model of the ZrCo bed. [15]

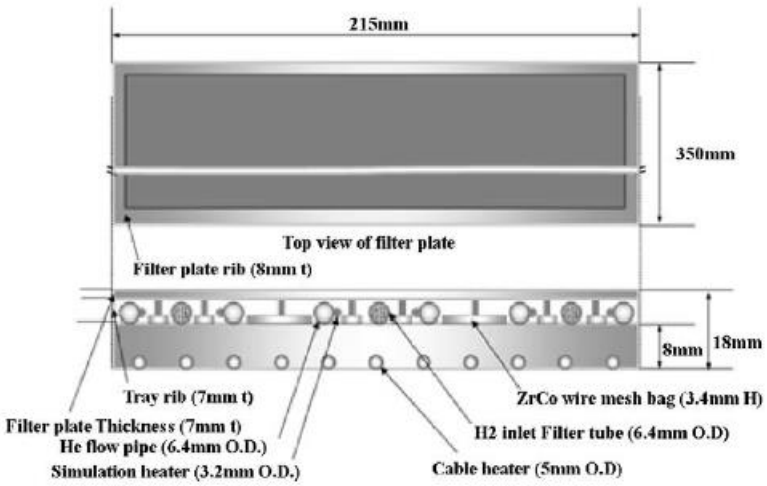

Figure 11 Cross section and top views of a tray. [15]

As previously said given the similarity of power produced, ARC can be compared to ITER. Consequently, it is certainly useful to exploit the analyzes and studies made on ITER to better understand the strengths and weaknesses of this type of bed. In the experiment below, two different modalities concerning the same Zr-Co bed will be considered.

Before starting the two-bed testing process, ZrCo was heated at 500 °C for 8 h under vacuum to remove any contaminants such as moisture and oxygen on the surface.

The high purity gas (99.999%) was then placed in the measuring vessel and then reacted with ZrCo at room temperature. Activation of the ZrCo was carried out through four consecutive cycles of evacuation and hydration. The recovery and delivery quantities of the bed was determined using the Sieverts method, which consists of monitoring changes in temperature and pressure of the measuring vessel and the bed itself. The bed absorbs hydrogen at room temperature from the measuring vessel with an initial hydrogen pressure of approximately 1.45 bar. To study the effects of operating conditions on the delivery properties of the bed, two hydrogen delivery modes were performed for the ZrCo bed. In dispensing mode 1, the desorbed hydrogen during the heating process is confined within the bed before reaching the desired temperature. Once the desired temperature is reached, the hydrogen is transferred directly from the bed to the measuring vessel by means of a metal bellows pump. In contrast, delivery mode 2, there is an additional buffer tank and a scroll pump. In dispensing mode 2, the desorbed hydrogen during the heating process is confined within both the bed and the buffer tank before reaching the desired temperature. Here too, once the desired temperature has been reached, the hydrogen is transferred from the buffer vessel to the measuring vessel via the scroll pump and the metal bellows pump.

Figure 12 shows the hydrogen recovery performance of the ZrCo bed after activation. It has been found that the initial recovery rate is very fast, up to almost 400 Pa m^3/s. However, as the amount of recovery increased, the rate slowed. It is clear that the manufactured ZrCo bed was indeed capable of storing more in a shorter time. Analyzing Fig. 13 it can be seen that using the two delivery modes described above, the curves of the average delivery rate are very close to the start of the processes. However, the dispense rate in mode 2 has become faster than that in mode 1. As a result, the amount of dispense required in mode 2 could be achieved before reaching the critical average dispense rate which is around 20 Pa m^3/s. The improved hydrogen delivery performance in Mode 2 indicated that the additional use of the buffer tank and scroll pump could aid the hydrogen delivery process of the bed. The problem arises by analyzing the total quantities of delivery which in both modes could only reach about 68% of the recovery quantity [15].

The evolution of the pressure inside the ZrCo bed before and during the delivery and heating of the bed in the two modes are shown in Fig. 14. According to Fig. 14a, it was possible to find that the pressure inside the bed ZrCo increases rapidly with

increasing temperature. When the temperature approached approximately 220 °C, the pressure is almost 10 bar. It is known that the disproportion of ZrCo can be induced under high hydrogen pressure and high temperature. Therefore, it can be inferred that the decrease in pressure is caused by the disproportion of ZrCo. Since the disproportion of ZrCo will cause degradation of hydration and dehydration properties, the operating condition of mode 1 is not recommended for the hydrogen release process of the ZrCo bed. Conversely, from Fig.14b, it can be seen that the pressure inside the ZrCo bed was actually relieved during the bed heating process in Mode 2. Even though the ZrCo bed temperature was already above 300 °C, the pressure is less than 0.5 bar. This allows us to decrease both the possible disproportion of ZrCo, but also the risk of reaching high values of temperature and therefore of pressure. Therefore, it can be concluded that the use of the second mode, that is the one that presents the buffer tank, is a good choice that can be adopted in these operations.

Figure 15 shows the ICT (industrial computed tomography) images of the ZrCo bed before and after activation. In the sequence of images a to c, it can be seen that the fabricated bed structure is a thin, double-layered annular. It was possible to observe the structural integrity and structural symmetry of the bed. Meanwhile, the grainy morphology is clearly visible. The granulate is evenly distributed along the cylindrical double layers (Fig. 15b), although most of it is relatively far from the bed wing due to the gravity acting during the loading process (Fig. 15c). As shown in the sequence from d to i, the fine powder of ZrCo is obtained after four activation cycles and is subsequently compacted again by gravity. The structure is however very rigid in fact no deformations or cracks have been observed in the copper fins or filters after activation. No traces of ZrCo dust were detected in the empty space of the bed, suggesting that the filter successfully prevented the transfer of ZrCo dust from the metal hydride layers to the empty space. The expansion of the powder was observed only after activation. However, the volume occupied by the activated ZrCo remains unchanged between the state of hydration and that of dehydration, resulting in about 80% [15].

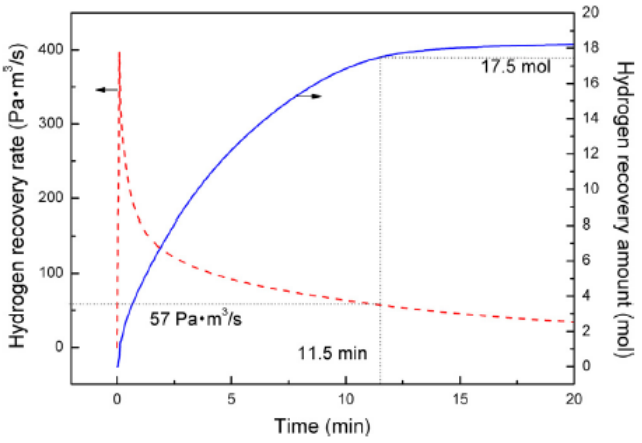

Figure 12 Hydrogen recovery amount curve and average hydrogen recovery rate curve of the ZrCo bed. [12]

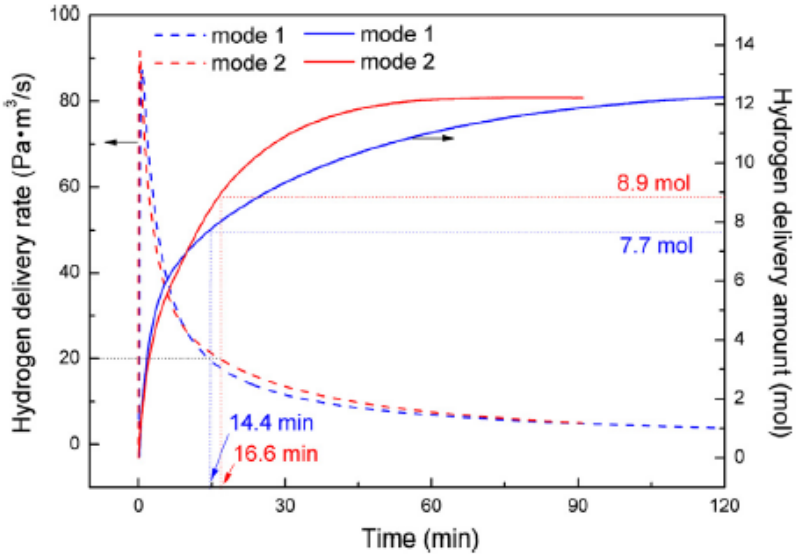

Figure 13 Hydrogen delivery amount curves and average hydrogen delivery rate curves of the ZrCo bed under delivery mode 1 and mode 2 at 400 °C. [12]

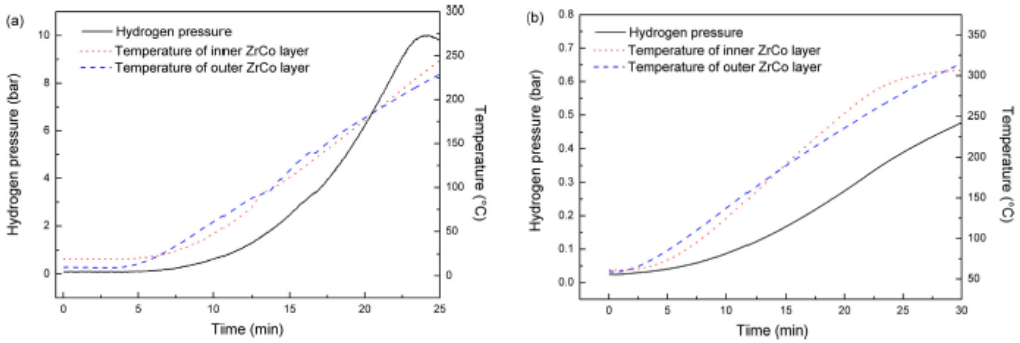

Figure 14 Evolutions of hydrogen pressure inside the ZrCo bed before hydrogen delivery when heating the bed under mode 1 (a) and mode 2 (b). [15]

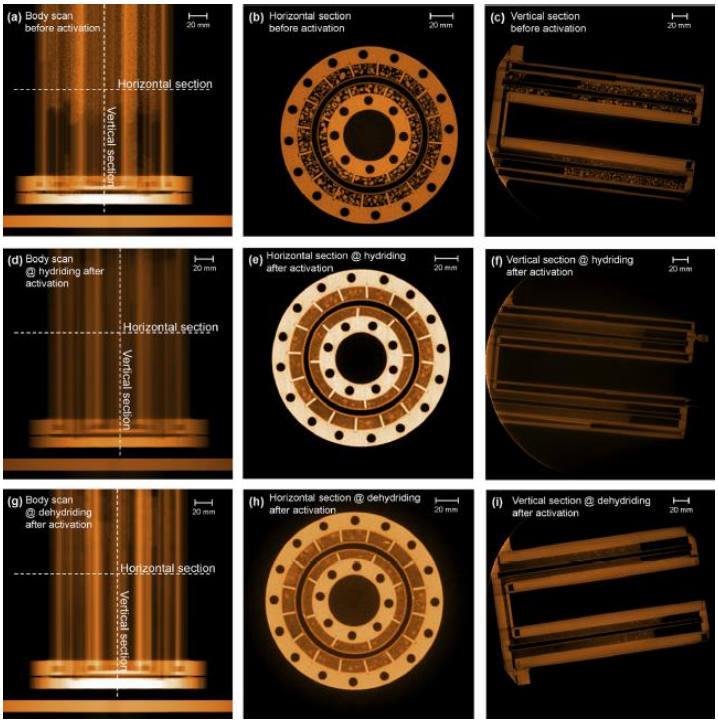

Figure 15 ICT images of the ZrCo bed before activation (a–c), after activation at state of hydriding (d–f), after activation at state of dehydriding (g–i). (a), (d), (g): scan of the bed body; (b), (e), (h): horizontal section; (c), (f), (g): vertical section. [15]

In the case of ITER, as a reference design, the storage capacities of tritium by each bed vary from 70 to 100 g thus requiring the construction of a number of beds ranging from 35 to approximately 50 for storage and delivery of the entire quantity produced. The aspects of design and improvement of the properties will certainly be the basis of all the changes to be addressed from now until the actual realization. All this obviously always with an eye to safety. Furthermore, it is also important to consider the effects of tritium decay, such as heat transfer and ^{3}He behavior, which can cause an increase in temperature and pressure in the tritide vessel. Taking into account that an increase in temperature and pressure are two highly undesirable parameters. Regarding the safety operation, the sequence of the hydrogenation-dehydrogenation cycle is a key issue for the storage system requirements in the melting plant. In addition, emergency situations such as rapid recovery of hydrogen and loss of normal cooling function need to be addressed [18].

For the stable and safe storage of a large amount of tritium it is necessary to take into account the decay heat of tritium which is approximately 0.324 W/g [19]. Consequently, it is necessary to prevent this heat dissipation through appropriate thermal shielding measures intrinsic in the bed itself. The adoption of these measures serves to ensure functional stability of the entire process. Protection from tritium permeation occurs through the use of a secondary vacuum jacket. However, special attention should be paid to treating thin parts, such as the thermocouple and heater sheath and the secondary circuit for decay heat recovery. In particular, tritium is released through the sheath as a hydrocarbon during bed heating and the tritium hydrocarbon is more difficult to remove. The generation of ^{3}He contributes to the

formation of a covering on the walls which is a well-known phenomenon during the recovery of tritium. Therefore, an adequate circulation circuit passing through the storage bed is required.
For this reason, primary vessel pressure monitoring and periodic ^{3}He removal have been implemented as a standardized operating procedure of the tritium storage bed. Based on over 30 years of experience with ZrCo tritium storage beds, the key issues regarding the safety design and safe operation of this bed type have been summarized. Disproportion should be avoided by adjusting the ZrCo bed to a lower hydride temperature and lower hydrogen isotope pressure using a vacuum pump. However, periodic heating up to 873K is recommended to maintain the effective storage capacity of the ZrCo bed throughout the life of the bed. During long-term storage, continuous monitoring and effective control of ^{3}He released from tritide is also recommended, especially if ^{3}He accumulates to more than 0.2 of the ^{3}He / ZrCo atomic ratio [19].

The main characteristics of the Zr-Co beds were also analyzed. From these two chapters it will now be important to draw conclusions since ARC needs a definitive choice on the best tritium storage methodology in the shortest possible time.
in the next chapter, therefore, all the positive and negative aspects of both technologies will be analyzed in detail, up to a personal proposal for a final choice.

4.5 Final comparison and proposal choice

It is now necessary to go back to the pros and cons of the two technologies in order to be able to make final considerations that can lead to a proposal for choosing which bed is more appropriate to use in ARC.
Starting from the uranium beds, it can be said that these have longer application history and therefore also more years of study and improvements, a point that can be seen as positive but not completely, given the technological stalemate that this technology has reached.

	Uranium bed	**Zirconium-Cobalt bed**
PROS	does not present problems of disproportion	not being a nuclear material and therefore not radioactive
	good resistance to high temperatures and pressures	low pyrophoricity with oxygen
	isothermal pressure curve characteristic	low reactivity with nitrogen
	good tolerance to impurities	greater trapping capacity of hydrogen isotopes
	balance of the stability of materials	higher anti-spraying capacity (about 7 cycles)
		possible future advances given the numerous efforts that are taking place on this technology
CONS	being a nuclear and radioactive material	disproportion to high temperatures and pressures
	high pyrophoricity with oxygen	reduction in storage capacity
	high reactivity with nitrogen	need to keep temperatures and pressures lower
	pulverization after a few cycles (2/3)	greater control over the decay heat of tritium required
	need for greater care and a dedicated place for construction	need for intervention on the microscopic material to improve the characteristics
	need for continuous monitoring of uranium quantities by responsible entities	

Table 4 Pros and cons of the U bed and the Zr-Co bed.

Thinking about everything that has been stated and making a comparison between the two technologies, it is possible to immediately understand that there is currently no dominant technology over the other, it is therefore necessary to examine the details and details in order to make a choice proposal. But surely there are a couple of aspects that could make lose the choice on the one hand over the other.
The first is certainly the radioactivity of Uranium towards Zr-Co. In a world that is increasingly moving towards low-risk technologies, introducing into the fusion that has as its cornerstone that of overcoming fission for safety and for non-production of radioactivity, the choice of uranium would be a bit counter-trend.
The public opinion that in some ways has often been opposed to fission precisely because of the problems linked to radioactivity is the same one that has gladly accepted fusion precisely because of its objective of not using "nuclear" materials.
The second fundamental aspect to take into account is the one concerning the preface made on the two materials. Uranium has already reached the peak of its development as a storage technique, while Zr-Co still has many years ahead to improve and overcome the one big problem that remains with this technology, disproportion. Taking into account the probable start of construction of ARC no earlier than 10 years to date, surely the time available to improve the technology is sufficient to take the "risk" of choosing it.
When choosing the technology, it is also necessary to understand the quantity of beds that will have to be available for the plant to satisfy the quantities of tritium produced.
The quantification was carried out in the first part of the thesis and provided a result of tritium produced which is between approximately 2 Kg. According to these data it is possible to make a parallelism with the quantities and numbers of ITER as they do not differ from too much.
About 30 to 50 storage beds will be used in ITER, this number is variable and not yet certain because the storage quantities for each bed have not yet been chosen. The storage capacity can vary from 70 to 100 g for each bed.
Taking into account that approximately 7g of material is required to store about 0.1g of tritium, obviously the most important trade-off is whether to increase safety by reducing the quantity stored in each bed so as to be able to better control a possible breakdown of the system and consequently a spill, but thus increasing the number of beds to be produced, consequently also increasing costs. Or increase the quantity stored in each bed, thus assuming a higher degree of risk in case of system breakdown but decreasing both the number of beds to be produced, and consequently also the costs.
At the moment there are still no studies and scientific bases that can make the choice hang on one side over the other.
However, it can also be said that this is not a big problem to solve, and also that it will be resolved once the risk calculations on possible breakages or spills have been

made, but also following the monetary calculations based on the funds available for the construction of the whole reactor.

Chapter 5

The case study

5.1 Choice of the site

In this second part, the focus will be on the research of the environmental impacts and on the population dose of a possible accidental release of tritium which occurred from the breakdown of one of the previously analyzed storage units. The causes of rupture can be many, but what interests the analysis most are the effects.
First of all, it is essential to locate the ARC construction in an appropriate site, so as to be able to analyze the place and know how to obtain useful information for the actual study.
For this analysis the appropriate site was identified to be the Pilgrim Nuclear Site, figure 16, in Massachusetts, which already has a nuclear qualification and has already high safety levels. [20]

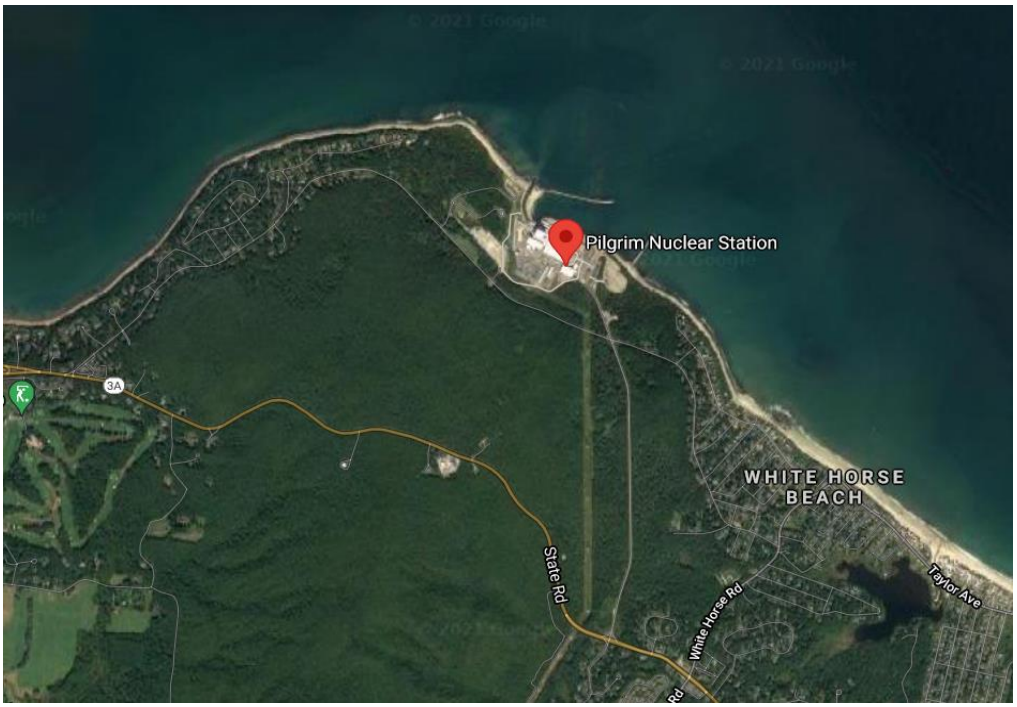

Figure 16 Site map of Pilgrim nuclear station

The site is suitable for several main reasons, the first because it has a good safety range without housing, has a continuous supply of water being along the ocean shore and in addition the site itself is already nuclearized, that is, it already has some basic and fundamental safety measures to be able to build a fusion reactor. Not least the proximity to less than 50 km from MIT, where ARC is under development. [20]

5.2 Software description

To face this analysis it is necessary a software specialized in the calculation of the dose and concentration in the nuclear field, therefore RESRAD will be used: the RESRAD family of codes is developed at Argonne National Laboratory to analyze potential human and biota radiation exposures from the environmental contamination of RESidual RADioactive materials. The codes use pathway analysis to evaluate radiation exposure and associated risks, and to derive cleanup criteria or authorized limits for radionuclide concentrations in the contaminated source medium. The RESRAD family of codes is widely used by regulatory agencies, the risk assessment community, and universities in more than 100 countries around the world. [21]

In the specific case, RESRAD OFFsite will be used which is particularly suitable for assessing radiation exposures of a human receptor located on top of or at some distance from soils contaminated with radioactive materials.

The objective of this software is estimate radiation doses and cancer risks to an individual located on site, i.e., within the boundary of the primary contamination in soil (as considered by RESRAD-ONSITE), off site, or both on site and off. It also derives radionuclide soil guideline levels for the primary contaminated area corresponding to a specific dose criterion. Because of the extension to off-site locations, RESRAD-OFFSITE simulates actual exposure conditions more realistically, and can be applied to evaluate more complicated decommissioning conditions (with restriction) and the effectiveness of environmental monitoring.

The modeling approach of RESRAD-OFFSITE is the same as that of RESRAD-ONSITE. The calculation of dose and cancer risk is scenario driven, by activating exposure pathways and using parameter values commensurate with the scenario under consideration. Nine exposure pathways are modeled, as in RESRAD-ONSITE, which include external radiation, inhalation of airborne radionuclides, ingestion of plant foods, meat, milk, water, and aquatic food, incidental ingestion of soil, and inhalation of radon. Unlike RESRAD-ONSITE, which assumes the residence and agricultural fields are collocated with the primary contamination, RESRAD-OFFSITE allows them to be at different locations, each at a specific distance and direction from the primary contamination as well as of specific dimensions. The same option extends to a well and a surface water body, which can be off the center line of the groundwater flow direction. Accumulation of radionuclides due to air deposition and irrigation at off-site locations is also modeled, contributing to radiation exposures of the receptor.

A Gaussian plume model based on area source release was incorporated into RESRAD-OFFSITE to calculate air concentrations at off-site locations. For groundwater transport modeling, in addition to radiological decay and ingrowth, advection, and sorption and desorption between solid and liquid phases in soil, one-dimensional and three-dimensional dispersion are also considered for the

unsaturated zones and saturated zone, respectively, and the transport extends beyond the boundary of the primary contamination. [21]

The latest enhancement to the RESRAD-OFFSITE code is the incorporation of a source term model, which allows time-delayed and distributed release of radionuclides to infiltration water and provides options to calculate the release as instantaneous, uniform, or 1st order rate controlled. With this enhancement, the application of RESRAD-OFFSITE is no longer limited to the typical soil contamination for which it was initially designed.

The RESRAD-OFFSITE code maintains most of the features of the RESRAD-ONSITE code, including a user-friendly interface to facilitate data entry, performing calculations, displaying calculation results, and accessing general and context-specific help, the choice of ICRP-38 or ICRP-107 radionuclide transformation database, the selection of dose conversion factors and cancer risk slope factors library, and the specification of a cut-off half-life for defining short-lived associated progenies. In addition to performing deterministic sensitivity analyses, it is capable of performing probabilistic sensitivity analyses and features an interface that steps the user through the probabilistic analysis.

RESRAD-OFFSITE maintains and enhances the functions of RESRAD-ONSITE. It is approved by NRC for use in decommissioning and license termination risk assessment. In addition to typical soil contamination, it can also be applied to evaluate human health and environmental impacts associated with radioactive waste disposal, e.g., the evaluation of post-closure performance of the conceptual disposal facility designs at sites that were considered for the greater-than-class C (GTCC) and GTCC-like low-level wastes. RESRAD-OFFSITE accepts direct input of radionuclide release rates to deeper soils, groundwater aquifer, air, or a surface water body, and can thereby be used with other environmental fate and transport models.

Among the preliminary inputs of RESRAD is the nuclide concentration which is a parameter of not so simple research. To do this another software called hotspot was used, in fact, it is not always necessary to use it, but in this specific analysis it was fundamental to arrive at the correct value. The HotSpot Health Physics codes were created to provide emergency response personnel and emergency planners with a fast, field-portable set of software tools for evaluating incidents involving radioactive material. The software is also used for safety analyses of facilities handling nuclear material. [22]

The HotSpot atmospheric dispersion models are designed for near-surface releases, short-range (less than 10 km) dispersion, and short-term (less than 24 hours) release durations in unobstructed terrain and simple meteorological conditions. These models provide a fast and usually conservative means for estimation of the radiation effects associated with the atmospheric release of radioactive materials.

HotSpot is a fast-running, local-scale, steady-state Gaussian plume model for radiological releases developed at LLNL that provides predictions of time-integrated

effects (such as dosage from the entire plume passage). HotSpot was created to provide emergency response personnel and emergency planners with a fast, field-portable set of software tools for evaluating incidents involving radioactive material. The conservative assumptions used in the model also make it suitable for safety and hazard analyses. [22]

The useful model for the analysis is that of the release of tritium, which assumes that it's been set as initial parameter, or the quantity of tritium released which in this case will be 1g, then calculate its total activity in Bq and all the other parameters required, summarized in the figure 17. [23]

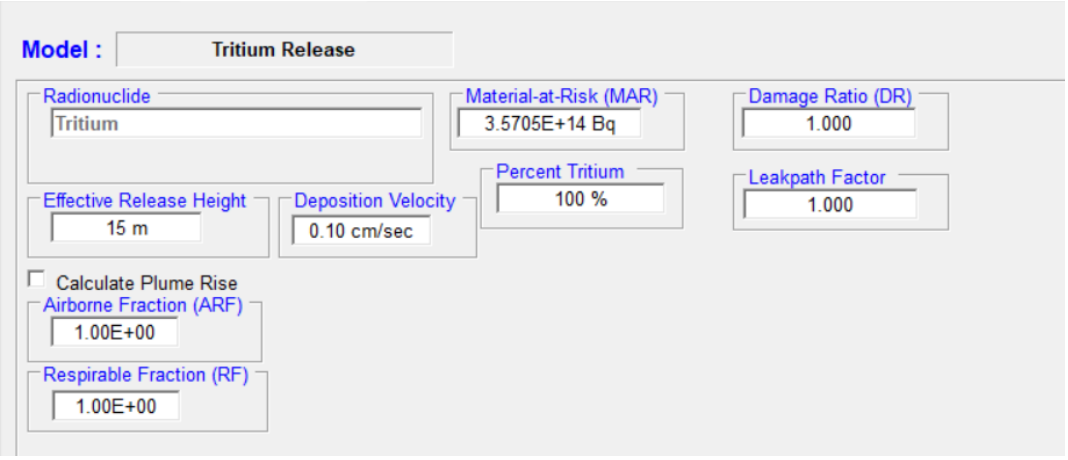

Figure 17 HotSpot input release parameters

In this first frame of the program it was necessary to enter the total activity of the radionuclide involved in the release scenario, the actual release height and the deposition rate.

All the values attested to the standards of these types of release taking into account, thanks to the numerous control systems in the releases from the Zr-Co beds that the actual release on a 70g bed of tritium can be on the unit or 1g of tritium released into the environment.

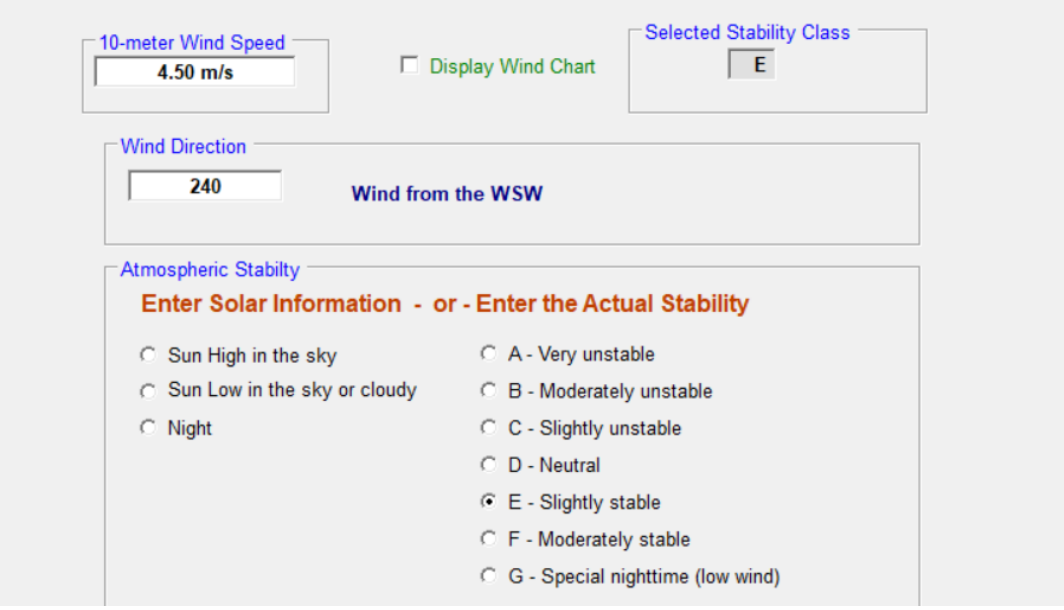

Figure 18 HotSpot meteorological parameters

In the figure 18, on the other hand, thanks to the meteorological data already obtained previously, it has been added the wind speed in the affected area, its annual average direction and consequently the stability class of the area considered.

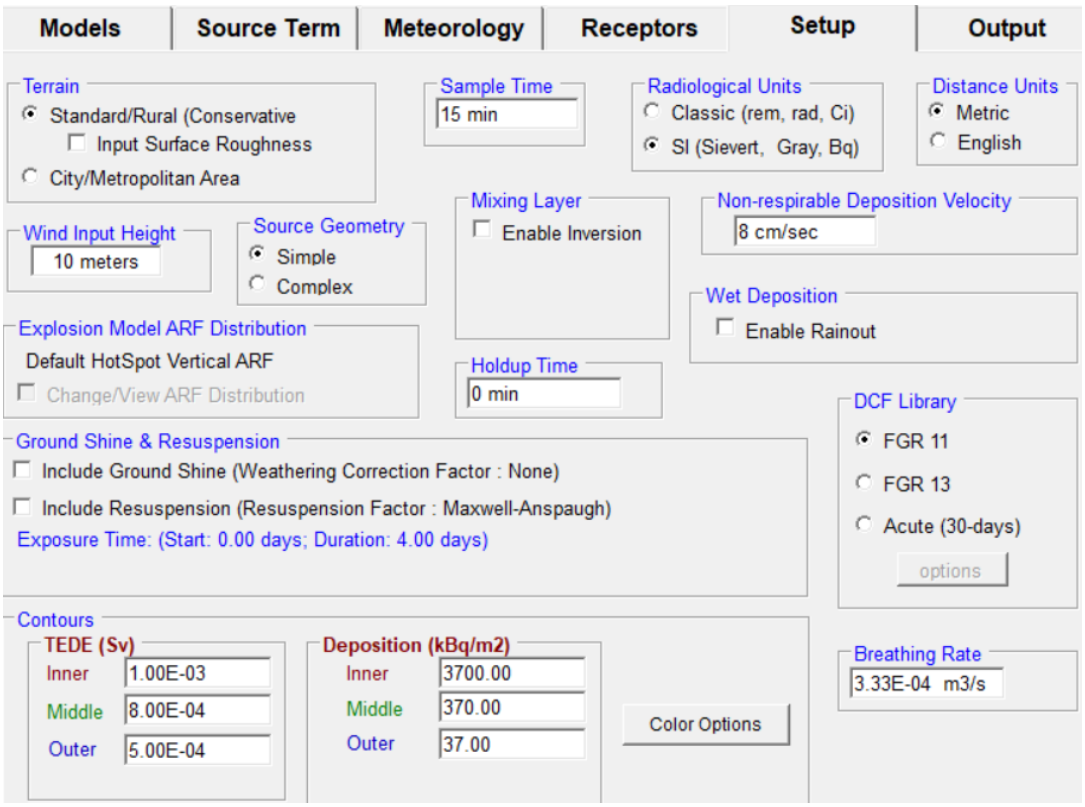

Figure 19 HotSpot setup parameters

In the last step, like in figure 19, before being able to run the program, it was necessary to enter all the setup values, with particular attention to the subdivision into zones (inner, middle and outer), which is fundamental for the analysis of the

results in the three zones, starting from the most critical one, still inside the nuclear site until gradually moving away to analyze the surrounding area.

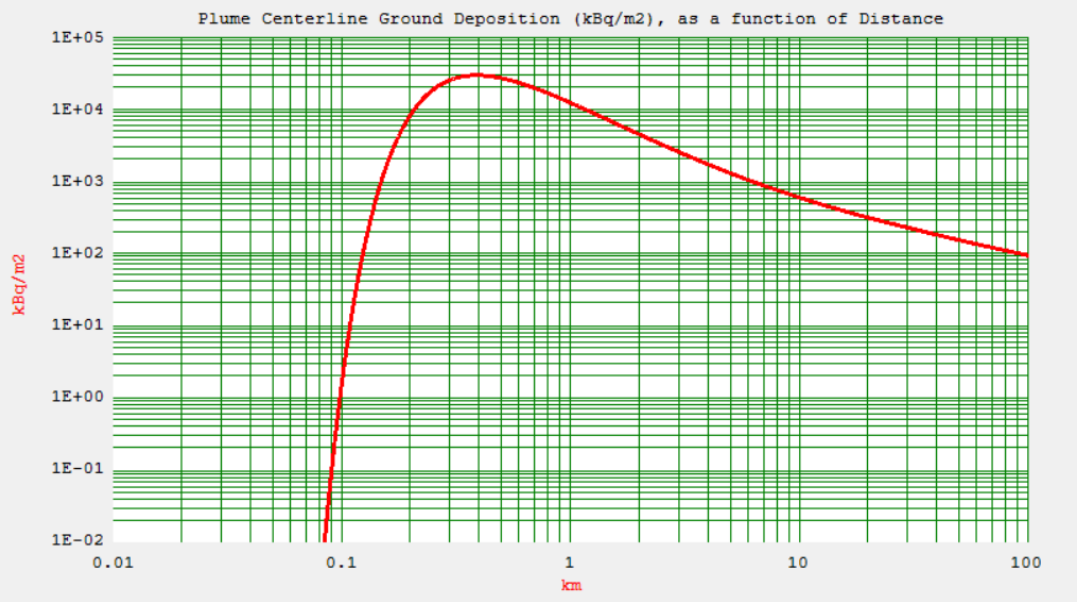

Figure 20 Centerline ground deposition

In the graph, in figure 20, of the results the ground deposition is highlighted as a function of the distance and it is possible to notice how the first values are found at about 100m from the source, with the peak about 400m and a subsequent slow decrease as the kilometers increase. It should be noted that the graph is logarithmic, which allows even more to appreciate the quantities and distances.

5.3 RESRAD implementation data

Given the confirmations on the values that have been provided by HotSpot, it is therefore possible to move on to the actual analysis of the effects of a possible release of tritium in the area considered on the population and on all the pathways that will be considered.

To do this, the RESRAD-Offsite software was used, already analyzed in its basic characteristics.

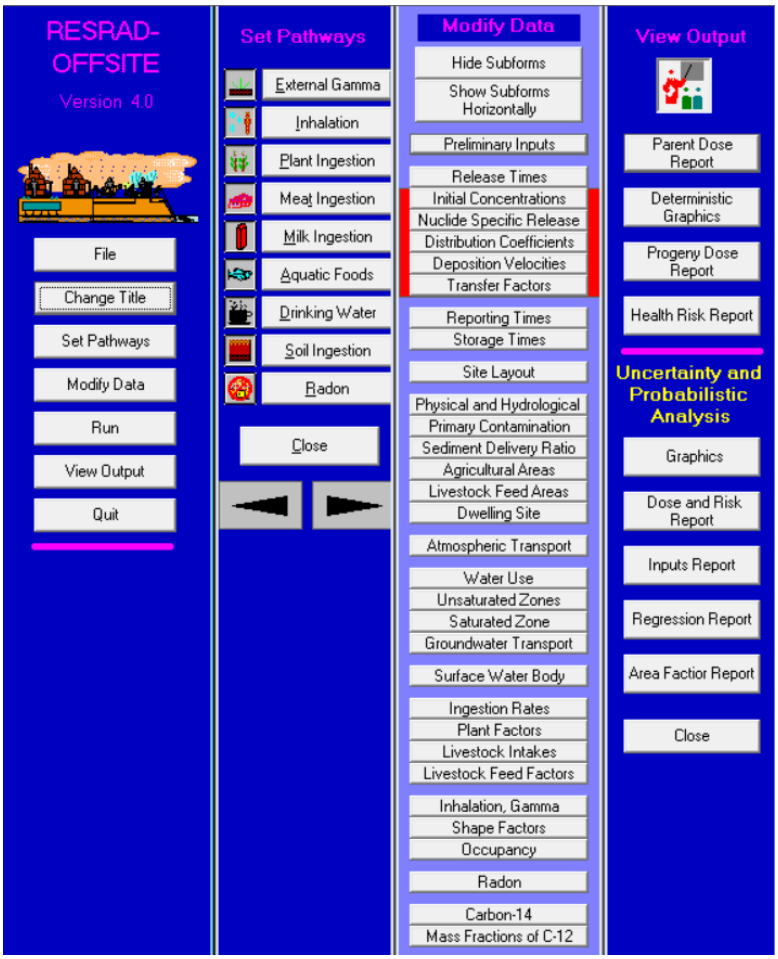

Figure 21 RESRAD general parameters

First of all, the pathways that are considered must be set and in the specifics of my analysis, 4 different situations have been considered selected on the basis of the interaction characteristics of tritium with the different environmental matrices:

1) "drinking water"
2) "ingestion"
3) "external gamma + inhalation"
4) "all pathways"

After selecting the types of pathways, it is necessary to introduce the parameters that characterize the site. Concentration is first introduced and then the site layout is defined by locating target points.
Analyzing them all one by one would be long-winded, the most relevant will be mentioned.
The nuclide concentration given by the quantity and the results provided by HotSpot is 0.3 MBq / g in the contaminated area. The reporting times, that are all the times in which the various parameters are processed and analyzed to arrive at the final results, are 1, 2, 3, 5, 10, 15, 20, 30, 50 years. The display map is the following, with the positioning of the source at the Pilgrim nuclear center and the respective pathways following an analysis of the territory surrounding, making an average of the hectares of land occupied at various distances for each of them. In fact, figure 22 shows the icons of the targets considered to represent the well considered for drinking water, the agricultural area divided into fruit and vegetables for the contamination of the products that will then be consumed by the neighboring population, the breeding area with the considerations on meat and milk, the surface water body, a fundamental parameter of contamination and the houses for direct exposure of the population living in the surrounding area.

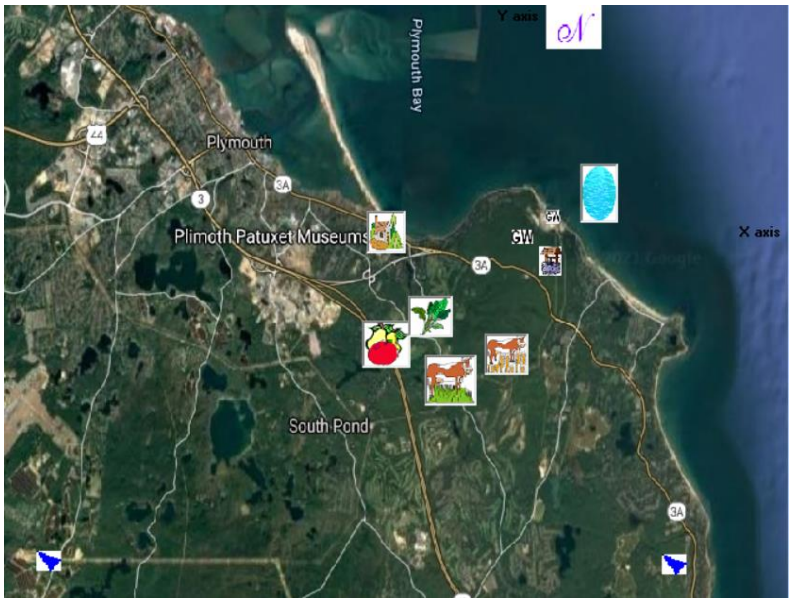

Figure 22 RES-RAD site map

Subsequently it is necessary to enter the data relating to transport in the atmosphere, to do this the software has a data library relating to the main American cities, which made it possible to enter the area of Boston, not far from the Pilgrim site as shown in the figure 23.

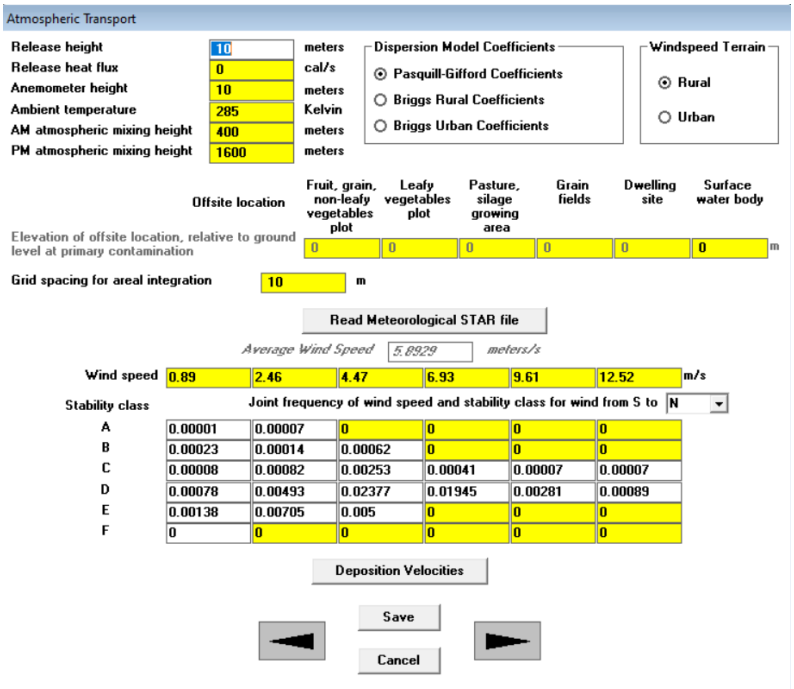

Figure 23 RES-RAD atmospheric transport parameters

In conclusion, it was necessary to enter all the atmospheric transport data associated with each pathway of the American lifestyle and in particular in Massachusetts. The program allows to obtain default data once entered in the specific area on which it want to carry out the analysis. In addition, a cross-check was done with different sites to search for the veracity of the preset values and it turned out that the values correspond to reality. Other very important data concern the consumption rate inherent to the previously considered pathways, examples are included below in table 5.

Precipitation	1.29 m/y
Water use indoor per person	225 L/d
Meat consumption rate	63 kg/y
Drinking water consumption rate	510 L/y
Milk consumption rate	92 L/y
Fruits and vegetables consumption rate	160 Kg/y

Table 5 Massachusset consumption rate

5.4 Dose results

Once the model is implemented, it is possible to estimate the dose to the population, with the run of the program.
The equivalent dose is a dose quantity representing the stochastic health effects of low levels of ionizing radiation on the human body which represents the probability of radiation-induced cancer and genetic damage. It is derived from the physical quantity absorbed dose, but also takes into account the biological effectiveness of the radiation, which is dependent on the radiation type and energy. In the SI system of units, the unit of measure is the sievert (Sv).

It is therefore essential to know and mention the regulatory limits of dose for workers and the population in order to have a subsequent better understanding of the results provided by the specific analysis of the case study considered.
Dose limits recommended by the I.C.R.P. 60: (relating to a population not subjected to radiotherapeutic treatments).

- Professional exposure:
- effective dose limits (cumulative): 20 mSv per year, as an average over defined periods of 5 years, with an annual maximum limit of 50 mSv;
- admissible effective dose limits (annual): 50 mSv per year;
- specific annual dose limits for lens and skin:
  - per crystalline: 150 mSv per year;
  - for skin: 500 mSv per year for the hands and feet, however 500 mSv per year, understood as an average value on 1 cm^2, regardless of the exposed surface.
  - Dose limit for exposure of the fruit of conception: a total of 2 mSv on the abdominal surface.
- Public exposure (annual):
- effective dose limits: 1 mSv per year (in particular cases a higher annual value may be allowed as long as the average dose over 5 years does not exceed 1 mSv per year).
- specific limits recommended for skin and crystalline:
  - 50 mSv per year for the skin, hands and feet, in any case 50 mSv per year, understood as an average value on 1 cm^2, regardless of the exposed surface;
  - 15 mSv per year for the lens.

The above dose limits are not to be confused with LDR or Reference Levels in Diagnostic Radiology. [24]

In this work, four different cases have been analyzed. Each of them refers to the four groups of pathways listed in the previous paragraph.

Now starting from the first group, the results will be analyzed and commented to have a greater understanding and a global idea of the actual damage caused by the release. The goal remains to demonstrate the feasibility of the project and the mechanisms chosen for use, always taking into account the critical issues that may be encountered such as a possible release of tritium caused by a loss of storage systems.

5.4.1 Drinking water

Starting from the least relevant in terms of dose quantity, it can be found the "drinking water". The maximum peak is about $5*10^{-15}$ mSv/yr.

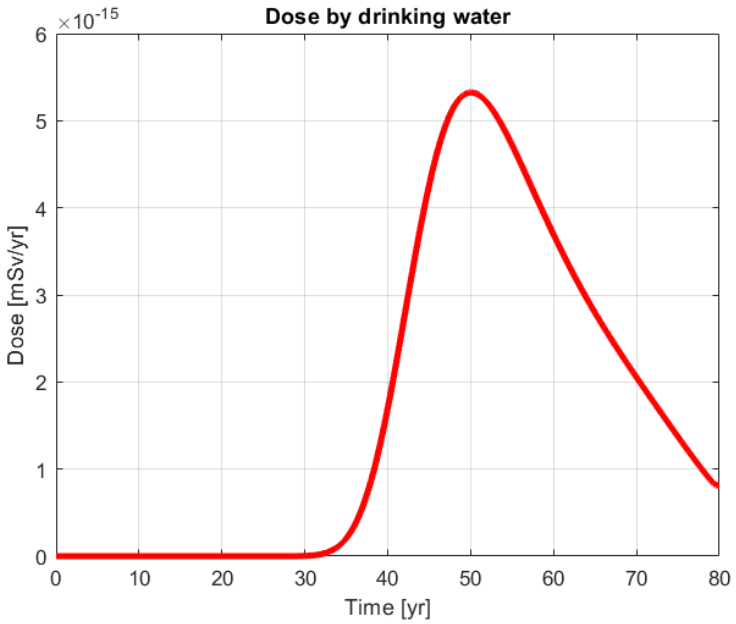

Figure 24 RES-RAD drinking water results

The dose trend presents an initial zone, up to 5 years, stationary at zero due to the time taken from the tritium to react with water. In fact, tritium can substitute one or both hydrogen atoms in the water molecule (H in H2O).
If released as gas, tritium can mix with rain and contaminate drinking water.
Then the curve reaches its peak at 50 and from there begins a rapid descent almost comparable to the ascent to the peak (figure 24).
Given the values in the order of 10^{-15} it's possible to state that the contamination of drinking water is almost zero, in fact the values do not even come close to the limits imposed by the regulations. Consequently, the effects of the release on this pathway are negligible.

5.4.2 Ingestion

Proceeding in increasing quantities of dose it's possible to find the second group that of ingestion which groups 5 pathways internally:
-meat ingestion

-milk ingestion
-plant ingestion
-soil ingestion
-aquatic foods
All of these have as their common denominator the fact that they are ingested by the human body during everyday life.

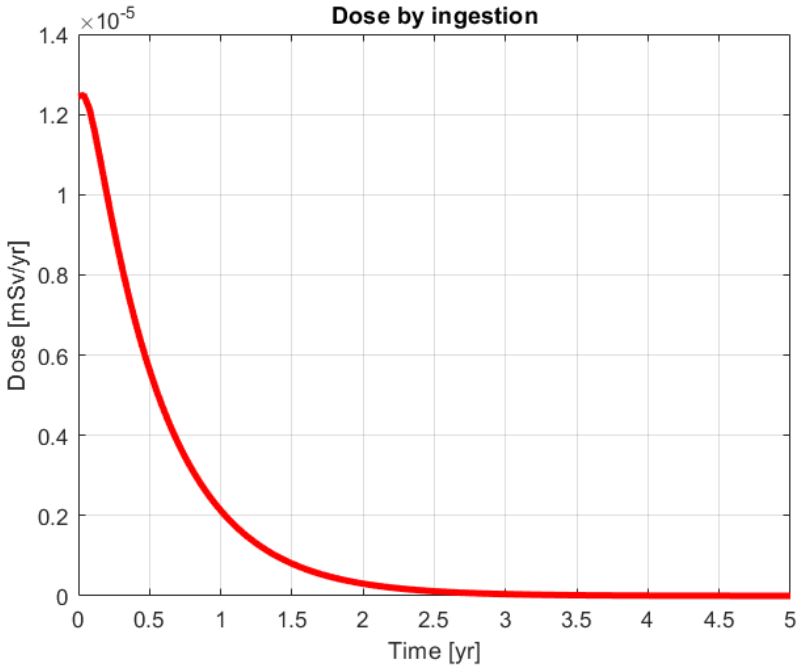

Figure 25 RES-RAD ingestion results

It can be immediately noticed how the dose trend reaches its peak of about 1.2 * 10-5 at the beginning as predictable given the direct link between tritium and the food ingested and then begins an exponential descent reaching negligible values after about 3 years (figure 25).
By focusing more attention on the peak value, it is possible to say that too high numbers are not reached, but not negligible as in the previous case. It is therefore necessary to take this aspect into consideration even with the awareness that the spacing measures of the plant and the containment methods correctly fulfill their purpose. In fact, it can be said that even these values are well below the limits imposed by the regulations.

5.4.3 Gamma rays and inhalation

Moving on to the most relevant contribution, it's possible to face with the pathways of gamma rays and inhalation.
Right from the start it is evident that the values are just below the unit of the mSv, important values and not at all to be underestimated.
Between the two pathways, the bulk is represented by inhalation as unlike all the others analyzed above, it has direct contact with operators, workers and nearby

inhabitants. In fact, as far as inhalation is concerned, an elaboration period is not necessary as for food or water, but the contact occurs directly.

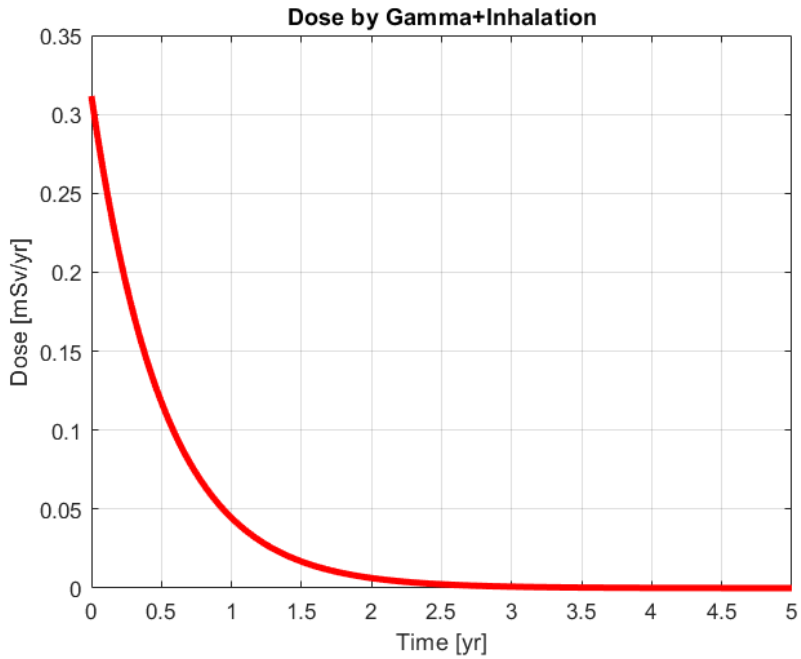

Figure 26 RESRAD gamma+inhalation results

Going into more detail of the graph curve, it can be seen that the trend is very similar to that of ingestion, with the initial peak above 0.3 mSv/yr, a rapid exponential decline in the first 2 years and a slower one in the following years (figure 26).
Here, as never before for the other case studies, it is increasingly necessary to take into consideration the best possible containment strategies because now the dose values obtained are very close to the limits (1mSv/y) and therefore it is essential to keep them under constant control.
Consequently it will be the most important technological challenge from the point of view of radiation protection.

5.4.4 All pathways

Going now to sum all the pathways previously analyzed, it is possible to immediately notice how the trend is almost identical to that generated by gamma plus inhalation since those values represented by far the highest of all with four orders of magnitude more than the ingestion and 14 more than drinking water (figure 27).
Consequently, as regards the trend of the graph and the specific analysis of the values, it is possible to refer to those of the figure 26.

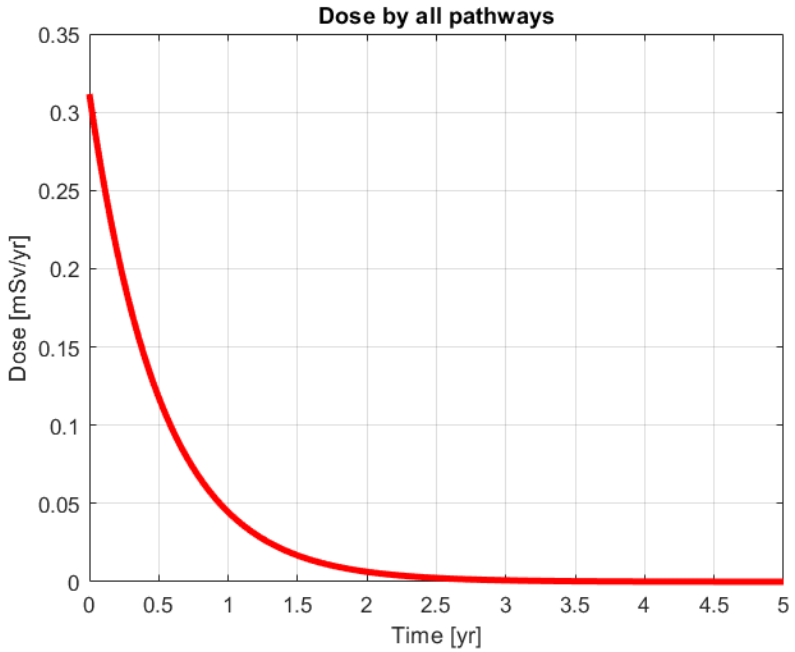

Figure 27 RESRAD all pathways results

In conclusion to this analysis it is necessary to reiterate how it was possible to achieve the objective of radiation protection, remaining below the limits imposed by the regulations. All this makes the study made even more realistic and usable, albeit always taking into account that the site chosen is one of the options, but not yet the definitive one.

Chapter 6

6.1 Conclusions

For the dose calculation to which the population should undergo in case of release of 1 gram of tritium from the ARC site, it made use of calculation programs such as RESRAD and HotSpot. As said the main objectives were to understand the feasibility of the ARC project in terms of possible release of tritium following accidental events. First of all, the design of the reactor was analyzed to understand its main characteristics and then to be able to analyze the correct production of tritium during operations at full capacity. To do this, the principles of nuclear fusion were exploited through two main factors such as TBR and AF. Once these results were obtained, they were compared with the values from the bibliography and from the experience given by ITER and DEMO and these confirmed the values obtained.
Once this step was completed, it was necessary to analyze, study and choose which was the best and most suitable method for ARC as regards the storage of tritium through the Zr-Co or U-shaped beds. Following the analysis of the pros and cons of both technologies and calculations resulted in the former being the most appropriate for ARC.

As a last chapter, the case study of the ARC reactor was analyzed, which envisages a possible location in the vicinity of the city, more specifically in the Pilgrim nuclear station, which already presents several safety measures suitable for the construction of a fusion reactor. The first thing of fundamental importance is certainly the fact that it is already a nuclear site thus already having an adequate safety distance from homes, second thing the proximity to a source of water which in this case is the ocean and finally all the already present nuclear safety measures at the site.
Once the choice of the possible site has been studied, through the use of two software specialized in the analysis of data, specifically first with HOTSPOT, which made it possible to obtain useful values such as ground deposition and quantities of the main areas affected by the release. Subsequently, thanks to these data and others obtained from morphology, meteorology and demography, they were inserted into the other RESRAD software, in order to obtain the fundamental results of the entire analysis such as the dose values obtained from the input of the different types of pathways proposed by the software.

The main result obtained is the demonstration of the possible containment of the equivalent dose values, all below the limits imposed by world regulations.

Having to analyze more specifically, it is possible to state that the group of pathways that includes inhalation and gamma rays is undoubtedly the most relevant in terms of results, reaching values very close to the unit of mSv/yr but still below the limits.

The work done and the results obtained certainly lead to significant and positive future prospects.
Mainly concerning safety issues in case of accidental release of tritium, but secondly also of all other related aspects. Since fusion, unlike fission, does not produce radioactive waste and activation of the components and materials used, or at least in very small quantities, we can consider the release of tritium as one of the worst accidental events that can happen in a nuclear fusion power plant.
With this it can therefore be said that the installation and safety of fusion plants is completely feasible and would not cause significant damage to the population even in the event of accidental or incidental events. The focus is now entirely on the technological side as there are still several areas to study and implement, but the results obtained in recent years are a great push to continue and thus be able to complete the goal of having a fusion plant functioning on as soon as possible.

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