Proposal of a Deuterium-Deuterium fusion / PWR fission hybrid reactor

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Revision B

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The first version (revision A) of this article has been published in the "World Journal of Nuclear Science and Technology" journal: https://www.scirp.org/pdf/wjnst2024144_31090543.pdf

It has been modified by an Erratum published in the WJNST journal [2025, 15, 53-57]: https://www.scirp.org/journal/paperinformation?paperid=141448. This revision B takes into account this Erratum. Vertical dashes in the margin indicate the Revision B modifications.

The differences are on the references 1, 2 and 33 which are, for the previous document, the articles published by the author in the "Energy and Power Engineering" journal and in the "World Journal of Nuclear Science and Technology" journal, whereas, for this document, they are the author articles published in Zenodo (the articles have almost the same content, but the Zenodo articles have a table of contents, hyperlinks and sometimes a different numbering).

Abstract

This article proposes to associate a Deuterium-Deuterium (D-D) fusion reactor with a PWR (fission Pressurized Water Reactor) in a hybrid reactor. Even if the mechanical gain (Q factor) of the D-D fusion reactor is below the unity and consequently consumes more energy than it supplies, due to the high energy amplification factor of the PWR fission reactor, the global yield is widely superior to 1. As the energy supplied by the fusion reactor is relatively low and as the neutrons supplied are mainly issued from D-D fusions (at 2.45 MeV), the problems of heat flux and neutrons damage connected with materials, as with D-T fusion reactors, are reduced. Of course, there is no need to produce Tritium with this D-D fusion reactor. This type of reactor is able to incinerate any mixture of natural Uranium, natural Thorium and depleted Uranium (waste issued from enrichment plants), with natural Thorium being the best choice. No enriched fuel is needed. So this type of reactor could constitute a source of energy for several thousands of years, because it is about 90 more efficient that a standard fission reactor, such as a PWR or a Candu one, by extracting almost completely the energy from the fertile materials U238 and Th232.
For about the fission part, the PWR technology is mature. For about the fusion part, it is based on reasonable hypotheses done on present Stellarators projects.
The working of this reactor is continuous, 24 hours a day. In this paper, it will be targeted a reactor able to provide a net electric power of about 1400 MWe, as a big fission power plant.

Keywords

Fusion Reactor, Fission Reactor, Hybrid Reactor, Nuclear Energy, Deuterium-Deuterium reactor, Deuterium, Racetrack, Stellarator, Power plant, PWR

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1. Introduction

1.1 Goal, presentation and notations used

The goal of this article is to describe a hybrid power plant formed by:

- A Deuterium-Deuterium (D-D) fusion reactor proposed by the author in a previous article [1] which is based on the article [2]. This fusion reactor uses a magnetic confinement and a plasma heating system.

  The standard way to heat the plasma, so as to stabilize the plasma temperature, is to use devices such as neutral atoms/molecules injection, but also radio frequencies (ECRH or ICRH) and possibly magnetic compression at the heating beginning, etc. Moreover, the injection of pellets of Deuterium ice permits to feed the reactor. The set of plasma heating systems is symbolized here by two opposite beams of D+ ions and electrons injected by ions and electrons beam guns, even if in reality these beams could not reach the plasma core as the magnetic confinement would prevent it. However, for the calculations, it will be considered that it is possible, which would not change the results, but simplify the analysis. Moreover, two opposite beams permit to avoid a net plasma current.

  > Note 1: the injection of neutral atoms/molecules at high energy would be more or less equivalent because these particles are mainly ionized at the center of the plasma and give their kinetic energy in excess to the plasma. The advantage is that these neutral particles are not confined by the magnetic field, so they can enter into the plasma core. However, in that case, the kinetic energy is mainly carried by the sole D+ nuclei, not by the electrons.
  >
  > Note 2: the plasma neutrality is supposed to be controlled, through a Divertor electronic or ionic sheath, implicitly taking into account the particle losses and the particles injected. This complex problem is not addressed in this paper.

  The two opposite beams (symbolized here by D+/electrons particles, but neutrals particles in reality), initially directed axially, circulate inside a figure of “0” configuration, also called a Stellarator “racetrack”. The global injected current is nil. This reactor would produce nuclear fusions with a mechanical gain (Q), i.e. fusion power / mechanical injection power, depending on the pipe radius (Rp) superior to 1 in [1] but inferior to 1 for this hybrid reactor. For example, at Rp=2 m, Q=0.184, versus Q>10 for a D-T Stellarator of the same radius, as for example the Helias 5-B project.

- A PWR fission reactor, used in a sub-critical state. Note that a PWR is normally critical to work. This PWR is supplied in neutrons by the fusion reactor. Thanks to fission reactions, the PWR amplifies the power carried by the 2.45 and 14.06 MeV fusion neutrons by a factor of around 130 depending on the configuration chosen. Thanks to this gain, the fusion reactor has not to be powerful, as it is just a neutron generator, the power mainly coming from the PWR. As a sub-critical reactor, this PWR can “incinerate” (i.e. “burn” in a nuclear way) all the fertile material from natural Uranium or natural Thorium (i.e. U238 or Th232). It works as a sub-critical breeder reactor. Note that with natural Thorium as fuel, the plant consumes electricity for the first 18 years (see §5.3) before becoming a powerful electricity generator.

These fusion and fission reactors are completely independent of one another, i.e. an evolution on the fission reactor, as the reactivity or the temperature, has no impact on the fusion reactor.

It is reminded that Deuterium (D) and Tritium (T) are hydrogen isotopes comprising, besides one proton, either one neutron (Deuterium) or two neutrons (Tritium).

The problems of cryogenic systems, ultra high vacuum, particles diversion and neutral particles pumping in the “Divertor” (to “clear” plasma), radiation hygiene, possible instabilities and way to realize toroidal and poloidal fields are not addressed in this paper.

This article is only concerned with the fusion and fission aspects, at the level of principles, the physics used being relatively simple.

The main goal of this article is to propose a physical model of this hybrid reactor. A small program called "D_D_PWR_hybrid_reactor" (§1.5) based on the model developed in [1] and in this article, is proposed, with its Delphi 6 source code. Thanks to this model, any D-D/PWR hybrid reactor can be roughly designed.

It is obvious that the reactor presented in this paper would be rather large for a practical realization, i.e. 200 m long according to the figure 3. To reduce the size of such a reactor, one can increase the Beta factor (§3.2) if possible or, more slightly, the maximum K_eff (§4.5).

Interest of such hybrid reactor

The interest of such reactor is that the source of energy could last a very long time. Below is explained why. First about the Deuterium, a m^3 of sea water contains 32.4 g of Deuterium (D2). So the reserve of Deuterium would last millions (see billions) of years (see [1] for details).

Besides the important stock of depleted Uranium, there are two natural materials that can be incinerated in this hybrid reactor: the natural Uranium and the natural Thorium.
Moreover, the possible stock of enriched Uranium and the stock of Plutonium issued from reprocessing plants for nuclear waste could be slowly incinerated (for example, mixed with depleted Uranium or with natural Thorium), but they are not taken into account in this document as the available quantities of these materials are not important.

The conventional Uranium reserve, at less than 130$/kg in 2021, is equal to 6,078,500 tons (from [3]). Moreover, 1.2 millions of tons of depleted Uranium (at about 0.3% of U235) are also available (from [3]). The Thorium reserve, at less than 80$/kg in 2016, is equal to 6,355,000 tons (from [4]).

Using the proposed hybrid reactor in its default configuration, it is found that the net electrical energy produced by one ton of:

- Natural Uranium (in the form of UO2) is equal to 1.45E16 J/t.
- Depleted Uranium (in the form of UO2) is equal to 1.44E16 J/t.
- Natural Thorium (in the form of ThO2) is equal to 1.65E16 J/t.

The world electric energy consumption is equal to 29471 TWh in 2023, according to [5] page 15, with 29471 TWh equivalent to 1.0609E20 J.

Now let's suppose that all the electricity produced in the world be supplied by plants using the type of hybrid reactor proposed in this paper. In this case and neglecting the O2, the consumption of electricity would be covered for:

- 832 years by Natural Uranium,
- 162 years by depleted Uranium,
- 990 years by Natural Thorium.

The total makes 1984 years. Now, it must be taken into account that the conventional and unconventional reserves of natural Uranium are much larger than the value given above. For example, see the reference [6] page 94. The time during which Uranium and Thorium could supply all the world electricity would rather be about 8900 years. This is calculated without taking into account the unconventional reserves of Thorium. Moreover, a part of the heat produced by these reactors could be used without being transformed into electricity, for heating networks, desalination plants, etc.

Comparison with the Candu and PWR fission reactors

The Candu reactor uses natural Uranium up to an average burn-up of 7.5 GWd/t. A standard PWR uses enriched Uranium at 3.4% up to a burn-up of about 42 GWd/t. This ton of enriched Uranium is obtained from 7 tons of natural Uranium at 0.7 % of U235, leaving 6 tons of depleted Uranium at 0.25% of U235 as waste. Reported to the ton of natural Uranium, the burn-up is equal to 42/7=6 GWd/t. Let's say that thanks to certain improvements and the use of MOX fuel, the average burn-up grows up to 7.5 GWd/t as for the Candu.

In the reactor object of this article, the burn-up of the consumed fuel is very high: 723 GWd/t for natural Uranium and 819 GWd/t for natural Thorium, so close to the maximum possible. Hence this reactor is 723/7.5=96 times more efficient that a standard reactor (PWR or Candu). Now these values of burn-up must be considered as a bit optimistic because the fuel evolution taken into account (§4.12) is simplified so the efficiency gain is certainly a bit lower, but not much below 96.

Notations and units

In a formula, the × and / operations take precedence over the + and − operations, as for example:
$A \times B + C \times D = (A \times B) + (C \times D)$.

The comma is a figure separator. For example: “100,000” means a hundred thousand.

SI units, multiples and sub-multiples (cm particularly) are only used, with the exception of the “eV” (“electronVolt”), which is a unit of energy quantity used in the particles domain, i.e. 1 eV is equivalent to 1.60219E-19 J. It is the potential energy of a single charge submitted to a potential of 1 V.

Note that “We” means “W” (watt) for electrical power.

The fuel burn-up is expressed in “MWd/t”, i.e. “Thermal energy in MW for one day per ton of fuel”. 1 MWd is equivalent to 1E6x3600x24=8.64E10 J. It is sometimes used “GWd/t” rather than “MWd/t”, with 1 GWd/t=1000 MWd/t.

1.2 Quick Hybrid reactor historic and comparison with the option explored in this document

The hybrid fusion-fission reactor concept dates from the 1950s. At the end of the 70’s, a hybrid D-T tokamak/fission reactor was envisaged but the provisional cost was considered dissuasive.
For example, the problem of a blanket providing fissions and breeding Tritium was (and remains) complex to solve: if a neutron is used to breed Tritium via a reaction with Li6, this neutron is not used for fission, and reversely. In addition, Lithium must be separated from water due to an exothermic reaction between both.
Fusion devices using a Deuterium beam at high energy (1.5 MeV) were also envisaged as a source of neutrons for a fission reactor.

In 1979, Hans A. Bethe defended in [7] this type of hybrid D-T tokamak/fission reactor.

In 1993, Carlos Rubbia and his team proposed in [8] to use, instead of a D-T fusion reactor, a beam of protons at high energy (1 GeV) hitting a heavy metallic target (lead for example) which produces, by spallation, the necessary neutrons to breed the fissile materials.

Since then many configurations have been proposed, including “Stellarators as fusion-fission reactor candidates” in [9] page 195, for example.

In all cases, as the fission reactor is in a sub-critical configuration, hybrid reactors can be called “energy amplifiers” because the fission reactor multiplies the number of neutrons generated by the fusion reactor or by spallation and generates fissions at about 200 MeV with neutrons at high energy (14.06 MeV for the D-T reactors). If the source of neutrons stops, the fissions generated in the fissile materials quickly stop also. However, as in a standard fission reactor, after a stop the fission reactor must be cooled due to the residual heat produced by fission products.

About experimentation, the T-15 MD tokamak will explore the feasibility of a hybrid fusion/fission model. See [10].

The option explored in this article is the use, rather than a D-T fusion reactor, of a D-D fusion reactor using a “racetrack” Stellarator which neutrons breed a sub-critical PWR. The advantage is a relative simplicity as there is no need for Tritium. Moreover, the proportion of high energy neutrons at 14.06 MeV is low. So even if the mechanical gain Q of a D-D reactor is low and even if the reactivity of a PWR is low with natural Uranium or Thorium, the energy amplification remains sufficient, due to the generation of relatively low energy neutrons (2.45 MeV) for thermal fissions at about 200 MeV.

1.3 Method of the work

Step 1: the global working of the proposed D-D/PWR hybrid reactor is first described, at the level of principle, in §2.

Step 2: in §3 and §4, the fusion and the fission reactors working are modeled using classical formulas of physics. The previous fusion model developed in [1] is taken into account in the program, but it is not described, except for its modifications in §3. Only the fission part is described in this article (§4). The global working of the reactor is shown in figure 4 (§2.1).

Step 3: once the physical models are achieved, they will be used, in §5, to estimate this reactor with several types of fuel. A discussion about results, points to deepen, improvements and safety aspects follows. In §6, conclusions are drawn.

1.4 List of the main variables and acronyms used in this article

Below are the main variables used all along the article (local variables are not listed)

Beta	Diamagnetism factor
B	Toroidal magnetic field (T)
E_equi	Equilibrium energy of the plasma in eV
E_inj	Energy of the injected Deuterium ions and electrons in eV
K	Neutrons multiplication factor (in K_infinite and K_eff)
nD	Deuterium ions density (number of Deuterium ions per m^3)
ne	Electrons density (number of electrons per m^3)
pht	Per hundred thousand, i.e. in 1/100,000
Q	Mechanical gain (i.e. fusion power / mechanical injection power), without dimension
Rp	Interior pipe radius (m) of the fusion reactor

Below are the acronyms used:

"D-D" for "Deuterium-Deuterium"
"D-T" for "Deuterium-Tritium"
"PWR" for "Pressurized Water Reactor"
"UHV" for "Ultra High Vacuum"

1.5 "D_D_PWR_hybrid_reactor" program based on the fusion and fission physical models

It is proposed the program called "D_D_PWR_hybrid_reactor" V2.0 which implements the physical model described in this article (fission reactor), plus the one described in [1] (fusion reactor). The executable program with its Delphi 6 source code can be downloaded from this direct link: http://f6cte.free.fr/D_D_PWR_hybrid_reactor.zip. It is enough to paste this address in your Internet browser. Download the file. Create a folder (C:\D_D_PWR_hybrid_reactor for example), unzip the D_D_PWR_hybrid_reactor.zip file in it and then start: C:\D_D_PWR_hybrid_reactor\D_D_PWR_hybrid_reactor.exe.

In case of failure of this WEB address, this program will be available on the Zenodo WEB open repository, by searching with the title of this article.
Below in figure 1 is a snapshot of the program running the default configuration, the fuel being natural Uranium. It also appears the end of the results displayed on the black DOS window.

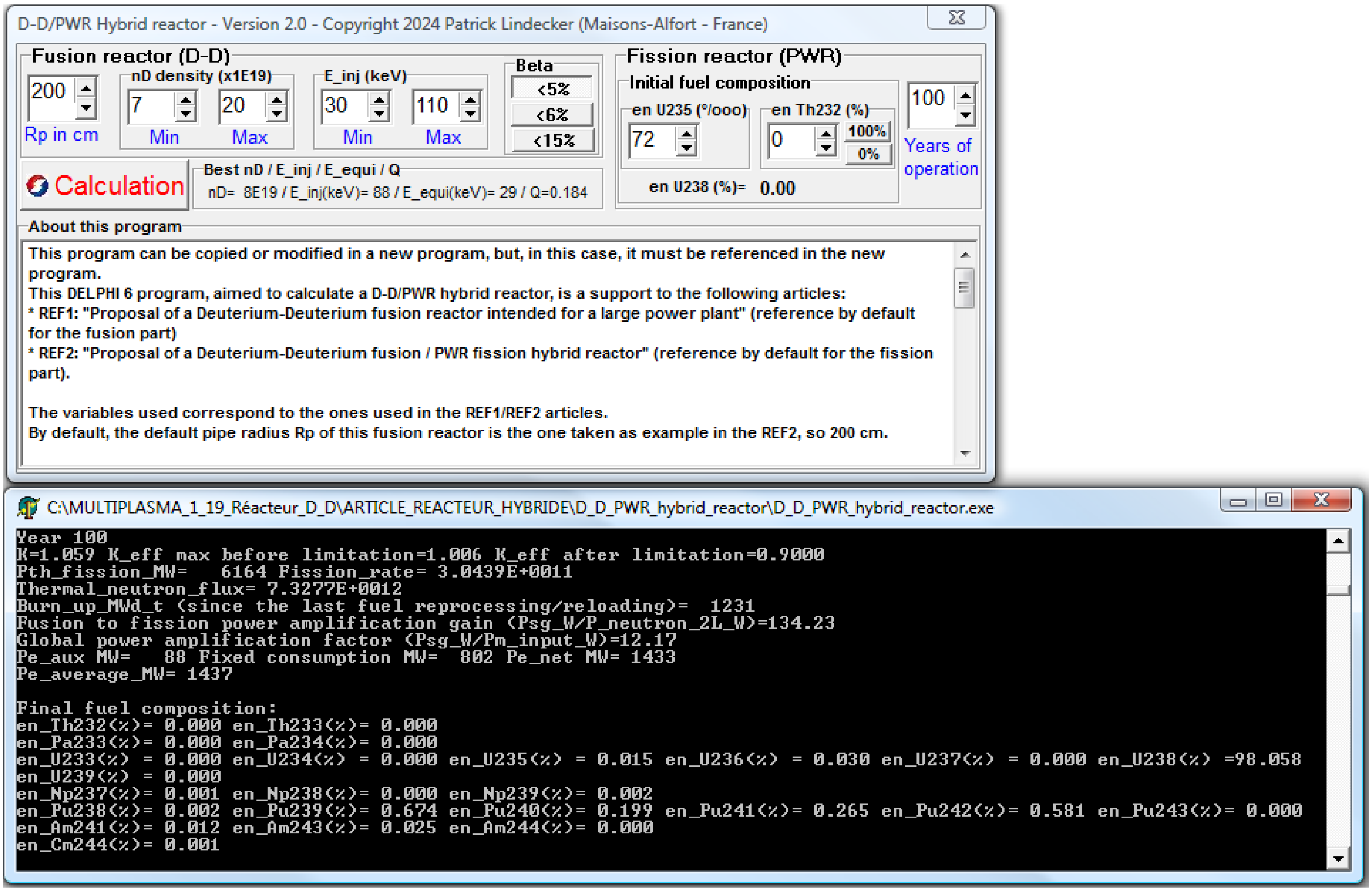


Figure 1. D-D/PWR Hybrid reactor V.2.0 program snapshot running its default configuration

2. Description of the hybrid reactor

2.1 Generalities

It is composed of a fusion reactor all along the axis. It is surrounded by a fission reactor along both straight parts only, i.e. between A and D and B and C in figure 2.

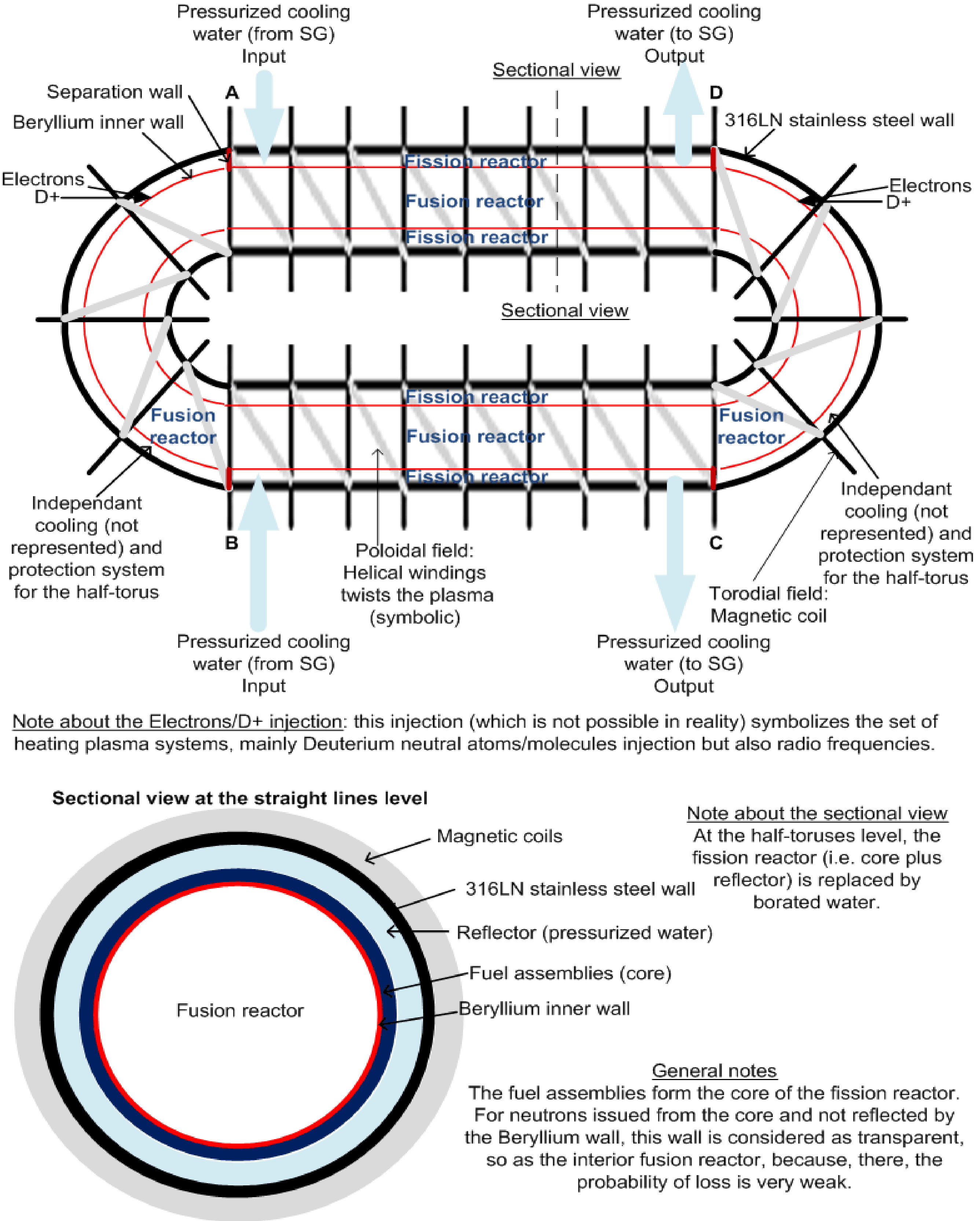


Figure 2. D-D/PWR fusion reactor principle diagram

About the half-toruses at each end of the reactor (A to B and D to C, in figure 2), the two volumes between the beryllium wall and the 316LN wall don't contain any fission reactor. About these two volumes:

- They are mechanically separated from the contiguous volumes of the straight parts by a separation wall (at the level of A, B, C and D).
- They are cooled independently from the straight parts, the cooling fluid being, for example, borated water. See also the second paragraph of §5.8 for about the way to take profit of the heat produced.
- They don't need to be pressurized, contrary to the fission parts.

About both half-toruses at the level of the fusion part:

- They support the divertor and the main fusion instrumentation.
- The necessary magnetic field is complex and the pipe section is not circular (see [11]).
- They can be considered as a sole torus as described for example in [11] and will not be described in this article. Only their predictable performance, in terms of energy loss and confinement time, is taken into account.

Reversely, the necessary magnetic field for the straight parts is supposed to be simpler and to permit a circular pipe section.

The fusion reactor is described in [1]. D+ ions and electrons are supposed to be directly injected, at the same energy, with elevated currents, up to the moment when the currents circulating in the figure of "0" reach their nominal values (the global current being nil). In permanent working, the electrons and the D+ ions are supposed to be injected (at E_inj) at a rate that allows to cover losses and fusions, so as to keep the beam neutral. The working of such a D-D fusion reactor is continuous, 24 hours a day.

> Note: in this document, the direct injection of D+ ions and electrons, which is not physically possible, is a simple way to take into account the necessary plasma heating. See §1.1.

In figure 2, the principle diagram of this figure of "0" hybrid reactor is displayed. On this diagram it appears stocky for representation necessity, but it is rather long and narrow: the overall height of the fusion reactor (H in figure 3) is, in fact, 16.71 times larger than the width (W in figure 3).

Still in figure 2, note that the thick 316LN exterior wall around the straight pipes (between A and D and B and C) corresponds to the metallic protection wall between the magnetic coils and the pressurized water. For a PWR, this wall would be called the "Reactor vessel body". Note also that, to simplify, the core barrel around the fuel assemblies is not taken into account.
At the level of the half-toruses, the 316LN exterior wall, jointly with the borated water, must provide the protection of the magnetic coils against heat and neutrons. Its thickness is supposed, a priori, to be the same as its thickness at the level of the straight parts, so 16.9 cm applies to the default configuration.

The geometry of the fusion reactor is detailed in figure 3.

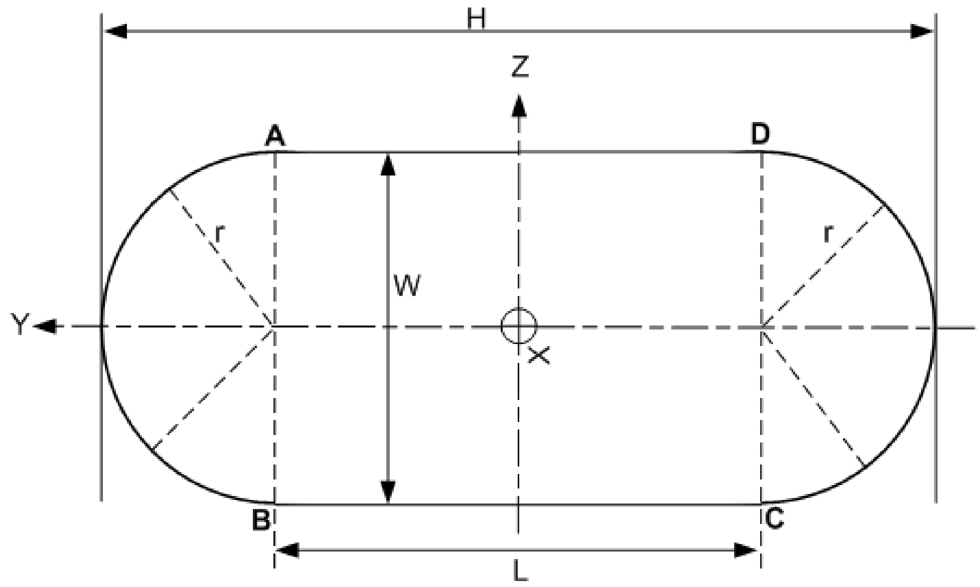


Figure 3. Geometry of the D-D fusion reactor on the YZ plane, along the axis of the reactor

- $H = L + 2 \times r$, H being the overall length of the reactor (between axes), with $L = 30 \times \pi \times Rp$
- $W = 2 \times r$, W being the reactor width (between axes) with $r = 3 \times Rp$

So $H = [(30 \times \pi) + 6] \times Rp$, $W = 6 \times Rp$ and $\frac{H}{W} = (5 \times \pi) + 1 = 16.71$

For the default configuration, Rp=2 m, r=6 m, L=188.5 m, W=12 m and H=200.5 m.

The figure 4 shows, in a general way, how the hybrid reactor energy balance is taken into account, relatively to the power plant.

The variables referred to the fusion reactor (Poutput, Pneu, PlostFi, PlostReactor, Pra, Q) are described in the Appendix B of [1].

The global fusion power (Pth_fusion_W calculated in §3.4) is generated by the half-toruses and the straight lines. The volume of the straight lines being equal to 60/66=91% of the fusion volume, it will be supposed that 90% of Pth_fusion_W will reach the fission reactor and finally the water/steam circuit via the steam generators, so:

$Psg\_fusion\_W = 0.9 \times Pth\_fusion\_W$ (Equation 2.1)

If the heat power generated by the fission reactor is called Pth_fission_W (calculated in §4.6), the global power supplied to the steam generators will be equal to $Psg\_W = Psg\_fusion\_W + Pth\_fission\_W$ (Equation 2.2)

Note that the mechanical power provided by the primary coolant pumps (figure 4) to the primary coolant is neglected in the energy balance. This power is normally added to the thermal power supplied to the steam generators (Psg_W).

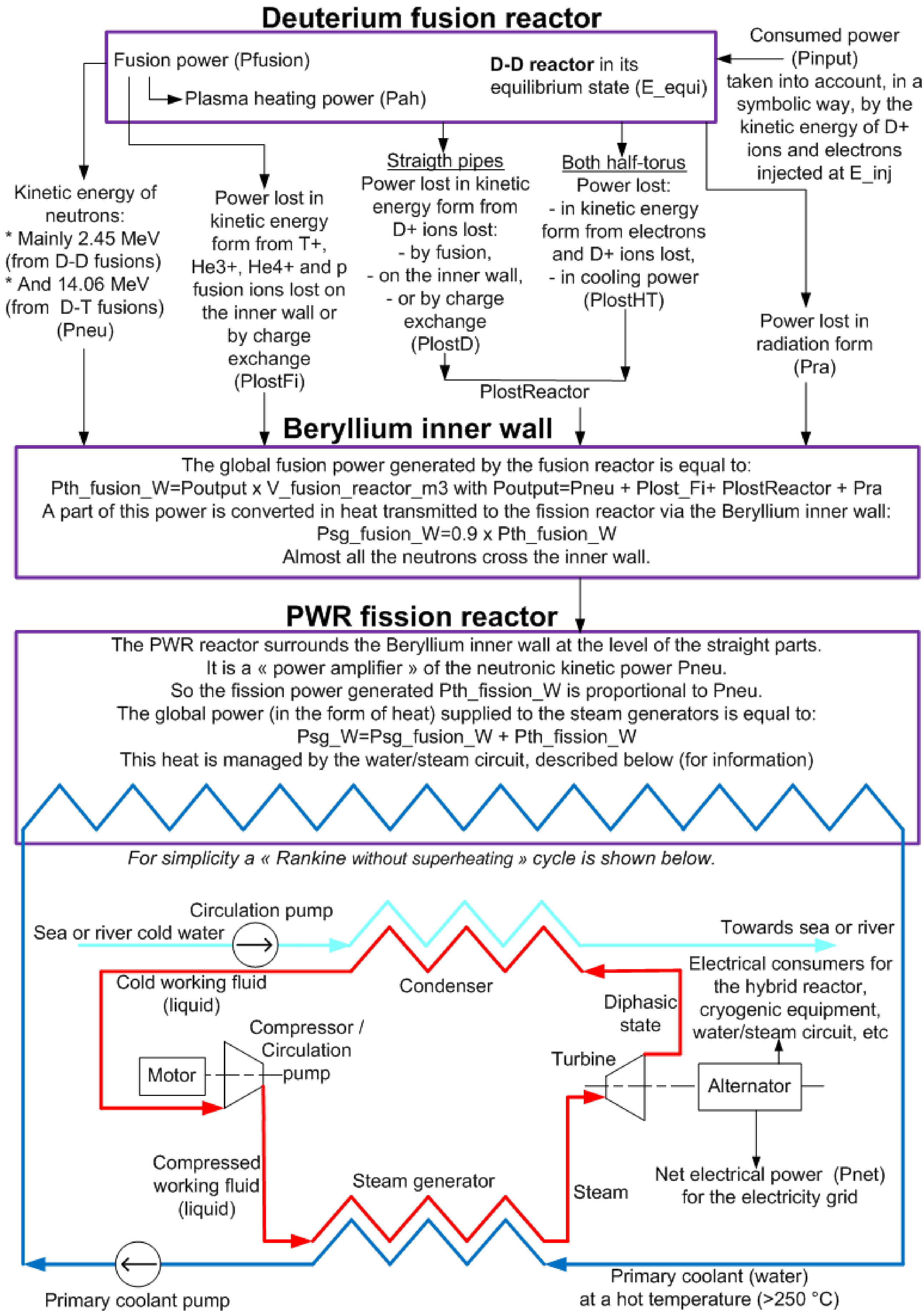


Figure 4. D-D/PWR hybrid reactor energy balance

2.2 Presentation and working of the fusion reactor - Why this type of fusion reactor

Presentation and working of the fusion reactor (see [1] for details))
From an initial rectilinear movement, the behavior of the injected particles (D+ ions / electrons) (figure 2) quickly becomes isotropic before thermalization. Note that in the real case of injected neutral Deuterium particles, these ones are quickly ionized in D+ ions and electrons.
The Coulomb collisions between D+ ions and electrons permit a permanent exchange of energy between these particles. Plasma is heated, for a part, by fusion products: T+, p, He3+ and He4+ ions and maintained at an equilibrium energy E_equi. Replacement particles (D+ ions / electrons) are injected at E_inj to replace the lost particles and to heat the plasma.

> Note that in the real case of injected neutral particles, the kinetic energy of these particles is mainly carried by atomic nuclei (D+) and not by electrons, due to the difference of masses, the speed being the same. So this lack of energy might be compensated either by a higher injected energy of the neutral particles, i.e. about twice the kinetic energy E_inj or by a radio-frequencies heating.

The toroidal magnetic field (B) must be axial relatively to the pipe, and maximum to confine particles (electrons + ions). The present industrial maximum B limit for superconducting coils is 5 T (Tesla). So this 5 T field will be supposed to be the default value.
At fusion densities, a poloidal field is indispensable to limit the particles shift inside loops particularly for the half-torus at each end of the reactor.

The half-torus at each end of the reactor will only be considered in terms of rate of energy loss, loss which is more important than on straight pipes.

There are, mainly, primary fusions between Deuterium nuclei, but also secondary fusions between Deuterium and Tritium nuclei and between Deuterium and Helium 3, as specified below.

D+ + D+ ->T+ (+1.01 MeV) + p (+3.03 MeV) (at 50%)

D+ + D+ ->He3+ (+0.82 MeV) + n (+2.45 MeV) (at 50%)

D+ + T+ ->He4+ (+3.52 MeV) + n (+14.06 MeV)

D+ + He3+ ->He4+ (+ 3.67 MeV) + p (+ 14.67 MeV) (aneutronic fusion).
Note that at an equilibrium energy of about 29 keV, the Deuterium/Helium 3 fusions are very rare.

In figure 5, it is displayed the reactivities used in this paper, for:

- The global D-D fusion reaction from [12].
- The D-T fusion reaction from [12].
- The D-He3 fusion reaction from [13].

> Note: the reactivity, written “<σ x w>”, is the integration of the product of the fusion cross-section σ by the relative speed w between particles over all the energies distributed according to the Maxwell-Boltzmann distribution, i.e. for a thermalized plasma.

Note that, in general, the reactivities are displayed with the abscissa in temperature (in keV) and not in energy (in keV). The relationship is Energy (keV) = 1.5 x Temperature (keV)

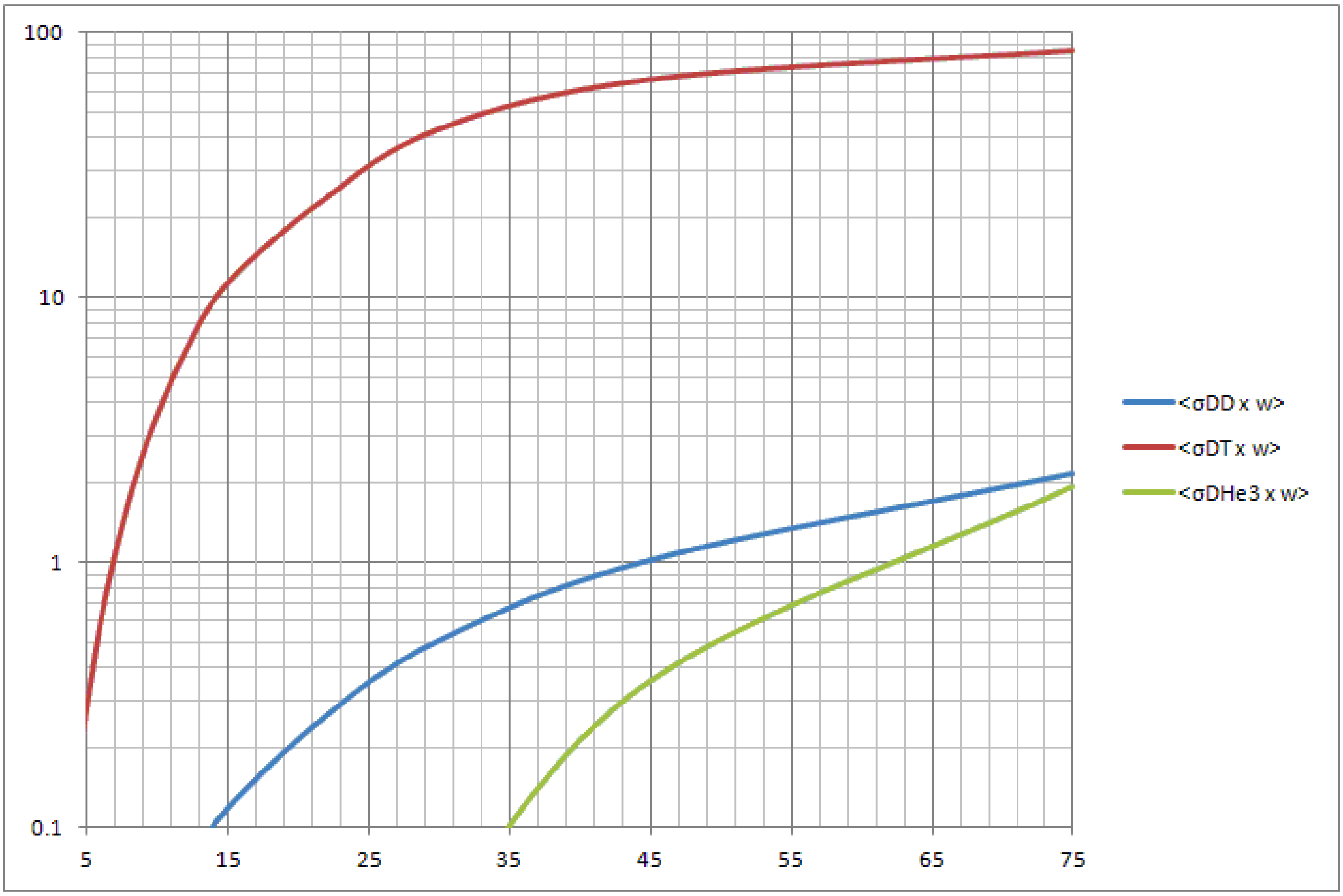


Figure 5. Reactivities of the D-D / D-T / D-He3 fusions.
The abscissa is the equilibrium energy (E_equi) in keV and the ordinate is the reactivity in m^3/s x 1E-23

Contrary to the fission neutrons which follow an energy spectrum, fusion neutrons have discrete values (2.45 MeV and 14.06 MeV). Both values can be exactly determined considering:

- The difference of binding energies from the fusion interactions.
- The respective masses of both fusion products for a given fusion interaction, to share the energy released and to conserve the global momentum.

Why this type of fusion reactor

It could be proposed a D-T reactor (as in [2]), as this one is able to supply energy with a mechanical gain superior to 10. But here the fission reactor amplifies the neutron power by a factor of about 130 (depending on the configuration), so there is no need to look for high performance of the fusion reactor.

The D-T reactor is complex due to the necessity to supply Tritium, which does not exist in nature.
The neutrons generated by the D-T reactor are mainly of 14.06 MeV energy, which limits the energy amplification of the fission reactor compared to the 2.45 MeV energy from D-D fusions. Moreover neutrons of 14.06 MeV energy drastically increase problem on materials. So a D-D reactor is preferable.

Here the D-D reactor is the one described in [1] and modified in §3, working in a reasonable domain, i.e. with a Beta factor (cf. §3.2) equal or inferior to 0.05 which is the present limit for Stellarators, see [14] and [15]. But the mechanical gain Q is inferior to 1.

Compared to a D-T reactor, the performance being weak compared to a D-T reactor, the neutrons flow and the average heat flow are relatively weak. Moreover, in the default configuration, only 26% of the neutrons have a 14.06 MeV energy. So relatively to the Beryllium wall, the mechanical resistance in front of such neutron flow is not critical. The cooling of the Beryllium wall will not be a problem, the heat flux being widely

inferior to 1 MW/m^2, i.e. about 85 kW/m^2 at the straight parts level and about 420 kW/m^2 at the half-toruses level.
Now it will be more complicated at the level of the Divertor where the heat flow is locally more important. However, it would be desirable to avoid Tungsten for the Divertor and to keep on with Beryllium, due to the possible erosion of Tungsten which has at a high atomic number that would cause a big loss by radiation.

The proposed D-D fusion reactor is probably feasible with the present technology because it is much simpler than the D-T Stellarators projects such as the Helias one ([11]), as no Tritium must be supplied and the thermal and neutrons constraints are minimal (see §5.5).

2.3 Presentation and working of the fission reactor - Why this type of fission reactor
Presentation and working of the fission reactor
The fusion neutrons (mainly at 2.45 MeV but also at 14.06 MeV for a small part) issued from the straight pipes (see figure 2) of the fusion reactor, cross the beryllium inner wall (5 cm thickness) with very little loss. This flow of neutrons is amplified by the sub-critical fission reactions in the fuel (about 12 cm thickness of fuel rods). The amplification of energy is partly due to the ratio between the fission energy (about 200 MeV) and the neutron average kinetic power (about 5.4 MeV). The reflector is a layer of 20 cm thickness of water (about the thickness used on PWRs) which goal is to limit the neutrons leak and to slow down neutrons.
The 316LN non-magnetic stainless steel wall, with a thickness of 16.9 cm, withstands the pressure (69 bar gauge) and protects the magnetic coils against fission neutrons.
The pressurized light water serves as a neutron decelerator, reflector and cooling fluid, transporting the heat produced in the fission reactor to the steam generators (see figure 4).

Why this type of fission reactor
The main goal here is to have the most compact fission reactor so as to limit the mass of this reactor and the dimension of the super-conducting coils, even at the price of a small loss of reactivity. Moreover this reactor must be able to incinerate natural Uranium and Thorium.

The heavy water or graphite moderated reactors have a very good reactivity due to an excellent moderation but they are very large and heavy.

The BWR (Boiling Water Reactor) uses light water as PWR but it is larger than the PWR due to a lower moderation power of the boiling water.

The FNR (fast–neutron reactor) is compact but it works on the fast spectrum of neutrons. So with natural Uranium or Thorium, its reactivity would be too low.

The best compromise solution for a fission reactor is the PWR, as it is compact and disposes of a reasonable reactivity. This is due to the light water moderator which moderates quickly but with more loss than heavy water. Moreover, the PWR is the most common fission reactor in the world and its technology is mastered for a long time.

3. Physical model of the D-D fusion reactor

3.1 Introduction
The physical model has been described in [1]. One of the differences between the D-D reactor, as described in [1], and the D-D reactor used here is that the Beta factor is now limited to a realistic value. Consequently, the working conditions are very different. In [1], it was possible to work with an equilibrium energy superior to 100 keV, the limit on Beta being equal to 0.5.
With a realistic Beta, the equilibrium energy is now around 29 keV, so it is not much above the value for a Tokamak or a Stellarator, i.e. around 15 keV.
Under these conditions, the mechanical gain Q for a Tokamak (ITER for example) or a Stellarator (Helias project for example) using D-T fusion, is expected to be equal or superior to 10. For a Stellarator using D-D fusion the mechanical gain is expected to be widely inferior to 1, because the fusion reactivity of the D-D fusion is low compared to the D-T one (see figure 5).

3.2 Beta factor considered

It is reminded the Beta factor as determined in the Appendix A of [1].

$$Beta = \frac{5.369E-25 \times ne \times E\_equi(eV)}{B^2}$$ (equation 3.1)

With E_equi the equilibrium energy of the plasma in eV, ne the electrons density (number of electrons per m^3) and B the toroidal magnetic field (T) in the fusion reactor.

The limit of Beta was 0.5 for the D-D reactor described in [1], which is not feasible for the present Stellarators, as the present actual limit is between 0.05 and 0.06 ([15]).
So it will be considered this value of Beta=0.05 as the default maximum value for the calculation of the D-D fusion reactor.
Simulations with Beta=0.06 and with Beta=0.15, this last value being considered as a possible target for the future according to [14], are also proposed by the "D_D_PWR_hybrid_reactor" program (§1.5).

As said in [14], if the improvement of the toroidal magnetic field seems difficult above the present values (5 or 6 T), the improvement of Beta above 0.05 seems promising. For example, from the default configuration, with Beta=5%: Q=0.184 and the average electric power is equal to 1437 MWe, whereas with Beta =6%: Q=0.215 (+17%) and the average electric power is equal to 2223 MWe (+55%). Moreover, with Beta=6%, for the same electric power of 1437 MWe, the radius only needs to be equal to 178 cm instead of 200 cm, which gives -30% in terms of volume.

Here is another example. With a maximum Beta equal to 0.15, the minimum size, for almost the same average net power (1381 MWe) is half of the size of the default configuration, i.e. a radius of Rp=1 m instead of Rp=2m. Note that in these conditions, with a maximum K_eff=0.94 instead of 0.9 (§4.5), the average net power would be about double (2779 MWe).

3.3 Energy confinement time Tct (s) in the half-toruses

The equilibrium energy (29 keV, so T=19 keV) is not so far away from the one of Helias Stellarators (15 keV maximum on [11] page 19, for example). So there is no need to apply special scaling formulas (for the half-toruses) to take into account a high equilibrium energy (i.e. >100 keV): in [1] §2.2.8, it was used the Kovrizhnikh (or the Bohm) scaling formula to estimate Tct.

For the present Stellarators, it is in general used the ISS04 scaling ([15] for example):

$$Tct = f_{ren} \times 0.134 \times a^{2.28} \times R^{0.64} \times P^{-0.61} \times ne^{0.54} \times B^{0.84} \times t_{2/3}^{\ 0.41}$$ (equation 3.2)

- f_{ren} is the confinement renormalization factor (example: 1.25 for the Helias 5-B, cf. [16]).
- a is the mean plasma minor radius in m (called "Rp" in this document).
- R is the major radius in m (called "r" in this document).
- P is the heating power in MW. As in [15], it will be considered the power needed to maintain the plasma stable, in excess of the fusion products heating the plasma. So $P = Ploss + Prad - P\alpha$
  Ploss is the average power lost by transport, Prad is the power lost by radiations and Pα is the power generated by fusion products. With the variables of [1], it can be written:
  $$P = P\_heating\_power\_MW = (Plost\,Reactor + Pr\,a\text{-}Pah) \times Torus\_volume\_m3 / 1E6$$
  (equation 3.3)
  With $Torus\_volume\_m3 = 2 \times \pi^2 \times r \times Rp^2$ (equation 3.4)
  It is implicitly supposed in the P calculation that the heating is the same in the entire reactor.
- ne is the average electron density in 1E19/m^3 (slightly above the D+ ions density)
- B is the toroidal magnetic field on the axis in T.
- $t_{2/3}$ is the rotational transform at 2/3 of the minor radius (example: 0.9 for the Helias 5-B, cf. [16]).

The problem with this formula (equation 3.2) is its big uncertainty on the confinement renormalization factor. For example according to [16], f_{ren} can be between 1.0 and 1.5.
To avoid this renormalization factor, a better solution is given by [17] in the form of 2 formulas which apply

to the Helias 5-B Stellarator ("Shearless" configuration), which constitutes the model of both half-toruses. An average of these 2 formulas will certainly give a better result than the ISS004 scaling. This average is, in practice, very close to the ISS04 result. These two formulas to average are (from [17]):

$$Tct\_s1 = \frac{0.0554 \times a^{2.17} \times R^{0.64} \times P^{-0.62} \times ne^{0.74} \times B^{1.25} \times t_{2/3}{}^{0.30}}{\left(1 + 1.44 \times \exp(-2 \times R / 1.8377)\right)}$$ (equation 3.5)

$$Tct\_s2 = \frac{0.0527 \times a^{2.15} \times R^{0.62} \times P^{-0.62} \times ne^{0.71} \times B^{1.20} \times t_{2/3}{}^{0.20}}{\left(1 + 1.24 \times \exp(-2 \times R / 1.8377)\right)}$$ (equation 3.6)

$$Tct = \frac{Tct\_s1 + Tct\_s2}{2}$$ (equation 3.7)

These formulas apply to both half-toruses because together they form a torus.

If applied to the straight pipes of the fusion reactor (see figure 2), these formulas would give an infinite confinement time due to an infinite R, which is not physical. In fact, this particular confinement time (TDexpsp) and the time confinement for the whole reactor (TDexp) are estimated in the Appendix A of [1].

3.4 About charge exchanges between ions and gas neutrals

In [1] (§2.2.4), it was supposed that permanent losses were caused by charge exchanges on neutrals, coming from the residual gas inside the reactor and from the gas stored on the wall surface. This behavior exists at the beginning of the heating operation but disappears progressively, so charge exchanges will not be considered. Consequently, the variables $\gamma cep = \gamma ceT = \gamma ceHe3 = \gamma ceHe4 = \gamma ceD$ will be forced to 0.

Note that there is a normal neutralization of charged particles of all kinds, lost by radial diffusion, on a Divertor plate or on the wall, followed by the molecular vacuum pumping of a part of these neutral particles at the Divertor level, and, for the other part, by ionization of these neutral particles. This complex problem is not addressed in this paper.

3.5 Important results of the fusion reactor calculation

It has been supposed an Rp radius equal to 2.0 m (default value). The main results of the fusion reactor calculation are the following:

- The Deuterium density (nD) = 8E19 D+ ions/m^3
- The injection energy of particles (D+ ions and electrons) = 88 keV
- The equilibrium energy of the plasma = 29 keV
- The mechanical gain Q = 0.184

Experimentally, it appears that a good performance for a fission reactor supplied by a D-D fusion reactor occurs when the product "Q x PnDDm3", used as a "quality" criterion, is maximum. Q is the mechanical gain and PnDDm3 the fusion power of 2.45 MeV neutrons. Now, this criterion is probably not the best one.

The rate of fusion neutrons generated from the two straight lines per sec (Qns_2L) is equal to:
$Qns\_2L = V\_straight\_lines\_m3 \times Qfnm3$ (equation 3.8)

- With $Qfnm3 = QfDDm3 / 2 + QfDTm3$ (equation 3.9)
  Qfnm3 is the rate of fusion neutrons (at 2.45 and 14.06 MeV) generated per s and per m^3.
  For QfDDm3 and QfDTm3 refer to [1].
- With $V\_straight\_lines\_m3 = 60 \times \pi^2 \times Rp^3$ (equation 3.10), the global volume in m^3 of both straight lines.

The consumed mechanical power, in W, for the whole fusion reactor (Pm_input_W) is equal to:
$Pm\_input\_W = V\_fusion\_reactor\_m3 \times Pinput$ (equation 3.11)

With $V\_fusion\_reactor\_m3 = 66 \times \pi^2 \times Rp^3$ (equation 3.12)

For Pinput refer to [1] .

The fusion power, in W, for the whole fusion reactor (Pth_fusion_W) is equal to:
$Pth\_fusion\_W = V\_fusion\_reactor\_m3 \times Poutput$ (equation 3.13)
With $Poutput = Pneu + PlostFi + PlostReactor + Pra = Pfusion + Pinput$ at equilibrium (equation B11 and B12 of [1]). For these variables, refer to [1].

The neutron power, generated by 2.45 and 14.06 MeV neutrons, issued from the volume of the straight lines of the fusion reactor, is equal to
$P\_neutron\_2L\_W = (PfnDDm3 + PfnDTm3) \times V\_straight\_lines\_m3$ (equation 3.13 bis)
With PfnDDm3+PfnDTm3=Pneu (global neutronic power per m^3, refer to [1] for details).

The proportion of neutrons at 2.45 MeV among all the neutrons generated at 2.45 and 14.06 MeV is equal to: $Pro\_n\_2\_45\_MeV = \dfrac{QfDDm3}{2 \times Qfnm3}$ (equation 3.14)

The surface heat power (SHP in W/m^2 §2.2.10 of [1]) is now calculated for both half-toruses where the heat power is maximum and for the straight pipes where the heat power is minimum:
$SHP\max = \dfrac{(PlostFi + PlostHT + Pra) \times Rp}{2}$ (Equation 3.15)
$SHP\min = \dfrac{(PlostFi + PlostD + Pra) \times Rp}{2}$ (Equation 3.16)

4. Physical model of the PWR fission reactor and the water steam circuit

4.1 Introduction

At this level, the fusion reactor has been calculated. The main parameters are known i.e.:
Rp_cm (Rp_cm = Rp x 100), Qns_2L, Pm_input_W, Pth_fusion_W and Pro_n_2_45_MeV.
Now the user proposes a mixture of U235, U238 and Th232 in terms of atomic concentrations, the total being equal to 1 (100%).
It will be first calculated the fission reactor dimensions, the mass of fuel etc.
Once the fission reactor is determined, the calculation will be incremental by step of one hour. It will be first calculated the reactivity, the thermal fission power generated, the net electric power and the evolution of the fuel. Results will be displayed per year. The main result is the average net electrical power.
The number of years calculated is determined by the user, but by default 100 years are proposed, 100 years being the minimum expected service life for this reactor (see §5.5).

4.2 First wall

The wall between the fusion reactor and the fission reactor must be a solid non-ferromagnetic metal able to support heat, such as Beryllium, Copper alloy, Austenitic stainless steel, etc. Now the sole metal able to let cross fast neutrons with few neutron absorptions is the Beryllium.

The estimation of the probability Pnc of non-collision of a 2.45 MeV neutron across a certain thickness of Beryllium is based on the method shown in [18] page 290, normally used for an infinite plate, so conservative here.

$$Pnc = \frac{\int_0^{\pi/2} \sin(\theta) \times \cos(\theta) \times \exp(-a \times \Sigma t / \cos(\theta)) \times d\theta}{\int_0^{\pi/2} \sin(\theta) \times \cos(\theta) \times d\theta}$$

With Σt the collision macroscopic cross section, a the beryllium thickness and θ the colatitude.
Note: the cross sections data of materials are taken from the library ENDF/B-VIII.0.

From Pnc and knowing the diffusion and the absorption cross sections at different energies, it can be calculated (with a small program) the average probability that a neutron crosses the wall and the average energy just after the crossing. For a Beryllium thickness of 5 cm, the probability for a 2.45 MeV neutron to cross the wall is equal to 0.915, with a final average energy of 1.12 MeV. This thickness of 5 cm is considered as the best compromise between resistance and neutrons transparency, and will be selected for the following.

For about the 14.06 MeV neutrons, it must be taken into account the n-2n reaction:
n + $^{9}Be \rightarrow 2\ n + 2\ ^{4}He$ -1.57 MeV
After a calculation similar to the previous one, it can be determined that for a Beryllium thickness of 5 cm, 2.9% of these neutrons are lost before reaching the energy of 2.45 MeV and, for the rest, the amplification factor is equal to 1.478. So the global probability PTr_Br for a neutron (initial or created via the n-2n reaction) to cross the 5 cm Beryllium wall is equal to:
$Ptr\_Be = 0.915 \times (\mathrm{Pr}o\_n\_2\_45\_MeV + (\mathrm{Pr}o\_n\_14\_06\_MeV \times 0.971 \times 1.478))$ (Equation 4.1)
With $Pro\_n\_14\_06\_MeV = 1 - Pro\_n\_2\_45\_MeV$ (Equation 4.2)

4.3 Temperature and pressure conditions of the main fission primary system

According to [19], the minimum elastic limit of Beryllium is equal to 170 MPa (1700 bar), for a maximum of 575 MPa. The average elastic limit is equal to 372.5 Mpa and it will be chosen an elastic limit of 280 Mpa.

> Note: it is supposed that the first wall in Beryllium and the exterior wall in 316LN are united by a fine structure, to strengthen the first wall against buckling and hence to avoid considering a too low elastic limit.

So the maximum pressure on the main primary system in bar abs (Pfr_bar_abs) to avoid the bucking can be estimated to:

$$Pfr\_bar\_abs = \frac{a\_Be\_cm \times Elastic\ \lim it\ (bar)}{Rp\_cm} = \frac{5 \times 2800}{Rp\_cm} = \frac{14000}{Rp\_cm} \text{ (Equation 4.3)}$$

The saturation temperature (Tsat_C), for a pressure between 20 and 155 bar abs, can be estimated to:
$T\_sat\_C = 100 \times \mathrm{Pfr\_bar\_abs}^{X}$ (Equation 4.4) with

$$X = 0.249755 \times (1 - 0.017357 \times ((\mathrm{Pfr\_bar\_abs} - 30) \times \mathrm{Sign(Pfr\_bar\_abs\text{-}30)}/124)^{0.5} \times \mathrm{Sign(Pfr\_bar\_abs\text{-}30)})$$

and Sign(Pfr_bar_abs-30)=1 if Pfr_bar_abs≥30, and Sign(Pfr_bar_abs-30)=-1 reversely.

The water inside the main primary system pressurizer being at T_sat_C and the difference between the pressurizer temperature and the primary temperature at 100% of the nominal power (Tfr_C) being supposed equal to 30°C, it can be deduced:
$Tfr\_C = T\_sat\_C - 30$ (Equation 4.5)

Note that for the default value Rp_cm=200 cm, the pressure is equal to 70 bar abs and the primary temperature is equal to 256°C, i.e. much less that the standard conditions of an EPR (155 bar abs and 313°C).

4.4 Gross electrical yield (Yg) and gross power delivered by the alternator (Pe_gross_W)

In what follows, it is considered a thermodynamic conversion of the output power in form of heat (Psg_W) supplied by the fission reactor, in electricity (see figure 4). The reference gross yield Yg = 0.39 of this conversion corresponds to the gross yield of a modern fission reactor, as the EPR one.
Let's suppose that for an EPR this Yg value is obtained for an average main primary temperature (Tfr_C) of 312.9 °C and a heat sink temperature of 34 °C. The theoretical Carnot yield would be equal to 0.4789. So, the EPR yield compared to the Carnot yield is equal to 0.82. Now, let's suppose that Yg is always proportional to the Carnot yield and to the EPR yield compared to the Carnot yield. It can be deduced the probable Yg for a given Tfr_C:

$$Yg = \frac{(Tfr\_C - 34)}{(Tfr\_C + 273.15)} \times 0.82 \text{ (Equation 4.6)}$$

So the gross power delivered by the alternator will be equal to $Pe\_gross\_W = Yg \times Psg\_W$ (equation 4.7)

With Psg_W calculated in §2.1.

4.5 Preliminary calculations relative to the fission reactor

The model of PWR used for calculations of this reactor about reactivity, thermal power, geometric values, etc is the P4/P'4 1300 MWe one developed in France by EDF + Framatome and based on a Westinghouse model (South Texas). This P4/P'4 model is chosen because a proposal for the complete calculation of this type of reactor is given in [20]. This calculation based of Uranium only has been extended, in this paper, to Plutonium, Thorium and minor actinides.

> Note 1: the necessary data as ν ("Nu") the average number of neutrons emitted by thermal fissions, the energy generated by fission, the thermal cross-sections, the specific gravities, the resonance integrals, the atomic masses, the number of atoms per cm^3, etc are extracted from [21].
>
> Note 2: it is supposed that the working of a PWR is known, as it will not be detailed. For example, see [22] for an abstract about PWR.
>
> Note 3: the calculations given in [20] and conventional calculations will not be (re)presented in the rest of this document, but will be present in the source code of the "D_D_PWR_hybrid_reactor" program, if relevant. However other calculations will be described below. Note that for neutronic calculations, nuclear research centers and engineering companies use powerful programs (see [21]).

About the maximum reactivity allowed

The reactor being sub-critical, there is no need of the full length rod control system. The negative reactivity weight of all these rods for a P4/P'4 reactor is, at maximum, a bit below 11000 pht ("pht" for "Per hundred thousand", i.e. in 1/100,000). So it will be considered that the reactor (normally critical) must work here, at best, at an effective multiplication factor (K_eff) corresponding to all rods inserted, so $K\_eff = \frac{1}{1-(-11000/1E5)} = 0.9$. For a real K_eff>0.9, the reactor water must be borated, so as to remain at K_eff=0.9. Reversely, for a real K_eff≤0.9 the water must be the least borated possible. This value of 0.9 will be commonly reached so the water will be, in general, borated.

This maximum of 0.9 will be used to calculate the fission reactor.

> Note: this hypothesis is important. For example for a maximum K_eff=0.95 instead 0.9, the average net electrical power is about 2.5 larger, as for example, for the default configuration: 3669 MWe instead 1437 MWe. However, with a maximum K_eff=0.95 instead 0.9, the core meltdown probability will be a bit larger even if still extremely low (see §5.9).

The number of fuel (U_Th_02) atoms per cm^3 of this fission reactor (N_fuel) is estimated to:

$$N\_fuel = \frac{Mean\_fuel\_specific\_gravity \times Avogadro\_number \times 0.281695}{A\_fuel} \times ((Tfr\_C + 273.15)/293.15)^{-0.023592}$$

(equation 4.8).

With A_fuel the atomic mass in g of the fuel, mixture of U235, U238, Th232 and O2 (i.e. U02+TH02).

The molar moderating ratio (MR) is estimated to:

$$MR = \frac{3.88553E22 \times ((Tfr\_C + 273.15)/293.15)^{-0.42173}}{N\_fuel}$$ (equation 4.9)

4.6 Maximum power produced by fission according to the fusion neutrons flux, for K_eff=0.9

The neutrons flux (Qns_2L) issued from the straight parts of the fusion reactor and having crossed the Beryllium wall is multiplied by a factor equal to 1/(1-K_eff), cf. [21] page 86.

The maximum effective multiplication factor K_eff=0.9 is used to size the fission reactor.

So $Qn\_fr = \frac{Qns\_2L \times Ptr\_Be}{(1-K\_eff)}$ (equation 4.10) with Ptr_Be defined in §4.2

The probability for a neutron to generate a fission is equal to $w = \frac{K\_eff}{Mean\_Nu}$

Mean_Nu, the average number of neutrons emitted by a thermal fission depends on the fuel.

So the number of fissions par sec is equal to $Qf\_fr = \frac{Qn\_fr \times K\_eff}{Mean\_Nu}$ (equation 4.11)

The maximum thermal power generated by fissions in the reactor (Pth_fission_MW) is equal to:
$Pth\_fission\_MW = Qf\_fr \times Mean\_energy\_release\_per\_fission\_MeV \times 1.60219E\text{-}19$ (equation 4.12) with Mean_energy_release_per_fission_MeV which depends on the fuel.

4.7 Dimension of the fusion reactor, for K_eff=0.9

The interior radius of the fission reactor is equal to:
$Ri\_fr\_cm = Rp\_cm + Beryllium\_thickness\_cm$ (equation 4.13)

The minimum volume of the core (fuel assemblies), on the basis of a P4/P'4 reactor (i.e. 99.64 MW/m^3 at 20°C, cf. [20]) at 20°C and at Tfr_C is equal to:

$Vco\_m3\_at\_20\_C = \frac{Pth\_fission\_MW}{99.64}$ (equation 4.14)

$Vco\_m3\_at\_Tfr = Vco\_m3\_at\_20\_C \times ((Tfr\_C + 273.15)/293.15)^{0.023592}$ (equation 4.15)

The exterior radius of the core is equal to:

$\mathrm{Re}\_co\_cm\_at\_Tfr = \sqrt{(Ri\_fr\_cm)^2 + \frac{(Vco\_m3\_at\_Tfr \times 1E6)}{(2 \times \pi \times L\_cm)}}$ (equation 4.16)

The thickness of the core is equal to:
$THco\_cm\_at\_Tfr = \mathrm{Re}\_co\_cm\_at\_Tfr - Ri\_fr\_cm$ (equation 4.17)

It is reminded that the core of the fission reactor is composed of a certain number of fuel assemblies. Each fuel assembly is composed of long fuel rods separated by water. For this reactor, it is implicitly supposed that the fuel assemblies are the ones used on P4/P'4 plants for about the reactivity calculation. However, the fuel assemblies will necessarily be modified as the rods will be longer and the fuel assembly will have a smaller section.

> Note that, for a P4/P'4 fuel assembly, there are 264 fuel rods for 289 passages. The remaining 25 passages are used for control rods, burnable poison, neutrons sources and instrumentation in the central passage. Only the central passage for instrumentation would be useful for this type of hybrid reactor. The remaining 24 passages could be used for fuel rods, which would increase the reactivity, without affecting the compactness.

Otherwise it has been estimated, for reasons relative to the minimum displacement of neutrons on the reactor and to the leakage of the reactor, that the core thickness (THco_cm_at_Tfr) must not be inferior to 12 cm.
So if the thickness result is inferior to 12 cm, it will be switched to 12 cm and the parameters Re_co_cm, Vco_m3_at_20C, Vco_m3_at_Tfr will be re-calculated using the reverse equations of the equations 4.14 to 4.16.

The reflector (light water) surrounds the fuel assemblies. Its thickness is supposed equal to 20 cm (estimated average value for a PWR). So the thickness reflector saving is equal to 8.27 cm according to [20]. Note that relatively to thermal neutrons, the Beryllium wall is considered as transparent.

The interior radius of the exterior wall in 316LN is equal to:
$\mathrm{R}i\_ew\_cm = \mathrm{Re}\_co\_cm + Reflector\_thickness\_cm$ (equation 4.18)

To bear the primary pressure (Pfr_bar_gauge), the thickness of the exterior wall in 316LN is estimated to:

$$TH\_ew\_cm = \frac{Pfr\_bar\_gauge \times Ri\_ew\_cm}{Minimum\ elastic\ \lim it\ for\ 316LN(bar) - (Pfr\_bar\_gauge/2)}$$ (equation 4.19)

With $Pfr\_bar\_gauge = Pfr\_bar\_abs - 1$ (equation 4.20)
The minimum elastic limit for 316LN is considered equal to 100 Mpa for a nominal value of 205 MPa ([23]).
TH_ew_cm is equal to 16.9 cm for the default configuration.

The exterior radius of the exterior wall in 316LN is equal to: $\mathrm{Re}\_ew\_cm = \mathrm{R}i\_ew\_cm + TH\_ew\_cm$ (equation 4.21)
Note that this exterior radius is the interior radius of the system supporting the magnetic coils cooled by the cryogenic fluid (see [11]).

4.8 Determination of the net and auxiliary power delivered by the alternator
Several consumers must be supplied with electricity by the alternator before supplying the first useful watt (see figure 4). They are quantified below.

The cryogenic system necessary to cool the superconducting coils is proportional to the area covered by the coils. From the Helias data, it has been determined that the specific electrical power for cryogenics (mainly used by magnetic coils) is about 13600 W/m^2.
The circumference in m of the fusion reactor (at the axis) is equal to $Cfr\_m = 66 \times \pi \times Rp\_cm/100$ (equation 4.22). So the necessary cryogenic power for superconducting coils is estimated to:
$Pe\_cryogenic\_W = Cfr\_m \times 2 \times \pi \times \left(Re\_ew\_cm/100\right) \times 13600$ (equation 4.23)

For UHV (Ultra High Vacuum) needs, it will be supposed that the electrical consumption is equal to 20% of the cryogenic power for superconducting coils, so $Pe\_UHV\_W = Pe\_cryogenic\_W/5$ (equation 4.24)

As it is supposed that the efficiency of the D+ ions/electrons injectors of the fusion reactor is equal to 0.8, the electrical power consumed is equal to $Pe\_input\_W = Pm\_input\_W/0.8$ (equation 4.25)

These 3 consumptions are fixed. The total is equal to:
$Pe\_fixed\_W = Pe\_cryogenic\_W + Pe\_UHV\_W + Pe\_input\_W$ (equation 4.26)

Moreover, the plant needs electrical power for auxiliary equipment (pumps, valves, instrumentation, control…). It will be considered that it is equal to a fraction of the thermal power delivered to the steam generators. Based on the P4/P'4 plants: $Pe\_aux\_W = 0.0131 \times Psg\_W$ (equation 4.27)
With Psg_W calculated in §2.1.
So the net power delivered by the alternator to the grid is equal to:
$Pe\_net\_W = Pe\_gross\_W - (Pe\_fixed\_W + Pe\_aux\_W)$ (equation 4.28)

Note: Pe_gross_W is always positive, while Pe_net_W can be:
- Positive, i.e. electrical power is delivered to the grid, which is the expected way.
- Negative which means that electrical power is taken from the grid. This happens at the beginning of the Thorium incineration, while there is not a sufficient amount of fissile materials.

From Pe_net_W, it is calculated the average electrical net power over the whole working of the reactor (Pe_average_MW).

4.9 Determination of the multiplication factors: infinite (K_infinite) and effective (K_eff)
For information about K_infinite, K_eff and the non-leakage probability, see [24] or [21] page 115 and 155.

The infinite multiplication factor K_infinite is calculated according to [20]. The calculation of [20] is done in hot standby (critical) for enriched U235/U238 fresh fuel. It will be carried to the "full power " reactor state and extended to a fuel containing Uranium, Plutonium, Thorium, minor actinides and fission products.

About the K_infinite factor, below is reminded the method. It is calculated supposing a reactor of infinite dimensions. $K\_\inf inite = \eta \times f \times p \times \varepsilon$ with:

- η ("eta") is the thermal fission factor, i.e. the number of neutrons produced per fission for one absorbed neutron of the fuel. It is not equal to ν because an absorption can be sterile.
- f is the thermal utilization factor, i.e. the probability for a neutron to be absorbed in the fuel rather than on the moderator or on the cladding.
- p is the resonance escape probability, i.e. the probability for a neutron not to be absorbed before reaching the thermal domain (about 1 eV), so escaping to the resonance hatches of the epithermal domain.
- ε ("epsilon") the fast fission factor is a corrective factor to take into account the few fast fissions, i.e. the number of native neutrons issued from fast and thermal fissions relatively to the number of native neutrons issued from the sole thermal fissions. ε is slightly superior to 1.

The reactor has of finite dimensions, so it is necessary to estimate the non-leakage probability of fast and thermal neutrons (Pnl). It is taken into account by K_eff, according to [25] page 80.

> Note: for the neutrons issued from the core and crossing the Beryllium wall (so without being reflected), this wall and the interior fusion reactor are considered transparent. The leakage in only considered towards the exterior.

$K\_eff = K\_\inf inite \times Pnl$ (Equation 4.29) and $Pnl = \dfrac{\exp(-Bg^2 \times Tau)}{Lth^2 \times Bg^2 + 1}$ (equation 4.30)

The slowing-down area Tau (also called "Fermi age") for a PWR is equal to 50 (from [21] page 409).
The diffusion area of thermal neutrons Lth^2 for a PWR is equal to 6 (from [21] page 409).
The geometric buckling Bg^2 is calculated, for a cylinder of radius R (cm) and height H (cm) by (from [21] page 155): $Bg^2 = \dfrac{j^2}{R^2} + \dfrac{\pi^2}{H^2}$ (equation 4.31) with j=2.40483

Here H=L_cm with L_cm the straight pipe length equal to $L\_cm = 100 \times L\_m$ and $L\_m = 30 \times \pi \times Rp$ (from [1]) (equation 4.31 bis)

R is the exterior core radius added to the thickness reflector saving (8.27 cm, cf. §4.7), so here:
$R = \mathrm{Re}\_co\_cm\_at\_Tfr + 8.27$

Now the core is not a cylinder but a ring of interior radius Ri_fr_cm (cf. §4.7). So it will be supposed that:
$R^2 = (\mathrm{Re}\_co\_cm\_at\_Tfr + 8.27)^2 - (Ri\_fr\_cm)^2$

The initial K_eff factor of [20] is calculated in hot standby, i.e. in a critical state, very close to 0% of the nominal power, the reactor temperature being close to the reactor nominal temperature at full power. The variation of the average moderator temperature between hot standby and full power, which depends on a temperature program, will be neglected relatively to the reactivity and consequently to K_eff.

However, increasing the nuclear power until the nominal power (100%) will introduce negative reactivity due to the Doppler effect on the fuel. Based on the Candu reactor data in [26], it can be estimated that:
$Doppler\_pht = Min(400, 850 - 450 \times (en\_Pu239 + en\_Pu241)/0.0026)$ (equation 4.32)
With en_Pu239 and en_Pu_241, the atomic concentrations of Pu239 and Pu241

Again from the Candu reactor data [26], the negative reactivity introduced by Xenon and Samarium to reach the full power, from 0% of power, is estimated to 3350 pht.

Still from the Candu reactor data [26], the negative reactivity introduced by fission products between two refueling operations, is estimated to: $nr\_fp\_pht = 0.115 \times Burnup\_MWd\_t$ (equation 4.33)
For the fuel burn-up, see §4.11.

The total negative reactivity (in absolute) Negative_reactivity_pht is equal to the sum of these previous terms (Doppler effect, Xe + Sm and fission products).

The initial reactivity at hot standby, in pht, is equal to $\rho\_hs\_pth = \frac{1E5 \times (K\_eff - 1)}{K\_eff}$ (cf. [21] page 119) (equation 4.34). It is negative because the reactor is in a sub-critical state.
With the calculated absolute negative reactivity, the reactivity at full power moves to:
$\rho\_fp\_pht = \rho\_hs\_pht - Negative\_reactivity\_pht$ (equation 4.35)

So at full power $K\_eff = \frac{1}{1 - \rho\_fp\_pht / 1E5}$ (equation 4.36)

4.10 Determination of the thermal neutrons flux

To simplify, the thermal neutrons flux (in neutrons/(cm^2 x s)) is the sole considered. The epithermal and the fast neutrons flux are not considered.

Neutrons flux due to the sole fusion neutrons

In absence of reactivity, i.e. for a 100% natural Thorium fuel, the sole neutrons flux is the one generated by the fusion reactor. So a minimum average neutrons flux across the core of the fission reactor must be calculated.

The total neutrons flowrate generated by the fusion reactor is known: Qns_2L (calculated in §3.5).
The source area (considered at the exterior of the Beryllium wall) is equal to
$Source\_Area\_cm2 = 2 \times \pi \times Ri\_fr\_cm \times (2 \times L\_cm)$ (equation 4.37)
With Ri_fr_cm calculated in §4.7 and L_cm in §4.9.

The flowrate of neutrons through this area is equal to $S\_n\_per\_cm2\_s = \frac{Qns\_2L \times Ptr\_Be}{Source\_Area\_cm2}$

Ptr_Be, the probability for a neutron to cross the Beryllium wall, is defined in §4.2.
To simplify, let's consider that the neutrons source is a plane surface.
Moreover, it will be supposed that neutrons are thermal ones, which is simpler to manage but conservative as initial fast neutrons displacement is much longer than the one of thermal neutrons. So the neutrons issued from the Beryllium wall are considered to immediately be slowed down to thermal neutrons.

The neutrons flux can be calculated using the formula from [21] page 141:

$$\mathrm{Flux}(x) = S\_n\_per\_cm2\_s \times \frac{\exp(-K \times x)}{2 \times K \times D}$$

With x the distance between the point considered inside the core and the exterior Beryllium surface (in cm).

According to [21] pages 142 and 144, $\mathrm{K} = \frac{1}{Lth(cm)} = \frac{1}{\sqrt{6}} = 0.408 cm^{-1}$ with Lth^2 defined in §4.9.

D the diffusion coefficient is equal to 0.2 cm ([21] p 144) for thermal neutrons.
The average flux across the core thickness (THco_cm_at_Tfr, as defined in §4.7) is calculated this way:

$$Thermal\_fusion\_neutron\_flux = \frac{\int_0^{THfr\_cm\_at\_Tfr} Flux(x) \times dx}{THco\_cm\_at\_Tfr}$$ and after some developments:

$$Thermal\_fusion\_neutron\_flux = \frac{15 \times S\_n\_per\_cm2\_s}{THco\_cm\_at\_Tfr} \times (1 - \exp(-0.408 \times THco\_cm\_at\_Tfr))$$

(equation 4.38)

Normal average neutrons flux due to fusion + fission neutrons

With reactivity (standard behavior), the neutrons flux is determined this way.
Let's calculate the fission rate in neutrons per cm^3 and s:

$Fission\ rate = \frac{Qf\_fr}{Vco\_m3\_at\_Tfr \times 1E6}$ (equation 4.39)

With Qf_fr calculated in §4.6 and Vco_m3_at_Tfr in §4.7.
So the average neutron flux is equal to (cf. [21] page 104):

$$Thermal\_fission\_neutron\_flux = \frac{Fission\_rate}{\varepsilon \times Sigma\_f\_th}$$ (equation 4.40)

The fission rate is divided by ε (calculated in §4.9) to subtract fast fissions. Sigma_f_th is the thermal fission macroscopic cross-section ($= \sum_i Ni \times \sigma fi$, cf. [21] page 69), detailed in the “D_D_PWR_hybrid_reactor“ program.

Final neutrons flux
It will be considered that for K_eff≤0.01, i.e. a fission reactor state with almost no reactivity, the final neutrons flux will be calculated on the basis of the Thermal_fusion_neutron_flux. Reversely for K_eff>0.01 it will be calculated on the basis of the Thermal_fission_neutron_flux.
This only applies to the Thorium fuel. For the Uranium (natural or depleted), K_eff is always widely superior to 0.01 so the sole Thermal_fission_neutron_flux is used.

Maximum neutrons flux
Note that about the thorium fuel incineration, in [8] page 13, it is calculated a maximum neutrons flux equal to $2.0E14 \times \sqrt{Treactor(K)/300}$, this to avoid an important bypass of the U233 fission. This limit is high and will not be considered in the present reactor design.

4.11 Refueling and fuel burn-up

The fuel reprocessing and refueling is supposed to be done as detailed in [27]. So the fission products and the minor actinides (americium, curium, neptunium), are removed from the spent fuel and finally stored.

> Note: the Protactinium (Pa) isotopes rapidly disappear, due to their weak radioactive half-life (27 days for the Pa233 and 6.7 hours for the Pa234). So the quantity of Protactinium isotopes is considered as nil after a refueling.

Not removing the minor actinides Am and Cm would reduce the net electrical power. For example, after 255 years of operation, the loss would be equal to 100% for the natural Uranium and less than 0.1% for the natural Thorium. So such hypothesis is not considered, even if the magnitude of this problem for Thorium is weaker.

> Note: as Am241 and Cm244 are considered as final actinides (see §4.12), they accumulate. In reality, transmutations continue. The result for natural Uranium (100% of loss) is probably exaggerated but not so much. Even for Thorium, the very slow accumulation of Am241, Cm244 and successors will finish to reduce the reactivity and the net electrical power. However, thanks to this weak accumulation, the refueling of the whole Thorium fuel could be made when the fuel burn-up reaches 100 GWd/t instead 10 GWd/t.

Reversely, not removing the Neptunium does not change almost anything on the results, except 0.1 % on the final net electrical power with Thorium. So the Np237 can remain in the fuel, but the other isotopes of the Neptunium rapidly disappear by radioactive decay. However, to simplify, all Neptunium isotopes will be set to 0 after each refueling.

The Uranium and the Plutonium are extracted (PUREX process, described in [27]). For the Thorium, a similar process to extract Thorium called “THOREX” is expected. See from §5.1 to §5.4 for recommendations about these processes.

To simplify the calculation, it is supposed a regular refueling of the whole fuel (and not by 1/3 or 1/4) each time the fuel burn-up (calculated from the previous refueling) passes 10000 MWd/t, so once each 7 to 9 years. Note that this value of 10000 MWd/t is inferior to the nowadays burn-up limit for PWR fuel assemblies, see [28] pages 57 and 58. Moreover, it is supposed that the fuel reprocessing is done immediately, and not after years.
So, after each immediate fuel reprocessing/refueling:

- The fission products have been removed. Note that, contrary to the minor actinides, the fission products are not directly considered as they are taken into account via their negative reactivity (cf.

§4.9).
- There is no more protactinium, americium, curium and neptunium in the fuel.
- The loss of fuel is completed with fresh fuel at the initial atomic composition (U235/U238/Th232).
- The burn-up is switched to 0.

Note: for Thorium, the refueling could possibly be done after a burn-up larger than 10000 MWd/t, because the accumulation of minor actinides is slow and weak.

The fuel burn-up (Burn_up_MWd_t) is calculated in the following way.
At each step (one hour of working), the thermal energy produced during this step is calculated, i.e. $\Delta ThermalEnergy(MJ) = Pth\_fission\_MW \times 3600$, with Pth_fission_MW calculated in §4.6
This energy is added to Energy_generated_MJ_from_the_last_refueling.
The fuel burn-up in MWd/t is calculated this way:
$$Burn\_up\_MWd\_t = \frac{Energy\_generated\_MJ\_from\_the\_last\_refueling}{86400 \times Fuel\_mass\_t} \quad \text{(equation 4.41)}$$
The fuel mass is calculated from the fuel volume:
$$Fuel\_volume\_m3\_20\_C = Vco\_m3\_at\_20\_C \times 0.281695 \quad \text{(equation 4.42)}$$
And $Fuel\_mass\_t = Fuel\_volume\_at\_20\_C \times Mean\_fuel\_specific\_gravity$ (equation 4.43)
With Vco_at_20_C calculated in §4.7 and the Mean_fuel_specific_gravity detailed in the "D_D_PWR_hybrid_reactor" program.

It is also calculated and displayed at each reloading the mass of fuel consumed from the beginning of operation, which reported to the total thermal energy produced, gives the average burn-up of the consumed fuel in MWd/t. A calculation of this average burn-up gives after 255 years of operation, the reactor being considered stabilized:
- 722,9 MWd/t for the natural Uranium.
- 819,0 MWd/t for the natural Thorium.

4.12 Fuel evolution

In the figure 6 below, it will be found the simplified chain of the managed fuel.

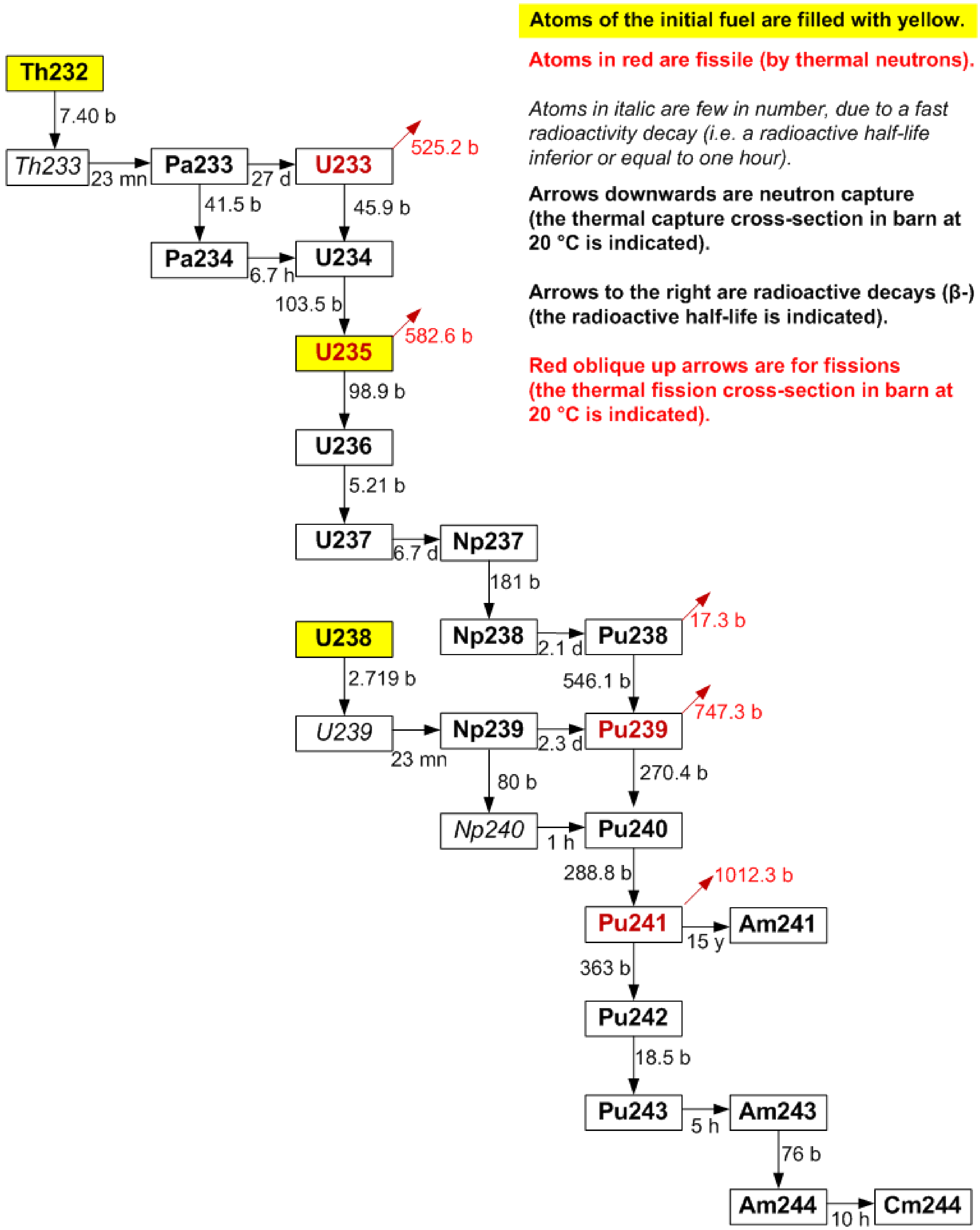


Figure 6. Simplified evolution chain of a Thorium (Th232) + Uranium (U238+U235) fuel

The calculation of the fuel evolution is repetitive and so it is not described here. It is detailed in the source code of the "D_D_PWR_hybrid_reactor" program.
Note that Am241 and Cm244 are the final minor actinides of this simplified evolution chain. In fact, there are other transmutations of both materials in other minor actinides, but without importance relatively to the target of this document.

To abstract, the equations are based on two main reactions (see also [21] page 260 for more details):

- Either a neutron capture or a fission due to the neutron thermal flux:
  $$\frac{dN}{dt} = -\sigma \times N \times Thermal\_neutron\_flux$$
  With N: the number of atoms at the time t and σ the capture or the fission cross-section.
- Or a radioactive decay (β-): $\frac{dN}{dt} = -\lambda \times N$ with λ the decay constant. If the decay constant is inferior or equal to 1 hour, the decay is considered as a by-pass.

Note 1: for fissile materials, a neutron absorption can generate either a capture or a fission. The n→2n reaction also exists at high energies, but it is not considered here.

Note 2: thermal cross-sections σ are supposed to follow a correction in 1/Speed (i.e. σ decreases when the temperature increases), even if it is only approximate compared to the reality, so:

$$Correction\_of\_\sigma = \sqrt{\frac{293.15}{Tfr\_C + 273.15}}$$

Note 3: resonance integrals Ia (absorption), If (fission) and Ia-If (capture) are supposed to increase linearly with the temperature (as p the resonance escape probability decreases with the temperature), so $Correction\_of\_Ia\_If = \frac{Tfr\_C + 273.15}{293.15}$

The fuel absorption data in the epithermal domain through the resonance integrals are given by [21] pages 466 and 467. However, the self-shielding factor which reduces each resonance integral is not known, because it depends on the neutrons spectrum. A self-shielding factor of 14 by default is taken but it is very imprecise.
For Uranium, these factors have been calibrated to be close to the fuel evolution results given in [29] page 25, for a PWR. So the calculated Uranium fuel evolution must be considered as not very far away from reality.

Note about comparison between PWR and Candu reactors, with natural Uranium as fuel
Fuel evolution of Candu, as the one given by [26] page 135, cannot be used for a PWR because its neutron spectrum is mainly only thermal due to the very good moderation, whereas the PWR spectrum is also epithermal for a small part. With the same neutron flux, the production of Plutonium by the PWR is stronger due to this difference of neutrons spectrum.
The initial K_infinite of a Candu reactor is higher than the initial K_infinite of a PWR (about 1.08 versus 0.88), because it is better moderated with heavy water in a large reactor.
But due to this PWR better production of Plutonium, the K_infinite evolution is different, as the initial increase of K infinite is stronger on a PWR compared to a Candu: 1.08 to 1.09 for a Candu ([26] page 128) but 0.88 to 1.09 for a PWR according to the author's calculations.
Note that the high thermal neutrons flux of Candu reactors (about 1E14 n/(cm^2.s)) compared to a PWR (about 3E13 n/(cm^2.s)) accelerates the process, but does not change much the fuel evolution.

Of course, there is no example of PWR working with Thorium. However, it has been verified that the Thorium fuel evolution results are coherent with the figure 4 of [8], which also shows a calculated Thorium fuel evolution. The fuel evolution results given for Thorium must be considered as inaccurate.

4.13 Description of the way to calculate the hybrid reactor

Here is described the global way to calculate the hybrid reactor, as written in the source code of the "D_D_PWR_hybrid_reactor" program.

For a given fusion pipe radius and a given maximum Beta, the best fusion reactor is calculated according to [1] and to the modifications described from §3.1 to §3.4. Note that the ranges of density of D+ ions and energy injection of particles (D+ ions + electrons) are defined by the user.
The main results about this fusion reactor, useful for the fission reactor, are described in §3.5 (Qns_2L, Pm_input_W, Pth_fusion_W, P_neutron_2L_W and Pro_n_2_45_MeV).

Preliminary calculations are done. They are described in §4.2 (Ptr_Be), §4.3 (Pfr_bar_abs and Tfr_C), §4.4 (Yg), §4.5 (K_eff maximum=0.9, N_fuel and MR). The geometry of the fission reactor is calculated in §4.7 (mainly THco_cm_at_Tfr) using the maximum fission power (Pth_fission_MW) determined in §4.6. The initial K_infinite factor and the non-leakage probability (Pnl) are calculated in §4.9. The minimal thermal neutron flux is calculated in §4.10.

Once the preliminary calculations done, it is determined the evolution of the fuel and all the parameters over 100 years by default (255 years maximum), by step of one hour. For one step it is calculated:

- K_infinite and K_eff (§4.9). Displayed once a year.
- The mean ν and the main energy by fission.
- The thermal fission power generated (Pth_fission_MW calculated in §4.6). Displayed once a year.
- The global power supplied to the steam generators (Psg_W calculated in §2.1)
- The standard thermal neutrons flux (§4.10). Displayed once a year.
- The fuel burn-up since the last fuel reprocessing/reloading in MWd/t (§4.11). Displayed once a year.
- The gross and net electrical powers (§4.4 and §4.8). Net power displayed once a year.
- The fuel evolution (§4.12).
- If the burn-up has passed 10000 MWd/t, a fuel reprocessing/reloading is immediately realized, with replacement of the fuel lost by fresh fuel, initialization to 0% of minor actinides and switching of the fuel burn-up to 0 MWd/t. It is displayed the mass of fuel consumed in t and the average burn-up of this consumed fuel in MWd/t (§4.11).

Each year, it is also given the following pieces of information:

- The fusion to fission power amplification gain $\dfrac{Psg\_W}{P\_neutron\_2L\_W}$ (=134 for the default configuration with natural Uranium).
- The global power amplification factor equal to $\dfrac{Psg\_W}{Pm\_input\_W}$ (=12.2 for the default configuration with natural Uranium).
- The average electrical net power in MW over the global working of the reactor (§4.8).

At the end of the calculation (so after 100 years of working time by default), the final fuel composition (atomic concentrations) is displayed.
Of course, as the source code is available, the user can modify the information displayed.

5 Results and discussion

In this chapter, it will be compared the main results obtained with natural Uranium, depleted Uranium, natural Thorium and a 50/50 mixture of natural Uranium and natural Thorium. These main results are:

- The K_eff multiplication factor before limitation which is more interesting that the limited K_eff. It is reminded that K_eff is limited to 0.9 with borated water (§4.5).
- The net electrical power in MWe.

Afterwards, points to deepen, ways to improve this hybrid reactor and safety aspects will be discussed.

5.1 Working with natural Uranium over 100 years

In figure 7 below, it is displayed the results obtained with natural Uranium as fuel, from 0 to 100 years. The discontinuities on K_eff are due to the refueling operations. The initial rise of both parameters is due to

the transmutation of U238 in Pu239, before stabilization.
After stabilization, the net electrical power reaches a value around 1432 MWe. The net electrical power is constant because the K_eff is limited to 0.9.
Note that even without borated water, the reactor would remain sub-critical (K_eff<1) except in year 10 for which it reaches 1.002.

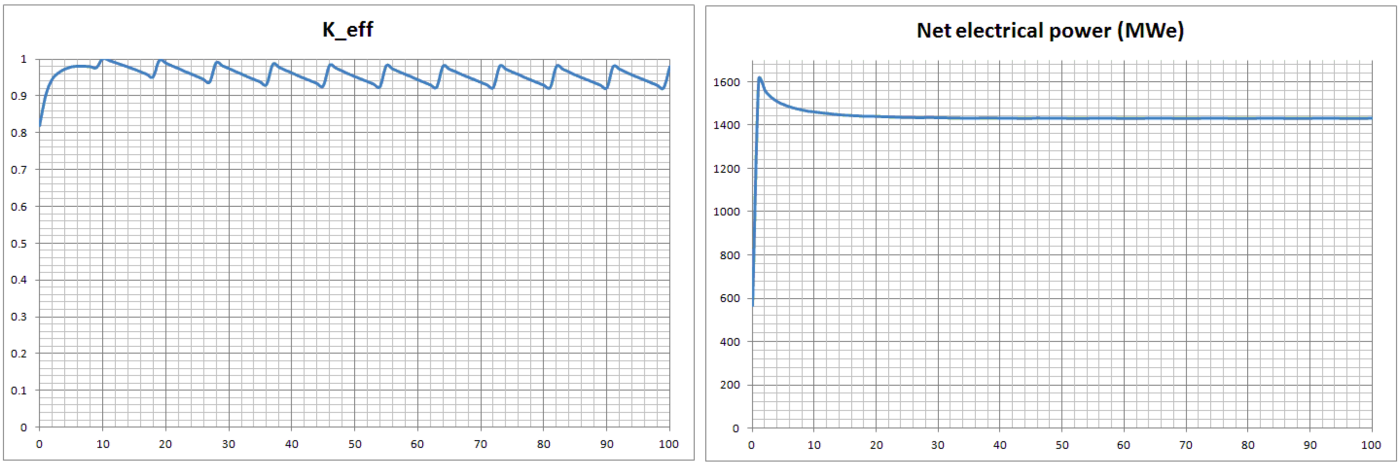


Figure 7. Main results for natural Uranium with respect to time in years

About the fissile Plutonium proliferation

The final composition shows:
Pu238=0.002%, Pu239=0.674%, Pu240=0.199%, Pu241=0.265%, Pu242=0.581% and Pu243=0.000%
The fissile materials (Pu239 and Pu241) are present at 0.94% whereas non-fissile materials are present at 0.78%. The percentage of fissile materials is too much important compared to the non-fissile materials, which might be avoided, for obvious reasons.
For this type of reactor and as no Uranium enrichment is required, the PUREX process might be modified so that no separation between Uranium and Plutonium be made. In that case, the Plutonium would be mixed with 98.058% of U238 (non-fissile) and no direct risk with this fuel would exist.

5.2 Working with depleted Uranium over 100 years

On the figure 8 below, it is displayed the results obtained with depleted Uranium as fuel (0.3% of U235), from 0 to 100 years.
The initial rise of both parameters is stronger than for natural Uranium due a smaller initial U235 concentration (0.30% instead 0.72%). The initial reactivity is equal to 0.505 versus 0.820 for the natural Uranium. The initial net electrical power is negative for about two years, but, afterwards, it reaches a stabilized value of around 1431 MWe.

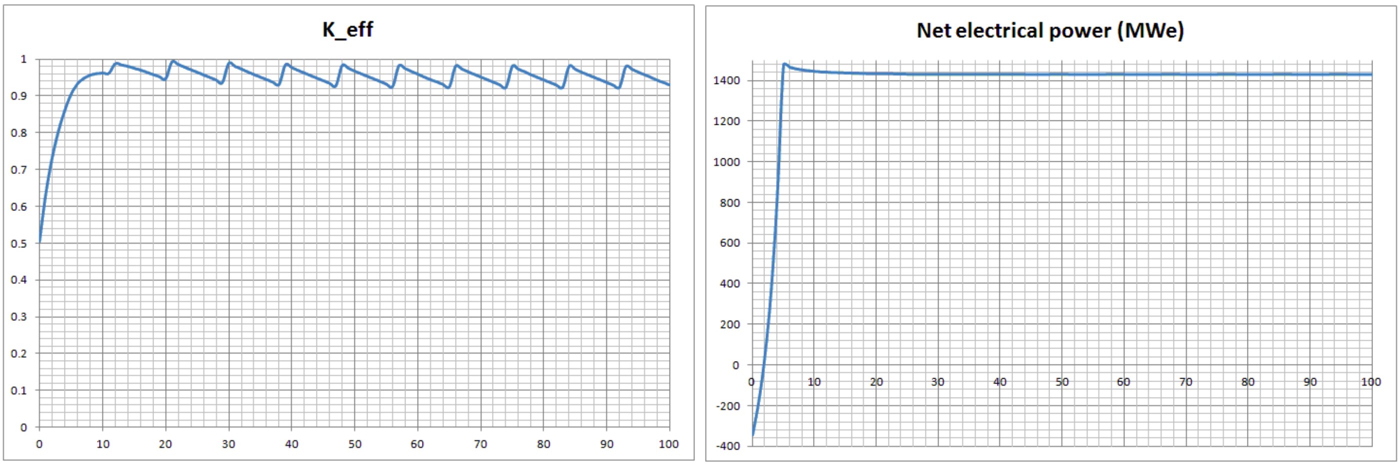


Figure 8. Main results for depleted Uranium with respect to time in years

About the fissile Plutonium proliferation: the results are about the same as with natural Uranium (§5.1), so the recommendation not to separate Uranium from Plutonium is the same.

5.3 Working with natural Thorium over 100 years

On the figure 9 below, it is displayed the results obtained with natural Thorium as fuel, from 0 to 100 years. The initial rise of both parameters is deeper and longer than for depleted Uranium due an initial zero reactivity and transmutation of the fertile material Th232 in the fissile material U233. The initial net electrical power is negative for the first 18 years, but, afterwards, it reaches a stabilized more elevated value than Uranium (1631 MWe instead 1432 MWe). The net electrical power is constant because the K_eff is limited to 0.9. Note that without borated water, the reactor would remain sub-critical (K_eff<1) for the first 47 years before being stable around 1.008.

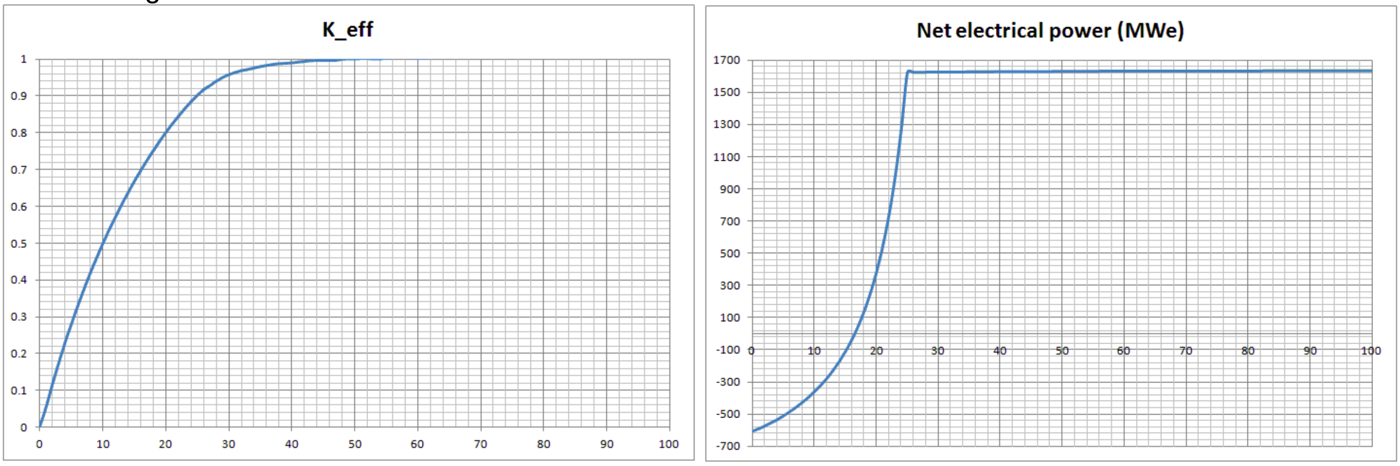


Figure 9. Main results for natural Thorium with respect to time in years

About the fissile Plutonium and Uranium proliferation

The final composition shows for Plutonium:

Pu238=0.007%, Pu239=0.003%, Pu240=0.001%, Pu241=0.001%, Pu242=0.002% and Pu243=0.000%

The problem is the same as with natural Uranium but with a much lesser magnitude. However, the recommendation not to separate Uranium from Plutonium remains.

The final composition shows for Uranium:

U233 =1.460%, U234= 0.857%, U235=0.210%, U236=0.158%, U237=0.000%, U238=0.000% and U239=0.000%

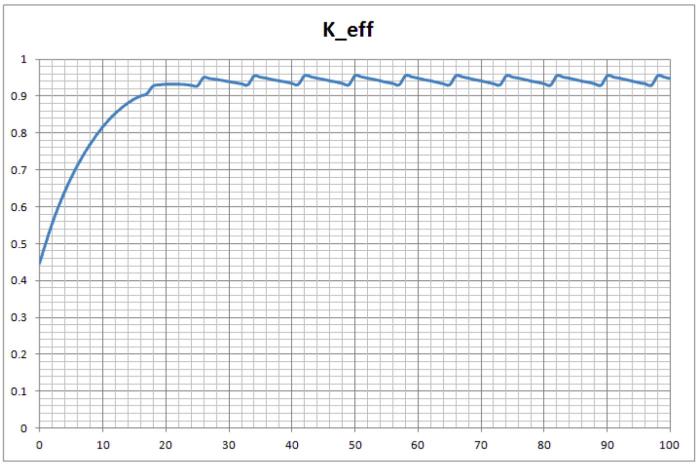
The fissile materials (U233 and U235) are present at 1.670% whereas non-fissile materials are present at 1.015%. The percentage of fissile materials is too much important compared to the non-fissile materials, which might be avoided, for obvious reasons.

For this type of reactor, the THOREX process might be modified so that no separation between Uranium, Plutonium and Thorium be made. In that case, the fissile Uranium and Plutonium would be mixed with 96.560% of Th232 (non-fissile) and no direct risk with this fuel would exist.

5.4 Working with a 50/50 mixture of natural Uranium and natural Thorium over 100 years

On the figure 10 below, it is displayed the results obtained with a mixture of 50 % of natural Uranium (0.36% of U235 and 49.64% of U238) and 50% of natural Thorium, from 0 to 100 years.

The initial rise of both parameters is due to the transmutation of U238 in Pu239 and Th232 in U233, before stabilization. The initial net electrical power is negative for the first 6 years, but, afterwards, it reaches a stabilized more elevated value than Uranium (1532 MWe instead 1432 MWe). Note that without borated water, the reactor would remain sub-critical (K_eff<0.96).



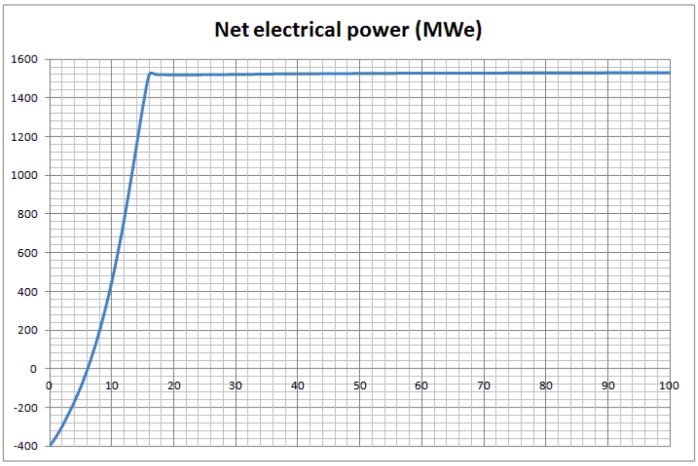


Figure 10. Main results for a 50/50 mixture of natural Uranium and natural Thorium with respect to time in years

About the fissile Plutonium proliferation
The final composition shows for Plutonium and Uranium:
Pu238=0.005%, Pu239=0.335%, Pu240=0.099%, Pu241=0.133%, Pu242=0.292% and Pu243=0.000%
U233 =0.738%, U234= 0.431%, U235=0.110%, U236=0.092%, U237=0.000%, U238=48.223% and U239=0.000%

The problem is the same as with natural Uranium but with a half magnitude. The recommendation not to separate Uranium, Thorium and Plutonium remains, even if in that case the Thorium could be possibly separated thanks to the U238 presence.

5.5 Estimation of the reactor service life limitations due to the neutrons flux
Due to the neutrons flux, the mechanical characteristics of the Beryllium first wall and the 316LN exterior wall will be progressively altered. The service life of this reactor mainly depends on the service lives of both walls.

> Note: the reactor service life also depends, equipment by equipment, on the number of operating transients. This reactor is supposed to work in a steady-state operation to minimize the number of transients.

As shown below, both service lives are much larger than the minimum expected service life for this reactor supposed equal to 100 years. So the reactor size could be reduced, if possible, for about this aspect of things.

About the Beryllium first wall
The main problem is the generation of 14.06 MeV neutrons due to fusions: see the interaction between neutrons and Beryllium in §4.2. From the default configuration results, about 26% of the neutrons are 14.06 MeV ones, for an average neutrons flux ("SNP") of 10600 W/m^2. Let's suppose that:

- Each 14.06 MeV neutron interacts one time with the first wall.
- The maximum sustainable fluence is equal to 2 MW.year/m^2 corresponding to about 20 dpa, 20 dpa being the limit for modest additional Helium effects ([30]) .

In these conditions, the service life of the first wall will be limited to: $\frac{2E6}{10600 \times 0.26} = 726\,years$

About the 316LN exterior wall
The P4/P'4 PWR was initially designed for a service life, mainly depending on the reactor vessel body, of 40 years. The thermal power is equal to 3800 MW and the surface of the vessel in front of the core is equal to about 59 m^2, whereas they are respectively equal of 6160 MW and 6015 m2 for this reactor in its default configuration.
The service life for the exterior wall which is, as first hypothesis, inversely proportional to the thermal power

and proportional to the surface receiving the neutrons flux, will be limited to: $40 \times \frac{3800}{6160} \times \frac{6015}{59} = 2516\,years$

5.6 Points to deepen

This is a non-exhaustive list of points to deepen. The points to deepen given in [1] could also be consulted even if they are less critical due to the more realistic fusion configuration used in this document.

- The way to calculate these reactors is very simplified. More accurate calculations would be necessary. Moreover, as indicated in the figure 2, for the neutrons issued from the core and crossing the Beryllium wall, this wall and the interior fusion reactor are considered as transparent (§4.9), which is a bit optimistic. The real loss of neutrons might be more precisely calculated and then included in the non-leakage probability.

- There is no way to calibrate the fuel evolution of natural Thorium because no PWR works with Thorium. So the results are inaccurate and might be improved.
  Now, it can be considered that natural Thorium is a better fuel than natural Uranium simply because, from its evolution, 4 possibilities of successive fission exist (via U233, U235, Pu239 and Pu241) versus 2 for U238 (Pu239 and Pu241), before finishing as a minor actinide (i.e. Cm or Am).

- This fusion reactor is mainly based on the Helias 5-B fusion reactor (project) for the half-toruses. It would be necessary to determine if the global confinement time of this “racetrack” fusion reactor (and consequently the mechanical gain Q) is correct, as no turbulent radial diffusion behavior is supposed to occur in the straight parts. Moreover, two hypotheses are implicitly taken for the straight parts:
  - The loss of energy is supposed only due to particles losses and not to heat loss through the first wall. However, the loss of energy by particles losses is probably conservative.
  - The particles lost on the wall are supposed to be the sole ions and not the electrons, as the heat loss which is mainly due to ions.

- It would be necessary to determine if the magnetic coils for the straight parts of the “racetrack” can be circular, as supposed. A complex geometry of the magnetic coils as for the torus would cause a problem to place the fuel assemblies.

- The divertor is supposed only be placed along both half-toruses but not along the straight parts, this because a divertor on the straight parts would reduce the room for fuel assemblies, normally placed around the fusion reactor. It would be necessary to confirm this possibility.

- The plasma heating using injection of charged particles (D+ and electrons), as taken into account in this document for the calculations, must be transformed by a set of real plasma heating systems as described in §1.1.

- The complex problems of plasma neutrality control, molecular vacuum pumping of neutral particles and management of the neutralized particles must be considered.

5.7 Possible slight improvement of the hybrid reactor by injection of Tritium with the Deuterium

This possibility is proposed for information because the gain is weak and the breeding system is complex.

Introduction

Even if not indispensable, several percents of Tritium in the Deuterium would boost the fusion reactor.
In figure 5, it appears that, at 29 keV, the D-T reaction rate is about 80 times the D-D one (and so about 160 times the D-D fusion rate) and even much more for D-He3. The production of Tritium by the PWR associated with the fusion reactor would be very weak, i.e. several g per year, mainly due to “ternary fissions” in the Uranium fuel.

Now, a part of the Tritium injected (if any) and produced by D-D reaction is lost on the wall. This tritium mixed with other gas is recovered thanks to the vacuum pumping system. Afterwards, the Tritium could be separated from the recovered gas and stored.

Another more efficient solution would be to install a Tritium breeding blanket (see [30] for information), with its Tritium extraction system, at the exterior of the fission reactor between the reflector and the 316LN exterior wall. Of course, this Lithium blanket will be separated from the water due to an exothermic reaction between both.

The fission neutrons crossing the reflector are slowed down by this one. In the Lithium blanket, these thermal neutrons will generate Tritium, according to the fission reaction between a thermal neutron and the Lithium 6 ([31]): n + ^{6}Li → ^{4}He + T + 4.78 MeV (kinetic energy).
The Tritium will be separated from Helium and stored.
Note that there is no possibility to multiply neutrons inside the Lithium blanket through n→2n reactions with Beryllium or Lead, because the neutrons are thermal.
The generated Tritium will be added to the Deuterium gas before being injected as D+ and T+ ions, into the fusion reactor. So the rate of D-T fusion neutrons will be increased. This gain will permit to increase the net electrical power.

See the sectional view in figure 11 and compare with the sectional view in figure 2.

The TBR (Tritium Breeding Ratio) of this Lithium blanket is close but inferior to 1, i.e. a neutron entering the Lithium blanket will generate, on average, a bit less than one atom of Tritium.

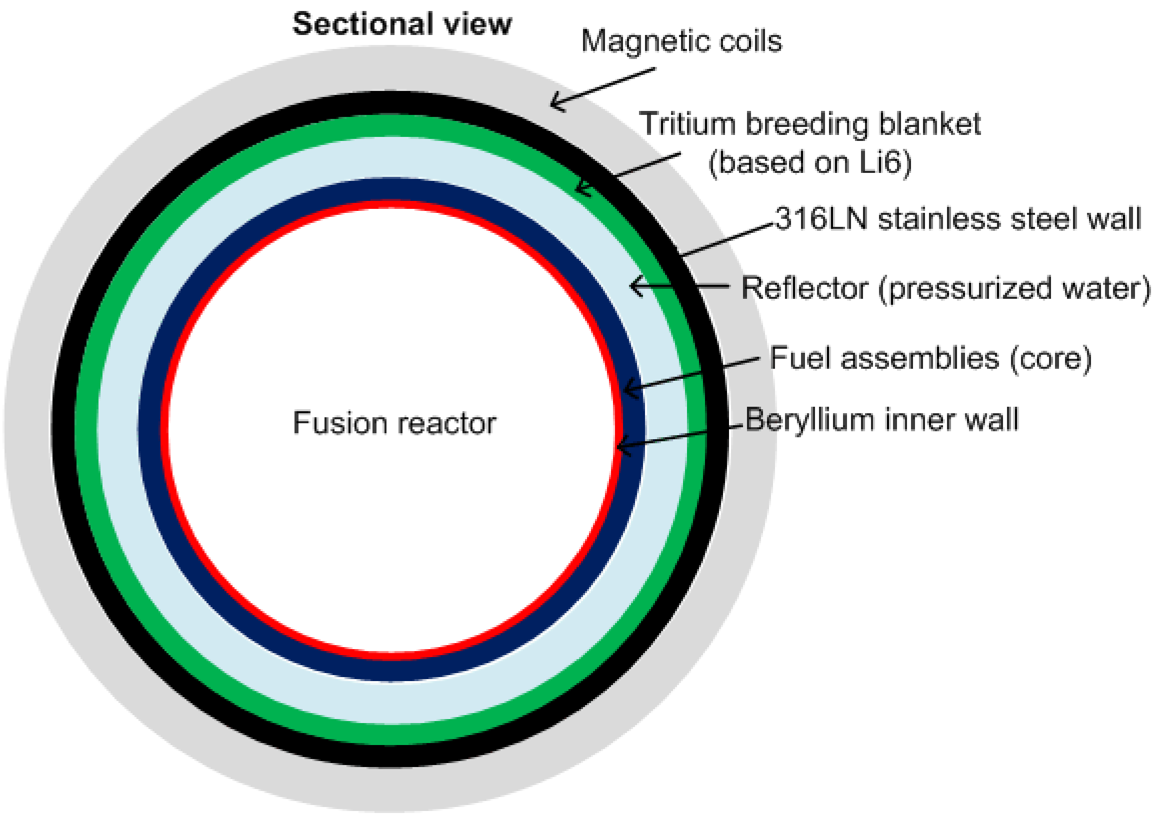


Figure 11. Sectional view at the level of the straight parts

<u>Estimation of the gain due to this Tritium contribution</u>
The rate of atoms of Tritium (per s and per m^3 of fusion reactor), generated by the Lithium blanket can be written: $QTgen\_m3 = Qfnm3 \times Kn \times TBR$ (equation 5.1)
With Qfnm3 the rate of fusion neutrons (at 2.45 and 14.06 MeV) generated per s and per m^3 of fusion reactor (§3.5): $Qfnm3 = QfDDm3/2 + QfDTm3$ (equation 3.9)

From §4.2, §4.6 and §4.9, it can be calculated the Kn factor giving the number of fission neutrons reaching the Lithium blanket for one fusion neutron: $Kn = \frac{10}{11} \times \frac{Ptr\_Be}{(1-K\_eff)} \times \frac{K\_eff}{Mean\_Nu} \times (1-Pnl)$ (equation 5.1)

Ptr_Be (§4.2) varies between 0.915 with 100% of 2.45 MeV neutrons and 1.313 with 100% of 14.06 MeV neutrons. (1-Pnl) represents the probability of a fission neutron to leak the fission reactor with $Pnl = \frac{K\_eff}{K\_\inf inite}$ (from §4.9). 10/11 is the ratio between the fusion volume of the straight parts and the total fusion volume.

$\frac{1}{(1-K\_eff)} \times \frac{K\_eff}{Mean\_Nu}$ is the neutrons multiplication factor due to the sub-critical fission reactor. It is reminded that the maximum K_eff has been supposed equal to 0.9 in §4.5.

Now according to the §2.2.6 of [1] and without considering the charge exchanges (§3.4), it can be written, at equilibrium, when the loss of Tritium is equal to the gain of Tritium:

$QiTm3 + QTgen\_m3 = (\frac{nT}{TD\exp}) + (nT \times \gamma\, fDT) = nT \times \left[(\frac{1}{TD\exp}) + \gamma\, fDT \right]$ so it can be deduced the density of tritium atoms nT: $nT = \frac{QiTm3 + QTgen\_m3}{(1/TD\exp) + \gamma\, fDT}$ (equation 5.2), from which it will be calculated the rate of D-T fusions (QfDTm3, §2.2.6 of [1]), which is proportional to nT. The larger is QTgen_m3 and hence Kn, the larger is nT and the rate of D-T fusions QfDTm3. Note that Qfnm3 depends on QfDTm3, so the process is iterative and converges towards a given value.

The problem is that the fusion reactor is calculated before the fission reactor, so Kn is not known.
To avoid a too complex calculation, it will be supposed a Kn_supposed factor. Then the calculation will be done by the program. The real average Kn_average will be given once the fission reactor calculated. Kn_supposed must be close to Kn_average.

A basic value for Kn is 0.16 with Ptr_Be=1, K_eff=0.9, Mean_Nu=2.5, Pnl=0.95, which is not much. To have a larger Kn, if K_infinite is sufficient large, the possibilities are:

- To increase the maximum K_eff, for example up to 0.95 instead 0.9. Now this would decrease the margin compared to the critical state (K_eff=1),
- To increase (1-Pnl) the probability of leak, by reducing the core thickness.

Moreover, a more accurate calculation of the neutrons multiplication in the Beryllium wall than the one, conservative, done in §4.2 will be welcome.

<u>Example</u>
First, let's suppose that, including the Tritium recovered by the vacuum pumping, the global TBR reaches 1. Now, let's suppose a radius Rp=100 cm. With the K_eff limited to 0.9, the average net electrical power is equal to 46 MW, for a mechanical gain Q equal to 0.062.

With the generation of Tritium in service (i.e. TEST_WITH_ADDED_TRITIUM_TO_THE_DEUTERIUM = TRUE in the program):

- If Kn_supposed=0 (no Tritium generated) and K_eff limited to 0.95, the average net electrical power is equal to 163 MW, with the same mechanical gain Q (=0.062).
- For Kn_supposed=Kn_average=0.214 and TBR=1, the average net power is equal to 179 MW, for a mechanical gain Q equal to 0.066.

The gain on the net power is equal to 16 MW, which is weak, for such complex equipment.

<u>Lithium blanket inside the both half-toruses</u>
To increase the Tritium production, a Lithium blanket could also be installed inside both half-toruses to generate Tritium. The rate of atoms of Tritium, generated by this Lithium blanket would be:

$QT'gen\_m3 = Qfnm3 \times K'n \times TBR$, with the K'n factor giving the number of fission neutrons reaching the Lithium blanket for one fusion neutron, for both half-toruses: $K'n = \frac{1}{11} \times Ptr\_Be$

In that case, the space just above the interior beryllium wall will be filled with light water (cooled by an exterior system) to slow down and thermalize the neutrons up to the Lithium blanket.

5.8 Other improvements of the hybrid reactor

Below is a non-exhaustive list of possible improvements to this hybrid reactor.

- In [32] pages 153 and following, to accelerate the conversion of fertile materials into fissile materials, it is proposed a PWR in which the moderator fluid would be heavy water instead light water to make work the reactor in a faster domain than the thermal one, at a price of a K_eff decrease. Heavy water moderates less efficiently than light water but it absorbs much fewer neutrons compared to light water (see [21] pages 144, 178 and 464). In fact, according to the time in the cycle, a mixture of heavy and light water is recommended: it would be added light water with the increase of burn-up to compensate for K_eff, i.e. heavy water at the beginning and light water at the end of the cycle. However, as the 20 cm of heavy water would be poor as reflector, a way to avoid bigger neutrons leaks might be found (bigger reactor, graphite layer, independent pressurized light water layer, etc). Moreover, as heavy water will make decrease K_eff and hence the fission multiplication 1/(1-K_eff), some compromise might be found (if any).

- Heat produced at a low temperature, let's say between 60 °C and the primary temperature of 257 °C, is considered lost in the heat sink, i.e. the cold source (sea, river, etc). This heat can come from the cooling system of the half-toruses or from the particles injection system as 20% of the electrical energy used is supposed lost in heat. In fact, these sources of heat could be used either in the secondary cycle for the heaters or to supply electricity using a dedicated working for low temperatures sources, which principle is described for example in [33]. Moreover such system, using, for example, the hot water of the spent fuel pit (where spent fuel is stored) carried to let's say 90°C, could supply a minimum amount of electricity for the plant in case of H3 accident (§5.9). Indeed, this electric power would be used for the control room, the electronic control systems, the instrumentation and for a minimum cooling system of the core and the spent fuel.

5.9 Safety

Main nuclear accident to mitigate

This preliminary accident analysis is to be considered very rough, as it is a very complex subject.

The K_eff multiplication factor being limited to a maximum of 0.9 and as there are no emergency shutdown rods, no criticality accident and no failure of the emergency trip have to be considered. So the probability of occurrence of accidents supposing partial or total failures of the emergency trip and leading to core meltdown, such as Anticipated Transient Without Scram (ATWS), Loss Of Coolant Accident (LOCA) and Steam Line Break (SLB), must be considered as much lower.
Note that if the fission reactor must be stopped, it will be stopped by a fusion reactor drop which is the sources of neutrons. Practically it will be done by either stopping the particles injection and, possibly, by injecting neutral gas to extinguish fusions. In this case, the fission reactor will be naturally stopped in several seconds.

An analysis based on a Probabilistic Risk Assessment of the most probable (or better say the least improbable) accidents for a PWR is given in [34] page 19. It appears that without APRP (LOCA), ATWS and RTV (SLB), the most probable accident (at 75%) would be the H3 accident (Total loss of external and internal electrical power). To date, several systems exist to mitigate this accident (turbine-driven auxiliary pumps, passive cooling system, exterior diesel generator sets, exterior cold sources, etc). Consequently, the core meltdown probability is extremely low.

Fissile Uranium and Plutonium proliferation

As explained from §5.1 to §5.4, for this type of reactor, the fuel reprocessing might be such that no

separation between Uranium, Plutonium and Thorium be made. In that case, no direct risk with the fuel would exist.
Moreover, no enrichment is needed for this type of reactor, which also limits proliferation.

6 Conclusion

This relatively simple hybrid reactor has been described in §2. Afterwards, it has been modeled:

- The D-D fusion reactor in §3 (as a complement to [1]).
- The PWR fission reactor in §4.

From §5.1 to §5.4, it has been presented the main results about the working of this reactor with different types of fuel, i.e. natural Uranium, depleted Uranium, natural Thorium or a 50/50 mixture of natural Uranium and natural Thorium. It appears that this reactor is able to successfully incinerate any natural or depleted nuclear fuel for thousands of years of world consumption as shown in §1.1.
The net electrical power with the Thorium fuel, at equilibrium, is better than the one with the Uranium fuel, even if at the beginning of the Thorium incineration, this reactor is an electricity consumer (§5.3). It must also be added that the Thorium fuel produces much fewer minor actinides (americium and curium) than the Uranium fuel, which would permit to space the reloading/reprocessing operations (see §4.11) and to reduce the volume and the toxicity of waste.

The net electrical power roughly depends on the reactor volume. For example, in this paper, the default configuration is a plant producing about 1400 MWe, as a big fission plant, the fusion radius being equal to 2 m. According to the same models, the power would be around 440 MWe for a fusion radius of 1.5 m and around 3000 MWe for a fusion radius of 2.5 m, but equal to 46 MWe for a fusion radius of 1 m.
Now, for a radius of 2 m, the reactor is 200 m long, which is rather large. To reduce the size of such reactor for the same net electrical power, one could increase the Beta factor (§3.2) if possible or, more slightly, the maximum K_eff (§4.5).

The expected life (§5.5) is equal or superior to 100 years, due to the low neutrons flux on the first wall in Beryllium and on the exterior wall in 316LN.

This reactor is safer than a PWR reactor due to its subcriticality (§5.9). Moreover, the absence of U235 enrichment and the uselessness to separate the Uranium, the Thorium and the Plutonium in the reprocessing operation makes the fuel safer in regards to proliferation. It must also be added that the final waste is constituted by fission products, minor actinides (mainly americium and curium), without any spent fuel (i.e. Uranium, Plutonium or Thorium).

Points to deepen and improvements of this reactor from §5.5 to §5.8 complete the analysis of this hybrid reactor.

The presented results are orders of magnitude due to a simplified modeling but they are sufficient to consider the possibility of a functional hybrid reactor.

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