

Proposal of a Deuterium-Tritium Fusion/PWR Fission Hybrid Reactor

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Abstract

This article proposes to associate a Deuterium-Tritium (D-T) fusion reactor with a PWR (fission Pressurized Water Reactor) in a hybrid reactor used as a power plant. A previous proposal of a D-D fusion associated with a PWR has been proposed by the author. The drawback of such D-D hybrid reactor is its enormous size (200 m). A D-T hybrid reactor is much smaller for the same power, but it is more complex due to the necessity to supply the necessary Tritium. In this proposal, the Tritium is produced thanks to leaky thermal fission neutrons rather than fusion neutrons. In this way, the production of Tritium is larger than the consumption of Tritium. Even if the mechanical gain (Q factor) of the D-T fusion reactor is below unity and consequently consumes more energy than it supplies, due to the high energy amplification factor of the PWR fission reactor, the global gain is widely superior to 1. As the energy supplied by the fusion reactor is relatively low, the problems of heat flux and neutron damage connected with materials are reduced. This type of reactor is able to incinerate natural Uranium and depleted Uranium (waste issued from enrichment plants). For natural thorium, which has no initial reactivity, a “primer” is necessary to start the reactor. A good “primer” is 1.8% of military Uranium (at 90% U235) mixed into 98.2% of natural Thorium. Once started, the natural Thorium is the best tested fuel. The high-level radioactive waste (stored spent fuel), cleansed of its minor actinides and of its fission products, taking advantage of the fissile materials in it, is the most reactive fuel tested in this paper. This type of reactor could constitute a source of energy for several thousand years, because it is about 90 times more efficient than 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. In this paper, it is targeted a reactor able to provide a net electric power of about 1000 MWe, as a standard fission power plant. For the fission part, the PWR technology is mature, even if the fuel assemblies will be more complex than the ones used in PWR. For the fusion part, it is based on a reasonable mechanical gain Q around 0.7 already reached by

certain tokamaks and, certainly in the near future, by Stellarators. The reference fusion model is supposed to be the Stellarator Wendelstein 7-X, but with an interior radius enlarged to increase the provided power. The working of this reactor is continuous, 24 hours a day.

Keywords

Fusion Reactor, Fission Reactor, Hybrid Reactor, Nuclear Power Plant, Nuclear Energy, Deuterium-Tritium Reactor, Deuterium, Tritium, Stellarator, Tokamak, Power Plant, PWR, Tritium Production, Uranium, Thorium, Fissile Materials, High-Level Radioactive Waste, Stored Spent Fuel

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-Tritium (D-T) fusion reactor. The reference fusion reactor is supposed to be a simplified model of the Stellarator Wendelstein 7-X [1]-[3]:
 - o Major plasma radius (r): 5.5 m.
 - o Average minor plasma radius (R_p): 0.53 m.
 - o Plasma volume: 30 m³.
 - o Magnetic induction on axis (by superconducting coils): 3 T.
 - o Average plasma temperature (goal): 100 million degrees (or $T_i = T_e = 8.6$ keV) [1].
 - o Maximum average plasma density (goal): 3E20 particles/m³ [1].
 - o Energy confinement time τ_e (goal): 0.2 s [2].
 - o Exterior heating mainly by ECRH (10 MW) (electron cyclotron resonance heating), by ICRH (4 MW) (ion cyclotron resonance heating) and by NBI (5 MW) (neutral beam injection): so presently a total of 19 MW [3]. 8 MW of ECRH will be added in 2030 [4], so 27 MW will be available in 2030. This maximum value of 27 MW will be taken into account in this document.
 - o Fuel is injected via frozen pellets.

Note that a Tokamak of the same global characteristics could have been considered. However, the Tokamak working is not continuous, but rather works by long pulses, which is a drawback. So the sole Stellarator is considered in this paper. Now, if the complex coil system of a Stellarator cannot house a fission reactor, a Tokamak might be envisaged.

- A PWR fission reactor is used in a subcritical state. Note that a PWR is normally critical to work. This PWR is supplied with neutrons by the fusion reactor. Thanks to fission reactions, the PWR amplifies the fusion power by a factor of around 78. Thanks to this gain, the fusion reactor does not need to be powerful, as it is just a neutron generator, the power mainly coming from the PWR. This PWR can “incinerate” a fertile material, mainly natural Uranium

(*i.e.* U238) but also natural Thorium (*i.e.* Th232). It works as a subcritical breeder reactor.

However, with natural Thorium as fuel, a “primer” is necessary to start the plant (see §6.4 for details).

Note that a change in the fission reactor, such as the reactivity or the temperature, has no impact on the fusion reactor. Conversely, the fusion reactor controls the fission reactor via the fusion neutron rate.

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, particle diversion on the “Divertor” (to “clear” the plasma), radiation hygiene, possible instabilities and ways 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 calculations in this paper are relatively rough, with many hypotheses and simplifications (see §6.5). The goal of this paper is not to study a specialized aspect of the reactor, such as neutronics, for example, but it is to calculate the whole reactor (fusion + fission) and to present the results and the possibilities of such a reactor. For this it is proposed a simplified physical model of this hybrid reactor. A small program called “D_T_PWR_hybrid_reactor” (§1.4), based on the model developed in [5] and in this article, is proposed, with its Delphi 6 source code. Thanks to this model, any D-T/PWR hybrid reactor can be roughly designed.

Interest of such hybrid reactor

By using the proposed hybrid reactor in its default configuration, it will be found that the net electrical energy produced by one ton of natural Uranium (in the form of UO₂) is equal to 2.09 E16 J/t and 2.08 E16 J/t for depleted Uranium. With 6.078 million tons of natural Uranium and 1.2 million tons of depleted Uranium ([5] §1.1), the world consumption of electricity would be covered for:

- 1197 years by Natural Uranium.
- 235 years by depleted Uranium.

The total makes 1432 years. Now, with the conventional and unconventional reserves of natural Uranium ([5] §1.1), the time during which Uranium could supply all the world with electricity would rather be superior to 10,000 years, and this without taking into account the natural Thorium (§6.4) and the high-level radioactive waste (§6.3).

Notations and units

In a formula, the $\times$ 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 1.60219×10^{-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 $1 \text{E}6 \times 3600 \times 24 = 8.64 \text{E}10$ J. It is sometimes used “GWd/t” rather than “MWd/t”, with $1 \text{ GWd/t} = 1000 \text{ MWd/t}$.

A temperature can be expressed either in °K or in eV (or keV) with $1 \text{ eV} \equiv 11604.5^\circ\text{K}$.

1.2. Method of the Work

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

Step 2: From §3 to §5, the fusion and the fission reactors working are modeled using classical formulas of physics. The simple fusion model is described in §3. The sizing of the whole reactor is made in §4. The fission model developed in §5 is strongly based on [5]. So only the differences with [5] are described. The global working of the reactor is shown in **Figure 3** (§2.1).

Step 3: Once the physical models are achieved, they will be used in §6 to estimate this reactor with four types of fuel. A discussion about results and points to deepen follows. In §7, conclusions are drawn.

1.3. List of the Main Variables and Acronyms Used in This Article

Below are the main variables used throughout this article (local variables are not listed)

B	Toroidal magnetic field (T)
Tp	Average temperature of the plasma in eV or in °K
K	Neutrons multiplication factor (in K_{infinite} and K_{eff})
nD	Deuterium ions density (number of Deuterium ions per m^3)
nT	Tritium ions density (number of Tritium ions per m^3), with $nT = nD$
ni	Deuterium + Tritium density (number of ions per m^3). Note that $n_i = nD + nT$
ne	Electrons density (number of electrons per m^3). Note that $n_e = n_i$
pht	Per hundred thousand, <i>i.e.</i> in $1/100,000$
Q	Mechanical gain (<i>i.e.</i> fusion power/heating power), without dimension
Rp	Interior pipe radius (m) of the fusion reactor
r	Major radius of the reactor (m)

Below are the acronyms used:

“D-D” for “Deuterium-Deuterium”.

“D-T” for “Deuterium-Tritium”.

“PWR” for “Pressurized Water Reactor”.

“EPR” (“Evolutionary Pressurized Reactor”) for the last PWR generation in France.

Note: the terms “incinerate” and “burn” the fuel are figuratively used here, because the process is nuclear, not chemical.

1.4. “D_T_PWR_Hybrid_Reactor” Program Based on the Fusion and Fission Physical Models

It is proposed the program called “D_T_PWR_hybrid_reactor” V1.0, which implements the physical models described in this article and in [5]. The executable program with its Delphi 6 source code can be downloaded from this direct link: http://f6cte.free.fr/D_T_PWR_hybrid_reactor.zip. It is enough to paste this address in your Internet browser. Download the file. Create a folder (C:\D_T_PWR_hybrid_reactor for example), unzip the D_T_PWR_hybrid_reactor.zip file in it and then start D_T_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.

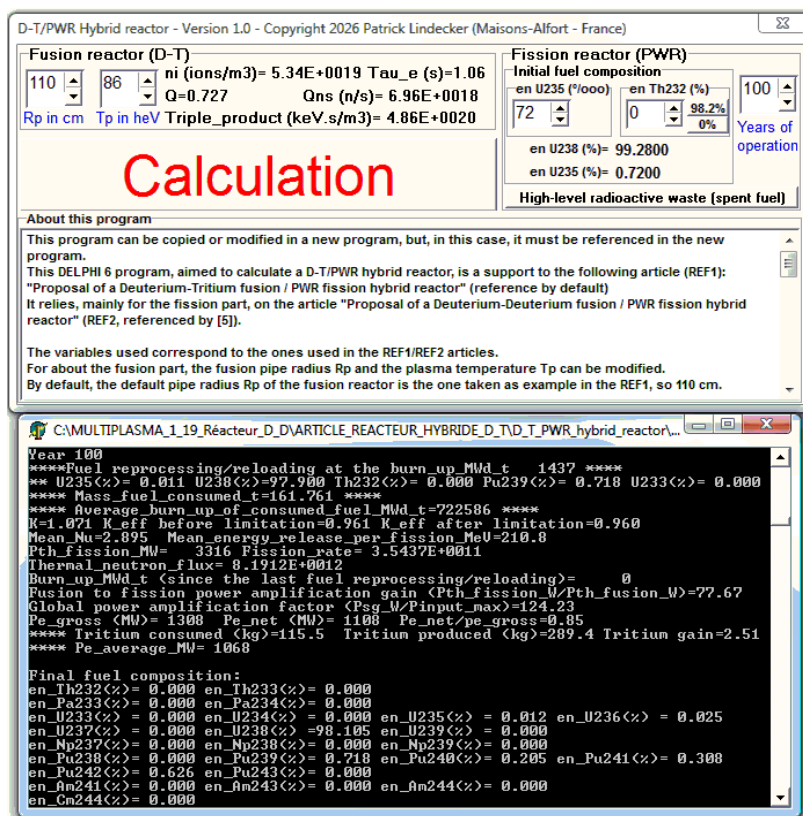


Figure 1. D_T_PWR_hybrid_reactor V1.0 program snapshot running its default configuration.

2. Description of the Hybrid Reactor

2.1. Generalities

It is composed of a fusion reactor along the torus axis. It is surrounded by the fission reactor, including the Tritium breeding blanket and the exterior wall (reactor vessel body), as shown in **Figure 2**.

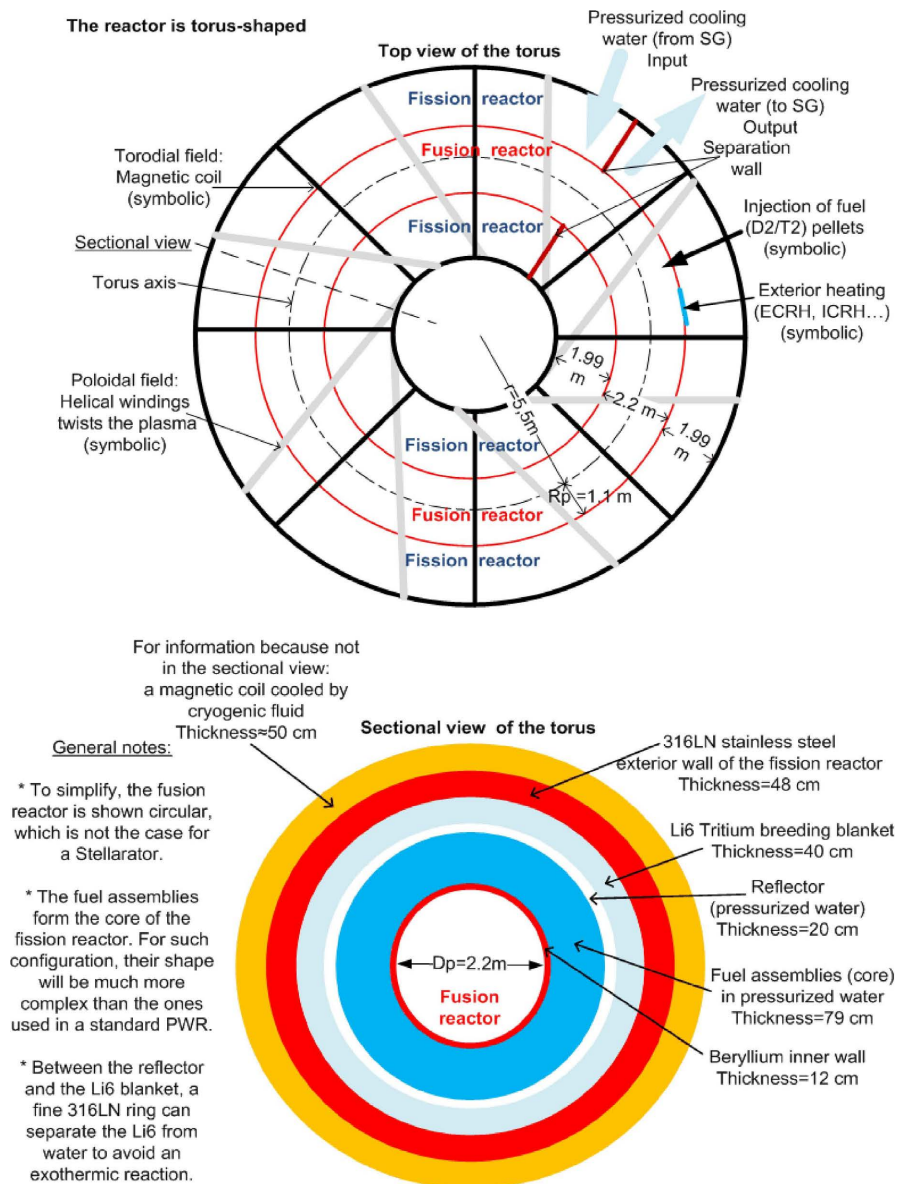


Figure 2. D-T/PWR fusion reactor principle diagram.

Note that the thick 316LN exterior wall also protects the magnetic coils from heat and neutrons.

For a PWR, this wall would be called the “Reactor vessel body”.

Figure 3 shows, in a general way, how the hybrid reactor energy balance is taken into account.

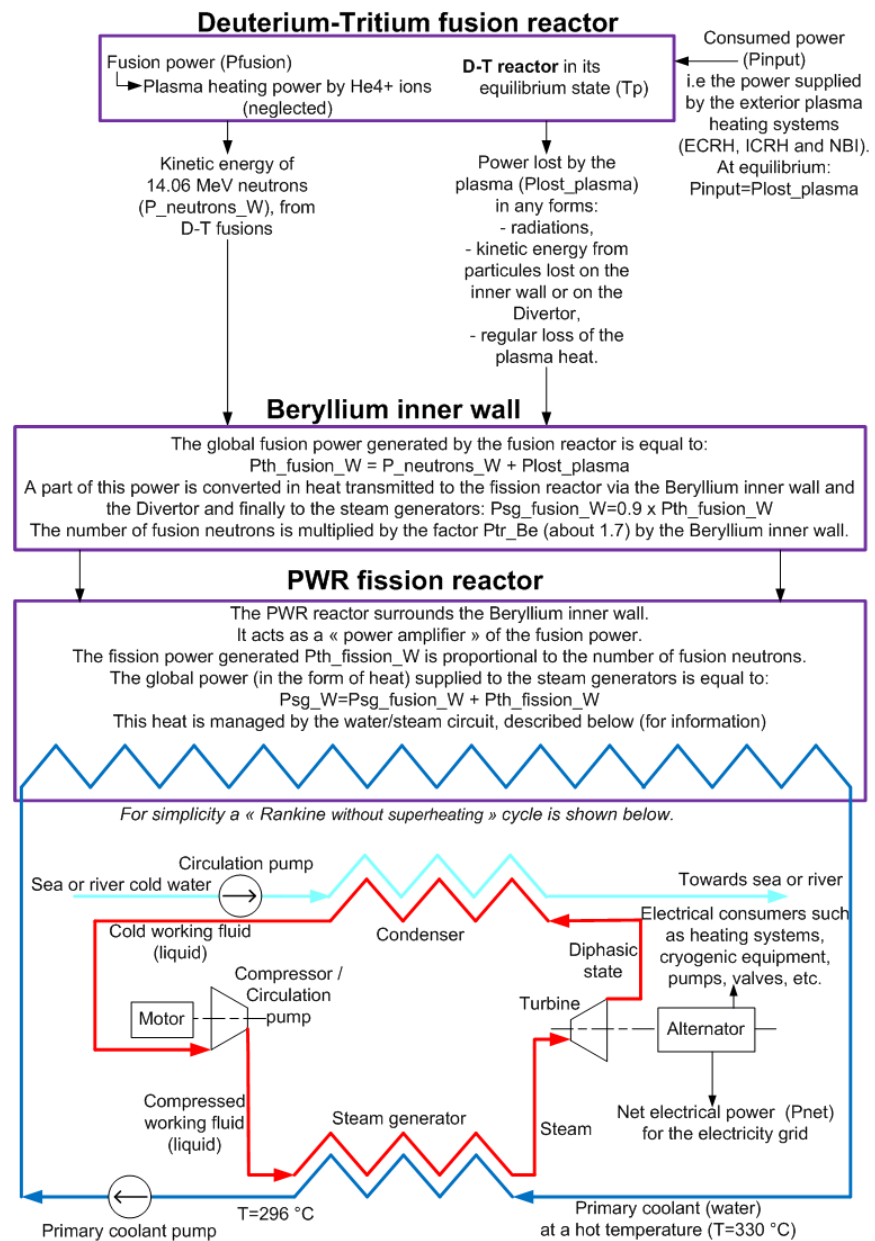


Figure 3. D-T/PWR hybrid reactor energy balance.

2.2. Presentation and Working of the Fusion Reactor

Presentation and working of the fusion reactor

The plasma is heated up and then maintained at the equilibrium average temperature T_p , which is, to simplify, equal to the average temperatures of the electrons (T_e) and the ions (T_i). The plasma is thermalized. The Coulomb collisions between D^+/T^+ ions and electrons permit a permanent exchange of energy between these particles. The plasma is heated, for a small part by the fusion products $\text{He}4+$ ions (neglected here), but mainly by the exterior plasma heating system. This heating system compensates for the small radiation power and, above all, for the loss of heat and particles ($P_{\text{lost_plasma}}$).

In this type of fusion reactor the Q factor (*i.e.*, fusion power/heating power ratio) is low (<1), far away from the target aimed by true fusion reactors ($Q = 10$ for ITER, for example). So this fusion reactor must rather be considered as a neutrons generator.

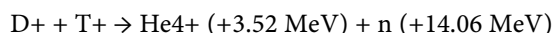
The toroidal magnetic field (B) must be axial relatively to the pipe, and maximum to confine particles (electrons + ions). The present industrial maximum B limit for superconducting coils is 5 T (Tesla). This field is equal to 3 T for the Stellarator Wendelstein 7-X.

At fusion densities, a poloidal field is indispensable to limit the particles shift inside loops.

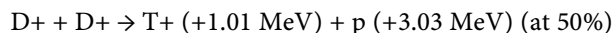
Note that the magnetic surfaces are complex (see [6] for an example).

Moreover, to simplify, it will also be assumed that the fusion reactor is circular (as shown in **Figure 2**), even if it is neither the case for Stellarators (see [6]), nor the case for modern Tokamaks.

There are, mainly, fusions between Deuterium and Tritium nuclei:



There are also very rare fusions between Deuterium nuclei, as specified below. These fusions will be neglected.



In **Figure 4**, the reactivities (written “ $\langle \sigma \times w \rangle$ ”) for the D-T and the D-D fusion reactions from [7] are shown.

Due to the weak specific fusion power, concerning the Beryllium wall, the mechanical resistance in front of 14.1 MeV neutron flow and in front of the heat flow is much less critical than for a true fusion reactor (see SHP and SNP in §3 and §4.1).

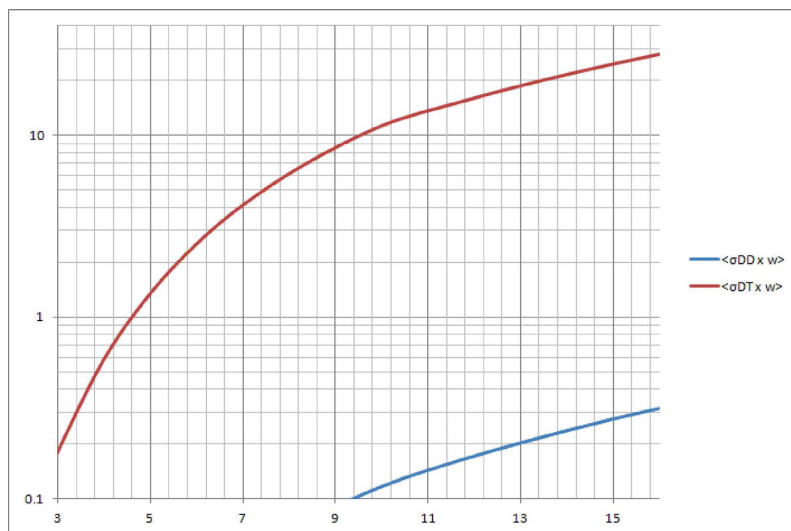


Figure 4. Reactivities of the D-D and D-T fusion reactions. The abscissa is the equilibrium average temperature of the plasma (T_p) in keV and the ordinate is the reactivity in $m^3/s \times 1E-23$.

It is reminded that contrary to the fission neutrons which follow an energy spectrum, the fusion neutrons have discrete values (14.06 MeV for the D-T fusion, for example).

2.3. Presentation and Working of the Fission Reactor

The amplification of energy by the fission reactor is first due to the fission energy (about 200 MeV) supplied by each fission reaction (triggered by a thermal neutron) compared to the 17.58 MeV of a D-T fusion reaction, and, secondly, by the neutrons amplification.

Indeed, the 14.06 MeV fusion neutrons issued from the fusion reactor, cross the Beryllium inner wall (12 cm thickness) with very little loss and a certain amplification. This flow of neutrons is then amplified by the sub-critical fission reactions in the fuel (79 cm thickness of fuel rods).

The pressurized light water serves as a neutron decelerator and a cooling fluid, transporting the heat produced in the fission reactor to the steam generators (see **Figure 2** and **Figure 3**).

The reflector is a layer of 20 cm thickness of water (about the thickness used on PWRs) which goal is to limit the neutron leak (by reflection of a big part of the neutrons) and to slow down neutrons. Then, thanks to the big number of fission neutrons that leak from the fission reactor, the Tritium breeding blanket (40 cm thickness) supplies the necessary Tritium for the continuous working of the fusion reactor. This necessary leak of fission neutrons will decrease the neutron multiplication factor K_{eff} , so some arrangement must be found.

The 316LN non-magnetic stainless steel exterior wall, with a thickness of 48 cm, withstands the pressure (154 bar gauge) and protects the magnetic coils from heat and fission neutrons.

Why this type of fission reactor

The PWR is a compact and mastered reactor. For details, see [5] §2.3.

3. Physical Model of the D-T Fusion Reactor

The physical fusion model is very simplified but sufficient relatively to the objective of this paper. As said in §1.1, the fusion reactor is supposed to be the Stellarator Wendelstein 7-X with an enlarged plasma radius, modified to house the fission reactor installed around the fusion reactor. This one works with a mixture of 50% D/50% T, which is not expected to be tested on the Wendelstein 7-X. The mechanical gain Q (*i.e.* fusion power/heating power) being low, *i.e.* inferior to 1, the plasma heating by He4+ (20% of the fusion power, so here less than 20% of the necessary heating power) will be neglected in the rest of the document, which is penalizing.

The fusion reactor is supposed to work in the following conditions which are supposed achievable (see §1.1):

- Exterior plasma heating (Pinput): 27 MW maximum (5 MW NBI, 4 MW ICRH and 18 MW ECRH).

- Major plasma radius (r): 5.5 m.
- Initial plasma volume (Vp): 30 m³ for an initial average plasma radius (supposed equal to the fusion reactor radius) Rp = 0.53 m.
- Energy confinement time (τ_e): 0.2 s.
- Average plasma temperature of the plasma ($\langle T_p \rangle$) with electrons and ions at the same average temperature of 8.6 keV (99.80 E6 °K).

In this paper, it is implicitly supposed that the Stellarator Wendelstein 7-X, with 27 MW of exterior plasma heating, will be able to reach a confinement time of 0.2 s for an average plasma temperature (T_p) equal to 8.6 keV, as expected.

From the ions density (n_i) equal to the electrons density (n_e), the plasma energy Eplasma of Vp m³ of plasma is equal to $E_{\text{plasma}} = V_p \times 3 \times \langle n_i \times T_p \rangle \times K_b$ (with K_b the Boltzmann constant = 1.3806E-23 J/K and T_p in °K).

Note that $\langle n_i \times T_p \rangle$ is the average value for all the plasma. According to [8] page 21, if the profiles of n_i and T_p are both supposed parabolic

$$\langle n_i \times T_p \rangle = \frac{n_i \text{ max at the center} \times T_p \text{ max at the center}}{3} \quad \text{and} \quad \langle n_i \times T_p \rangle = \langle n_i \rangle \times \langle T_p \rangle,$$

with T_p max at the center equal to $8.6 \text{ keV} \times \sqrt{3} = 14.9 \text{ keV}$ which can be considered as moderate. In this equation, n_i is in ions/m³ and T_p is in °K.

The exterior plasma heating power (Pinput) is equal to

$$P_{\text{input}} = \frac{E_{\text{plasma}}}{\tau_e} = \frac{V_p \times 3 \times \langle n_i \rangle \times \langle T_p \rangle \times K_b}{\tau_e}.$$

Such heating power is produced with an average efficiency, from the electric source to the effective plasma heating, supposed equal to 0.39. So the necessary electric power, for the maximum exterior plasma heating power (27 MW) is equal to:

$$\text{to: } P_{e_input_W} = \frac{P_{\text{input_max}}}{0.39}$$

With $P_{\text{input_max}} = 27 \text{ E6 W}$, then $P_{e_input_W} = 6.92 \text{ E7 W}$ (*i.e.* 69.2 MWe)

It can be deduced $\langle n_i \rangle$:

$$\langle n_i \rangle = \frac{P_{\text{input_max}} \times \tau_e}{V_p \times 3 \times \langle T_p \rangle \times K_b} \quad (\text{Equation 3.1})$$

Note that the average electrons density ($\langle n_e \rangle$) is equal to the average ions density ($\langle n_i \rangle$) and that n_i max at the center will be equal to $\langle n_i \rangle \times \sqrt{3}$ ions/m³.

The rate per second of fusion reactions and, consequently, the rate per second of 14.06 MeV neutrons (Qns) produced by the reactor is equal to:

$$Q_{\text{ns}} = V_p \times n_D \times n_T \times \langle \sigma.v \rangle = V_p \times \frac{n_i^2 \times \langle \sigma.v \rangle}{4} \quad (\text{Equation 3.2})$$

with $n_D = n_T = n_i/2$ and $\langle \sigma.v \rangle$ the mean fusion reactivity equal to 7.55E-23 m³/s at 8.6 keV (see **Figure 4**).

A D-T fusion reaction supplies a kinetic energy of 17.58 MeV (see §2.2), or $17.58 \text{ E6} \times 1.60219 \text{ E-19} = 2.817 \text{ E-12 J}$. So the fusion power (Pfusion) in W supplied by the reactor is equal to:

$$P_{\text{fusion}} = Q_{\text{ns}} \times 2.817 \text{ E-12} \quad (\text{Equation 3.3})$$

Note that:

- The mechanical gain Q is equal to

$$Q = \frac{P_{\text{fusion}}}{P_{\text{input_max}}} \quad (\text{Equation 3.4})$$

normally inferior to 1 for a hybrid reactor, to compare with $Q = 10$ for ITER.

- The triple product is equal to

$$T_e \times \langle n_i \times T_p \rangle = T_e \times \langle n_i \rangle \times \langle T_p \rangle (\text{keV}) \quad (\text{Equation 3.5})$$

This one will be much weaker than the objective of a functional true fusion reactor (*i.e.* about $3 \text{ E}21 \text{ keV}\cdot\text{s}/\text{m}^3$).

So for a hybrid reactor, the fusion reactor is just a powerful neutron generator, injecting Q_{ns} neutrons per second.

A 14.06 MeV neutron supplies a kinetic energy of $14.06\text{E}6 \times 1.60219\text{E}-19 = 2.253\text{E}-12 \text{ J}$. So the neutron power ($P_{\text{neutrons_W}}$) in W supplied by the fusion reactor is equal to:

$$P_{\text{neutrons_W}} = Q_{\text{ns}} \times 2.253\text{E}-12 \quad (\text{Equation 3.6})$$

At equilibrium, the power lost by the plasma ($P_{\text{lost_plasma}}$ in **Figure 3**) is equal to the consumed plasma heating power $P_{\text{input_max}}$, by neglecting the $\text{He}4+$ ions heating. So the global power generated by the fusion reactor associated to the exterior heating systems is equal to:

$$P_{\text{th_fusion_W}} = P_{\text{neutrons_W}} + P_{\text{input_max}} \quad (\text{Equation 3.7})$$

Note that, by hypothesis, 90% of this power is supposed used in the thermodynamic cycle.

The average surface heat power (SHP in MW/m^2) on the first wall is equal to:

$$\text{SHP} = \frac{P_{\text{input_max}} \times 1\text{E}-6}{(2 \times \pi \times R_p) \times (2 \times \pi \times r)} \quad (\text{Equation 3.8})$$

Note that SHP might preferably be inferior to about $0.4 \text{ MW}/\text{m}^2$.

The average surface neutrons power (SNP in MW/m^2) on the first wall is equal to:

$$\text{SNP} = \frac{P_{\text{neutrons_W}} \times 1\text{E}-6}{(2 \times \pi \times R_p) \times (2 \times \pi \times r)} \quad (\text{Equation 3.9})$$

Let's suppose that:

- Each 14.06 MeV neutron interacts one time with the first wall.
- The maximum sustainable fluence is equal to $2 \text{ MW}\cdot\text{year}/\text{m}^2$ corresponding to about 20 dpa, 20 dpa being the limit for modest additional Helium effects [9].

In these conditions, the service life (in years) of the first wall (Beryllium) might be limited to:

$$\text{Service_life} = \frac{2}{\text{SNP}} \quad (\text{Equation 3.10})$$

4. Global Simplified Working of This Hybrid Reactor and Sizing

The average fusion reactor radius of the Wendelstein 7-X (R_p) being equal to 0.53 m, the fusion power would be too small for a power plant, as the net electric power would only be equal to 110 MWe for a gross electric power of 199 MWe (§4.1). Now, the target is a standard power plant of 1000 MWe. For such power, the R_p radius must be equal to 1.1 m, instead of 0.53 m.

Note that the R_p radius can be modified in the “D_T_PWR_hybrid_reactor” program (§1.4).

Note also that the principle of extrapolation of a Stellarator model to another model has already been used, as for example, the Wendelstein 7-X for the Helias reactor concept.

According to the ISS04 scaling ([10] for example), all other things being equal: $\tau_e \sim R_p^{2.28}$. Knowing that initially the energy confinement time $\tau_e = 0.2$ s for $R_p = 0.53$ m (§3), this leads, for $R_p = 1.1$ m, to:

$$\tau_e = 0.2 \times \left(\frac{R_p}{0.53} \right)^{2.28} \quad (\text{Equation 4.1})$$

So $\tau_e = 1.06$ s for $R_p = 1.1$ m.

Moreover, knowing that initially the plasma volume $V_p = 30$ m³ for $R_p = 0.53$ m, this leads, for $R_p = 1.1$ m, to:

$$V_p = 30 \times \left(\frac{R_p}{0.53} \right)^2 \quad (\text{Equation 4.2})$$

So $V_p = 129.2$ m³ for $R_p = 1.1$ m.

Note: the magnetic field will remain equal to 3 T, but it will be assumed 5 T for the electrical consumption. For information, according to the ISS004 scaling [10], to bring the magnetic field B_t from 3 T to 5 T would give a gain on the confinement time (τ_e) equal to $\left(\frac{5}{3} \right)^{0.84} = 1.54$.

4.1. Global Working of a Power Plant Hybrid Reactor

This description is accompanied by a rough calculation of the hybrid reactor, to have orders of magnitude.

The fusion reactor supplies Q_{ns} neutrons per second and a thermal power $P_{th_fusion_W}$ (§3). The number of fissions is equal to Q_{ns} multiplied by a factor K_p (§5.4) equal to about 14.1 thanks to the Beryllium first wall (§4.2) and the fission reactor itself (§5). These $Q_{ns} \times K_p$ fissions are produced by thermal neutrons, *i.e.* neutrons thermalized by the moderator (water) and not captured by the fuel. Each fission produces an energy of about 200 MeV, *i.e.* $3.2E-11$ J. So the fission power is equal to:

$$P_{th_fission_W} = Q_{ns} \times K_p \times 3.2E-11 \quad (\text{Equation 4.3})$$

Supposing that 90% of the fusion power ($P_{th_fusion_W}$ in §3) is converted in

heat in the reactor, it is produced a global thermal power:

$$P_{sg_W} = 0.9 \times P_{th_fusion_W} + P_{th_fission_W} \quad (\text{Equation 4.4})$$

It is supposed that the water-steam circuit of the plant supplies electricity with a gross yield of 0.39 (as for EPR plants). So the gross electric power will be equal to

$$P_{e_gross_W} = P_{sg_W} \times 0.39 \quad (\text{Equation 4.5})$$

Now it is necessary to supply electricity to the cryogenic system, the pumps, valves, instrumentation, control equipment, etc. A reasonable hypothesis, including a necessary magnetic field being supposed equal to about 5 T, is that 10% of this gross power is necessary for these auxiliary systems, the heating systems (ECRH, ICRH, NBI) being excluded from these auxiliary systems. Consequently, the yield is reduced down to 0.351 ($=0.39 \times 0.9$) for the thermodynamic cycle, without considering the heating power ($P_{e_input_W}$ in §3).

So the net electric power will be equal to

$$P_{e_net_W} = P_{sg_W} \times 0.351 - P_{e_input_W} \quad (\text{Equation 4.6})$$

The ratio

$$\mu = \frac{P_{e_net_W}}{P_{e_gross_W}} \quad (\text{Equation 4.7})$$

must be superior to 0.8 for an acceptable power plant.

With $R_p = 1.1$ m, which is our hypothesis, $Q = 0.727$, Service_life of the Be wall = 30 years (§3), $P_{e_gross} = 1240$ MWe whereas $P_{e_net} = 1047$ MWe, so $\mu = 0.84$. Note that the SHP = 0.113 MW/m² which is weak (§3).

Notes:

- With $R_p = 0.53$ m (plasma radius of the Wendelstein 7-X stellarator), $Q = 0.112$, Service_life of the Be wall = 95 years, SHP = 0.235 MW/m², $P_{e_gross} = 199$ MWe and $P_{e_net} = 110$ MWe, so $\mu = 0.55$ which would be acceptable for a demonstration hybrid reactor but not for a power plant.
- With $R_p = 1.3$ m, $Q = 1.114$, Service_life of the Be wall = 23.5 years, SHP = 0.096 MW/m², $P_{e_gross} = 1897$ MW and $P_{e_net} = 1638$ MW, so $\mu = 0.86$. The net power is high, but the gain Q is too high (>1) and the service life of the Be wall not very long.

About $P_{e_gross_W}$ and $P_{e_net_W}$: $P_{e_gross_W}$ is always positive, while $P_{e_net_W}$ 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.

Note that $P_{e_net_W}$ is precisely calculated once the fission part has been calculated (§5) and, in particular, the fission thermal power $P_{th_fission_W}$. Then, in the “D_T_PWR_hybrid_reactor” program (§1.4), from the whole results of $P_{e_net_W}$, the average electrical net power over the whole time of working of the reactor ($P_{e_average_MW}$) will be calculated.

4.2. First Beryllium Wall

The wall that separates the fusion reactor from 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.

Moreover, for about the 14.06 MeV neutrons, the Beryllium is a neutrons multiplier, according to the $n + {}^9\text{Be} \rightarrow 2n + 2 {}^4\text{He} - 1.57 \text{ MeV}$.

According to [11], a thickness (Beryllium_thickness_cm) of 12 cm gives a leakage neutron multiplication $\text{Ptr_Be} = 1.7$. This factor (Ptr_Be) is the apparent multiplication factor, *i.e.* the neutrons multiplied minus the absorptions.

So the rate of neutrons at the exit from the Beryllium wall is equal to:

$$\text{Qns} \times \text{Ptr_Be} = \text{Qns} \times 1.7.$$

According to [12], the minimum elastic limit of Beryllium is equal to 170 MPa (1700 bar), for a maximum of 575 MPa. Using the minimum elastic limit and the pressure on the main primary system (Pfr_bar_abs), to avoid the buckling, it must be checked that:

$$\frac{((\text{Rp} \times 100) + \text{Beryllium_thickness_cm}) \times \text{Pfr_bar_abs}}{\text{Beryllium_thickness_cm}} \leq \text{minimum elastic limit_Be (bar)} \quad (\text{Equation 4.8})$$

with Pfr_bar_abs equal to 155 bar abs (see §5.2) and $\text{Rp} = 1.1 \text{ m}$. After calculation, it is found $1576 \leq 1700$, so it is checked that the expected Beryllium_thickness_cm = 12 cm is mechanically sufficient.

4.3. Thicknesses of the Fuel Blanket and the Reflector

The interior radius of the fission reactor is equal to:

$$\text{Ri_fr_cm} = \text{Rp_cm} + \text{Beryllium_thickness_cm} \quad (\text{Equation 4.9})$$

the Beryllium thickness being equal to 12 cm.

The non-leakage probability (Pnl) for neutrons not to leak the fission reactor is equal to

$$\text{Pnl} = \frac{\text{K_eff}}{\text{K_infinite}} \quad \text{or} \quad \text{K_eff} = \text{Pnl} \times \text{K_infinite} \quad (\text{Equation 4.10})$$

with K_infinite and K_eff the multiplication factors of the fission reactor (see §5.5). K_eff is supposed fixed to a target of 0.96 (see §5.3) and depends on the reactor finite dimensions, whereas K_infinite depends on the fuel (§5.5).

Reversely, the leak probability Pl is equal to

$$\text{Pl} = 1 - \text{Pnl} = \frac{\text{K_infinite} - \text{K_eff}}{\text{K_infinite}} \quad (\text{Equation 4.11})$$

Now from [13] page 155,

$$\text{Pnl} = \frac{1}{1 + M^2 \times \text{Bg}^2} \quad (\text{Equation 4.12})$$

The value of M^2 for a PWR is equal to 56 according to [13] page 409.

The geometric buckling Bg^2 is calculated, for a cylinder of radius R (cm) and height H (cm) by ([13] page 155): $\text{Bg}^2 = \frac{j^2}{R^2} + \frac{\pi^2}{H^2}$ with $j = 2.40483$ and for a

$$\text{rectangular parallelepiped } Bg^2 = \pi^2 \times \left(\frac{1}{a^2} + \frac{1}{b^2} + \frac{1}{c^2} \right).$$

The reactor having a cylindrical crown shape, it is intermediate between both shapes. The most pessimistic one being the rectangular parallelepiped one, it is chosen. This crown being circular around two axis, b^2 and c^2 will not be considered. It remains

$$Bg^2 = \frac{\pi^2}{a^2} \quad (\text{Equation 4.13})$$

with

$$a = \text{THco_cm_at_Tfr} + 8.27 \quad (\text{Equation 4.14})$$

i.e. the sum of the core thickness (THco_cm_at_Tfr) and the thickness reflector saving (8.27 cm according to [14]). Note that the reflector is a layer of Reflector_thickness_cm = 20 cm of water (about the thickness used on PWRs) which goal is to limit the neutrons leak and to slow down neutrons.

Note that from Equations (4.10) to (4.14), it is found that:

$$\text{THco_cm_at_Tfr} = \left(23.51 \times \sqrt{\frac{1-Pl}{Pl}} \right) - 8.27 \quad (\text{Equation 4.15})$$

From now on, we will suppose that $K_{\text{infinite minimum}} = 1.03$ (at full power, which corresponds to about 1.07 at 20°C), which is a value which can be reached, for any fuel, after a certain time. In this case, to reach $K_{\text{eff}} = 0.96$, $Pl = 0.0680$ (from Equation (4.11)), which gives THco_cm_at_Tfr = 78.8 cm (from Equation (4.15)). So, it will be considered that THco_cm_at_Tfr = 79 cm for a leak probability $Pl = 0.068$.

Note: in [14], the calculation is based on the “punctual kinetics” method applied to a cylinder, which gives results coherent with the reality. Here, as shown above, this method is applied to a torus using a rectangular parallelepiped infinite in two directions. The uncertainty of the result is clearly superior to the one which would be obtained for a cylinder. Now, only a code could give accurate results considering spatial flux distribution, local power peaking, anisotropic scattering, taking into account the complex core geometry imposed by the Stellarator complex shape. For an example of code application, see [15]. The core and the reflector thicknesses, for the leak probability $Pl = 0.068$, would be accurately calculated by this code.

The exterior radius of the core is equal to:

$$\text{Re_co_cm_at_Tfr} = \text{Ri_fr_cm} + \text{Thco_cm_at_Tfr} \quad (\text{Equation 4.16})$$

The volume of the core Vco_m3_at_Tfr, *i.e.* at the average primary temperature Tfr_C (313°C in §5.2) is equal to:

$$\text{Vco_m3_at_Tfr} = \frac{2 \times \pi \times r \times \pi}{1E4} \times \left[(\text{Re_co_cm_at_Tfr}^2) - (\text{Ri_fr_cm}^2) \right] \quad (\text{Equation 4.17})$$

The volume of the core Vco_m3_at_20_C can be estimated to:

$$\text{Vco_m3_at_20_C} = \text{Vco_m3_at_Tfr_C} \times 0.9838 \quad (\text{Equation 4.18})$$

Note that it can be checked that, according to [5] §4.7, with such core volume, the density of power is widely inferior to the one of the P4/P'4 reactor (*i.e.* 99.64 MW/m³ at 20°C). In fact, the power density of this reactor is rather close to 10 MW/m³ and similar to the Candu power density (12 MW/m³ according to [13] page 409).

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 not be straight, but will rather have a more complex shape.

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.

4.4. Tritium Breeding Blanket at the Exterior of the Fission Reactor

Contrary to true fusion reactors, in this model the Tritium is not built thanks to the fusion neutrons but thanks to the leaky fission neutrons. For this, a Li-6 blanket is used to breed Tritium. This blanket is located at the exterior of the fission reactor between the reflector and the 316LN exterior wall (see the sectional view in Figure 2). This Lithium blanket will be separated from the water due to an exothermic reaction between both, for example using a fine 316LN ring.

The fission neutrons crossing the reflector are slowed down by this one to a thermal state. In the Lithium blanket, these thermal neutrons will generate Tritium, according to the fission reaction between a thermal neutron and the Lithium 6 [16]: $n + {}^6\text{Li} \rightarrow {}^4\text{He} + \text{T} + 4.78 \text{ MeV}$ (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 $n \rightarrow 2n$ reactions with Beryllium or Lead, because the neutrons are, for the most part, thermal. So only Lithium 6 is necessary.

Let's suppose $Pl = 0.0680$ (§4.3). The average rate of neutrons in the core is equal to: $Q_{ns} \times \frac{Ptr_Be}{1 - K_{eff}}$ (§5.4), so the rate of neutrons Q_{nl} reaching the Li6

blanket is equal to $Q_{nl} = Q_{ns} \times \frac{Ptr_Be}{1 - K_{eff}} \times Pl$.

Note: in fact, the neutrons spectrum in energy at the reflector entrance is, in fact, not only thermal, but continuous from thermal to fast neutrons (about 2 MeV). In the reflector, about 80% of the neutrons are reflected. For about the rest, a part is absorbed before crossing the reflector (*a priori* neglected) and the other part, which crosses the reflector, is mainly thermalized.

A neutronic program would be necessary to determine precisely the true rate of leaky neutrons (Q_{nl}), probably inferior to the one calculated above because a part of the neutrons is absorbed. Moreover, the spectrum energy of these leaky neutrons, at the exit of the reflector, might be determined for a more accurate Li6 thickness determination (done at the end of this paragraph).

The tritium breeding blanket produces Tritium with an efficiency called TBR (Tritium Breeding Ratio) limited to 1 and supposed, here, equal to 0.95. The goal is that the Tritium lost on D-T reactions (Q_{ns}) be inferior to the Tritium produced: $Q_{nl} \times TBR \geq Q_{ns}$. However, a margin coefficient equal to 1.05 (1.05 is the DEMO target [17], required to attain self-sufficiency) must be taken.

So the condition to be self-sufficient in Tritium is:

$$Q_{ns} \times \frac{Ptr_Be}{1 - K_{eff}} \times Pl \times TBR \geq Q_{ns} \times 1.05 \quad \text{or} \quad \frac{Ptr_Be}{1 - K_{eff}} \times Pl \times TBR \geq 1.05$$

$$\text{In our example, for } K_{eff} = 0.96: \quad \frac{Ptr_Be}{1 - K_{eff}} \times Pl \times TBR \approx 2.75 \geq 1.05.$$

At best, it would be produced about $2.75/1.05 = 2.61$ times the necessary tritium. So this reactor could also be considered as a small Tritium generator. However, this result is probably very optimist due to the absorptions in the reflector.

About Tritium, the “D_T_PWR_hybrid_reactor” program (§1.4) will display:

$$\text{The Global_Consumed_Tritium (kg)} = \int_0^t Q_{ns} \times 5.0082E-27 \times 1.05 \times dt \quad (\text{Equation 4.19})$$

$$\text{The Global_Produced_TriTium (kg)} = \int_0^t Q_{ns} \times 5.0082E-27 \times \frac{Ptr_Be}{1 - K_{eff}} \times Pl \times TBR \times dt \quad (\text{Equation 4.20})$$

Both variables are calculated over the time of working of the reactor (t), with $5.0082E-27$ being the weight in kg of an atom of tritium. The goal after the whole time of working, is that the produced Tritium be (widely) superior to the consumed Tritium. With $Ptr_Be = 1.7$ (§4.2), $Pl = 0.0680$ (§4.3) and $TBR = 0.95$. K_{eff} is a calculated parameter (with at best $K_{eff} = 0.96$).

The ratio

$$\text{Global_Gain_Tritium} = \frac{\text{Global_Produced_TriTium}}{\text{Global_Consumed_Tritium}} \quad (\text{Equation 4.21})$$

is also displayed.

Thickness of the tritium breeding blanket

As said above, the exact fission neutrons spectrum is not known. So it will be supposed an average $n + {}^6\text{Li} \rightarrow {}^4\text{He} + \text{T}$ reaction cross-section of 2 barn, which is probably a penalizing hypothesis because the spectrum is probably almost completely thermal. With this hypothesis, a 40 cm thickness of Li6 will permit a probability of the $n + {}^6\text{Li} \rightarrow {}^4\text{He} + \text{T}$ reaction equal to 0.975 (ideal TBR), neglecting absorption. So it will be considered a $Li6_thickness_cm = 40$ cm of Li6 for a practical TBR of 0.95, as a first hypothesis.

About the position of the Tritium breeding blanket

It can be of interest to know that in certain configurations (for example the one

presented in [15]), the blanket is positioned between the fission reactor and the reflector, this blanket being crossed by fast and thermal neutrons.

4.5. Thickness of the 316LN Reactor Vessel Body (i.e. the Exterior Wall)

The 316LN non-magnetic stainless steel external wall must withstand a pressure $P_{\text{gauge}} = 154 \text{ bar gauge}$ ($=P_{\text{fr_bar_abs}} - 1$) and must protect the magnetic coils against heat and fission neutrons. The interior radius of this wall is equal to

$$Ri_{\text{ew_cm}} = Re_{\text{co_cm_at_Tfr}} + \text{Reflector_thickness_cm} + \text{Li6_thickness_cm} \quad (\text{Equation 4.22})$$

In our example, $Ri_{\text{ew_cm}} = 261 \text{ cm}$.

For a ring, the required thickness is equal to:

$$Th_{\text{ew_cm}} = \frac{P_{\text{gauge}} \times Ri_{\text{ew_cm}} \times 2}{\text{elastic_limit_316LN_bar} - P_{\text{gauge}}} \quad (\text{Equation 4.23})$$

The elastic limit is equal to 185 MPa, so 1850 bar [18]. So the thickness $Th_{\text{ew_cm}}$, in our example, is equal to 48 cm.

The exterior radius of the exterior wall in 316LN is equal to:

$$Re_{\text{ew_cm}} = Ri_{\text{ew_cm}} + Th_{\text{ew_cm}} \quad (\text{Equation 4.24})$$

So $Re_{\text{ew_cm}} = 309 \text{ cm}$ in our example.

Note that this exterior radius is the interior radius of the system supporting the magnetic coils cooled by the cryogenic fluid (see Figure 2).

5. Physical Model of the PWR Fission Reactor

5.1. Introduction

At this level, the fusion reactor and the main parameters of the hybrid reactor, including the sizing, have been calculated (§3 + §4).

Now, in the “D_T_PWR_hybrid_reactor” program (§1.4), the user will have to propose a mixture of U235, U238 and Th232 in terms of atomic concentrations, the total being equal to 1 (100%). This data will permit to calculate the fission reactor.

About the “D T PWR hybrid reactor” program (§1.4)

Once the preliminary calculations of the fusion and the fission reactors are done, the calculation will be incremental by steps of one hour. The reactivity, the thermal fission power generated, the net electric power, the average net electrical power, and the masses of Tritium produced and consumed will be calculated. Results will be displayed per year. The final fuel composition is also given.

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 the plant, the maximum being 255 years.

5.2. Temperature and Pressure Conditions of the Main Fission Primary System

The primary pressure $P_{\text{fr_bar_abs}}$ is equal to 155 bar abs, which is the standard pressure for PWR reactors. The average primary temperature $T_{\text{fr_C}}$ is equal to 313°C, as for an EPR.

5.3. Preliminary Calculations Relative to the Fission Reactor

The model of PWR used for calculations relative to reactivity, thermal power, geometric values, etc. is the P4/P'4 1300 MWe reactor 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 [14]. These calculations based on Uranium only have been extended in [5] to Plutonium and Thorium. They will not be (re)presented in the rest of this document, but will be present in the source code of the “D_T_PWR_hybrid_reactor” program, if relevant. So calculations already done on [5] and [14] will not be described, whereas other calculations will be described.

For preliminary calculations about 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 , the number of fuel (U_Th_02) atoms per cm^3 of this fission reactor (N_fuel) and for the molar moderating ratio (MR), etc, refer to [5] §4.5 and the “D_T_PWR_hybrid_reactor” program.

About the maximum reactivity allowed ($K_{\text{eff}} = 0.96$) and about safety

In a hybrid reactor, the reactor is subcritical. Here, it is supposed a K_{eff} equal to 0.96. It was supposed 0.9 in [5], but 0.96 permits a better neutron multiplication, still, a priori, with good safety. Due to the subcritical working, there is no need of the full-length rod control system. $K_{\text{eff}} = 0.96$ corresponds to a negative reactivity of 4167 pht (“pht” for “Per hundred thousand”, *i.e.* in 1/100,000) with $K_{\text{eff}} = \frac{1}{1 - \text{reactivity in pht}}$. 4167 pht is superior to the required minimum

negative reactivity margin, estimated to about 3000 pht. Consequently, in case of accident, the fission reactor would remain subcritical. At this condition, a quick stop of the fusion reactor would immediately stop the fission reactor. It would remain to remove the residual heat, as for a PWR.

Note that, in normal working, for a real $K_{\text{eff}} > 0.96$, the reactor water must be borated, so as to remain at $K_{\text{eff}} = 0.96$. Conversely, for a real $K_{\text{eff}} \leq 0.96$, the water must be the least borated possible. This maximum of 0.96 will be used to calculate the fission reactor.

5.4. Thermal Power Produced by Fission

The rate of neutrons at the exit of the Beryllium wall ($Q_{\text{ns}} \times \text{Ptr}_{\text{Be}}$) is multiplied, inside the fission reactor, by a factor equal to $\frac{1}{1 - K_{\text{eff}}}$, cf. [13] pages 86 and 410.

Note that $\text{Ptr}_{\text{Be}} = 1.7$ for 12 cm of Beryllium (§4.2).

The maximum effective multiplication factor $K_{\text{eff}} = 0.96$ is used to size the fission reactor.

So the average rate of neutrons (of any energy) in the fission reactor is equal to:

$$Q_{\text{n_fr}} = \frac{Q_{\text{ns}} \times \text{Ptr}_{\text{Be}}}{1 - K_{\text{eff}}} \quad (\text{Equation 5.1})$$

The probability for a neutron to generate a fission is equal to $w = \frac{K_{eff}}{\text{Mean_Nu}}$

([13] page 118).

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

So the rate of fissions per second is equal to

$$Q_{f_fr} = \frac{Q_{n_fr} \times K_{eff}}{\text{Mean_Nu}} \quad (\text{Equation 5.2})$$

Note: Q_{f_fr} can be written $Q_{f_fr} = Q_{ns} \times K_p$ with

$$K_p = \frac{\text{Ptr_Be} \times K_{eff}}{(1 - K_{eff}) \times \text{Mean_Nu}}.$$

With Mean_Nu around 2.9, $K_p \approx 14.1$.

The maximum thermal power generated by fissions in the reactor (Pth_fission_MW) is equal to:

$$\text{Pth_fission_MW} = Q_{f_fr} \times \text{Mean_energy_release_per_fission_MeV} \times 1.60219 \times 10^{-19} \quad (\text{Equation 5.3})$$

with Mean_energy_release_per_fission_MeV which depends on the fuel (but around 200 MeV).

5.5. Determination of the Multiplication Factors: Infinite ($K_{infinite}$) and Effective (K_{eff})

For information about $K_{infinite}$, K_{eff} and the non-leakage probability (Pnl), see [19] and [13] page 115 and 155. The infinite multiplication factor $K_{infinite}$ is calculated according to [5] §4.9 and [14], taking into account the current fuel composition. The calculation of [14] 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.

The reactor having finite dimensions, it is necessary to estimate the non-leakage probability of a neutron emitted in the fission core (Pnl). In the same way as in

$$\text{§4.3, } Pnl = \frac{1}{1 + M^2 \times Bg^2} \quad (\text{Equation 4.12}), \text{ with } M^2 = 56 \text{ for a PWR ([13] page}$$

$$409) \text{ and } Bg^2 = \frac{\pi^2}{a^2} \quad (\text{Equation 4.13}) \text{ with } a = \text{THco_cm_at_Tfr} + 8.27 \quad (\text{Equation 4.14}).$$

The initial K_{eff} factor [14] is calculated from $K_{infinite}$ 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 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. The correction and its integration in K_{eff} is described in [5] §4.9.

In [5] §4.9, it is also taken into account the negative reactivity introduced by the Xenon-Samarium and by the fission products.

5.6. Determination of the Thermal Neutrons Flux

To simplify, the thermal neutrons flux (in neutrons/(cm² × s)) is the sole considered. The epithermal and the fast neutrons flux are not considered.

Neutrons flux due to the sole fusion neutrons

The method is the same as the one described in [5] §4.10, except that the Source area (considered at the exterior of the Beryllium wall) is equal to:

$$\text{Source_Area_cm}^2 = 2 \times \text{Pi} \times \text{Ri_fr_cm} \times 2 \times \text{Pi} \times r \text{ (cm)}$$

Moreover, Qns must replace Qns_2L to calculate the rate of neutrons through this area (S_n_per_cm2):

$$S_n_per_cm2_s = \frac{Qns \times \text{Ptr_Be}}{\text{Source_Area_cm}^2}$$

According to equation 4.38 of [5] §4.10, we have:

$$\begin{aligned} &\text{Thermal_fusion_neutron_flux} \\ &= \frac{15 \times S_n_per_cm2_s}{\text{THco_cm_at_Tfr}} \times (1 - \exp(-0.408 \times \text{THco_cm_at_Tfr})) \end{aligned}$$

Here, with THco_cm_at_Tfr = 79 cm, the Thermal_fusion_neutron_flux is almost nil and is considered as equal to 0. So this reactor cannot work with pure Thorium as fuel, due to an initial zero reactivity. Thorium must be initially mixed with a “primer” (see §6.4).

In [5] §4.10, the normal average thermal neutron flux due to fusion + fission neutrons is also determined. It is applicable to this reactor.

5.7. Refueling and Fuel Burn-Up—Comparison with PWR and Candu Reactors

The fuel reprocessing and refueling is supposed to be done as detailed in [20]. So all the fission products and all the minor actinides (Americium, Curium and Neptunium) are removed from the spent fuel and finally stored, the goal being to recover a fuel having its maximum (*i.e.* initial) reactivity.

Note that the Protactinium is ignored because it quickly disappears.

Note that there would not have much disadvantage in not removing Neptunium, because Neptunium is transformed into Plutonium. Indeed, keeping the Neptunium on the natural Uranium leads to a loss, in terms of electrical net power after 255 years, equal to about 1/1000. So Neptunium could be kept in the fuel and not be considered as a “minor actinide waste”. Conversely, Americium and Curium must really be removed.

To simplify the calculation, it is assumed that a regular refueling of the whole fuel (and not by 1/3 or 1/4) is done each time the fuel burn-up (calculated from the previous refueling) passes:

- For natural and depleted Uranium, and spent fuel: 1400 MWd/t, so once a year. It is obvious that 1400 MWd/t is weak, but it is justified by the weak margin

on the initial reactivity of the fuel used. For spent fuel, it could be more than 1400 MWd/t at the beginning due to its high initial reactivity, but after about 20 years the spent fuel K_{eff} tends to evolve as the natural Uranium K_{eff} .

- For natural Thorium: 9000 MWd/t, so once every 6 years. It is superior to the threshold for natural Uranium (1400 MWd/t), because once started and stabilized the reactivity is slightly higher.

For details, see [5] §4.11, taking into account that the refueling was proposed when the fuel burn-up passed 10,000 MWd/t and neither 1400 MWd/t nor 9000 MWd/t.

Comparison with PWR and Candu reactors

It is calculated and displayed at each reloading the mass of fuel consumed from the beginning of operation, which related with the total thermal energy produced, gives the average burn-up of the consumed fuel in GWd/t. A calculation of this average burn-up gives, after 255 years of operation, the reactor being considered stabilized: 704.3 GWd/t for the natural Uranium, versus 7.5 GWd/t for PWR and Candu reactors ([5] §1.1). Hence, this reactor is $704.3/7.5 = 94$ times more efficient than a standard reactor (PWR or Candu). For natural Thorium, after 253 years, the average burn-up is equal to 811.2 GWd/t. So a Th232 reactor is $811.2/7.5 = 108$ times more efficient than a standard reactor. It is more efficient than a natural Uranium reactor and produces less minor actinides (§6.4).

5.8. Fuel Evolution

In **Figure 5** below, the simplified chain of the managed fuel is shown.

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_T_PWR_hybrid_reactor” program and in [5] §4.12.

Note that Am241 and Cm244 are the final minor actinides of this simplified evolution chain. In fact, there are other transmutations of both materials into other minor actinides, but with a very small influence relative to the target of this document.

5.9. Description of the Way to Calculate the Hybrid Reactor

The global way to calculate the hybrid reactor is given in the source code of the “D_T_PWR_hybrid_reactor” program and based on this document and in [5] §4.13.

However, contrary to [5], the sizing of the reactor has already been calculated according to §4, so it is fixed.

Note that each year, it is also given the following pieces of information:

- The fusion to fission power amplification gain $\frac{P_{\text{th_fission_W}}}{P_{\text{th_fusion_W}}}$ (≈ 78 for the default configuration with natural Uranium).
- The global power amplification factor equal to $\frac{P_{\text{sg_W}}}{P_{\text{input_max}}}$ (≈ 124 for the

default configuration with natural Uranium).

- The masses of Tritium consumed and produced (§4.4), plus the global gain of Tritium (respectively after 100 years, for the default configuration with natural Uranium: 115.5 kg, 289.4 kg and 2.51).

After stabilization, in a year, the difference between the Tritium produced and the Tritium consumed is equal to 1.86 kg.

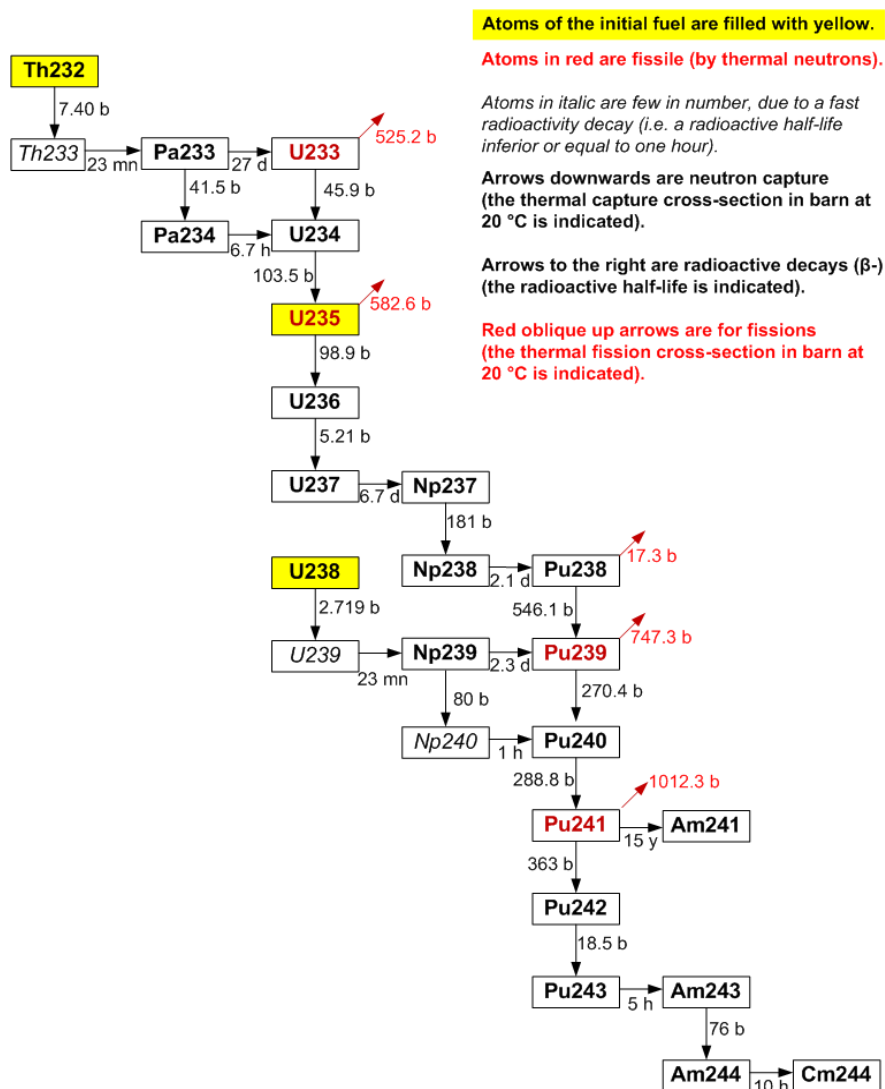


Figure 5. Simplified evolution chain of a Thorium (Th232) + Uranium (U238 + U235) fuel.

It is recalled that the average electrical net power in MW ($Pe_{average_MW}$) over the global working of the reactor is given each year (1068 MWe after 100 years for the default configuration with natural Uranium).

At the end of the calculation (so after 100 years of working time by default), the final fuel composition (in atomic concentrations) is displayed. Note that if a refueling has been done in the last year, all the minor actinides will appear at 0. As the source code is available, the user can modify the information displayed.

6. Results and Discussion

In this chapter, the main results obtained with natural Uranium, depleted Uranium, high-level radioactive waste (stored spent fuel) and an initial mixture of 1.8% military Uranium mixed into 98.2% of natural Thorium will be shown. The main results are:

- The K_{eff} multiplication factor before limitation, which is more interesting than the limited K_{eff} . It is reminded that K_{eff} is limited to 0.96 with borated water (§5.3).
- The net electrical power in MWe.

6.1. Working with Natural Uranium over 100 Years

In **Figure 6** below, the results obtained with natural Uranium as fuel are displayed, from 0 to 100 years.

The initial rise of both parameters (K_{eff} and the net electrical power) is due to the transmutation of U238 into Pu239, before stabilization.

After stabilization, the net electrical power reaches 1108 MWe. The net electrical power is constant because K_{eff} is limited to 0.96.

Note that even without borated water, the reactor would remain subcritical ($K_{\text{eff}} < 1$).

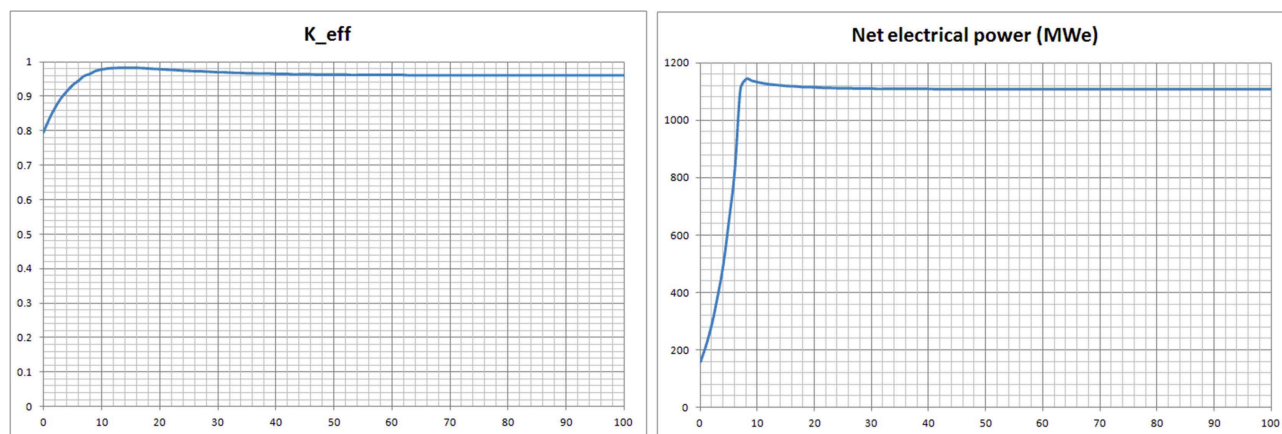


Figure 6. Main results for natural Uranium with respect to time in years.

About the fissile materials proliferation

The final composition of Plutonium shows:

Pu238 = 0.000%, Pu239 = 0.718%, Pu240 = 0.205%, Pu241 = 0.308%, Pu242 = 0.626% and Pu243 = 0.000%

The fissile materials (Pu239 and Pu241) are present at 1.03%, whereas non-fissile materials are present at 0.83%. If the plutonium were separated, the percentage of Pu fissile materials would be too much important compared to the Pu non-fissile materials, which might be avoided, for obvious reasons.

So for this type of reactor, and as no Uranium enrichment is required, no separation between Uranium and Plutonium must be made. In that case, the Pluto-

nium would be mixed with 98.105% of U238 (non-fissile) and no direct risk with this fuel would exist.

6.2. Working with Depleted Uranium over 100 Years

In **Figure 7** below, the results obtained with depleted Uranium as fuel (0.3% of U235) are displayed, from 0 to 100 years.

The initial rise of both parameters (K_{eff} and the net electrical power) is slower than in the case of natural Uranium due to the low initial reactivity being equal to 0.490 versus 0.795 for the natural Uranium. As with natural Uranium, the net electrical power reaches a stabilized value of 1108 MWe.

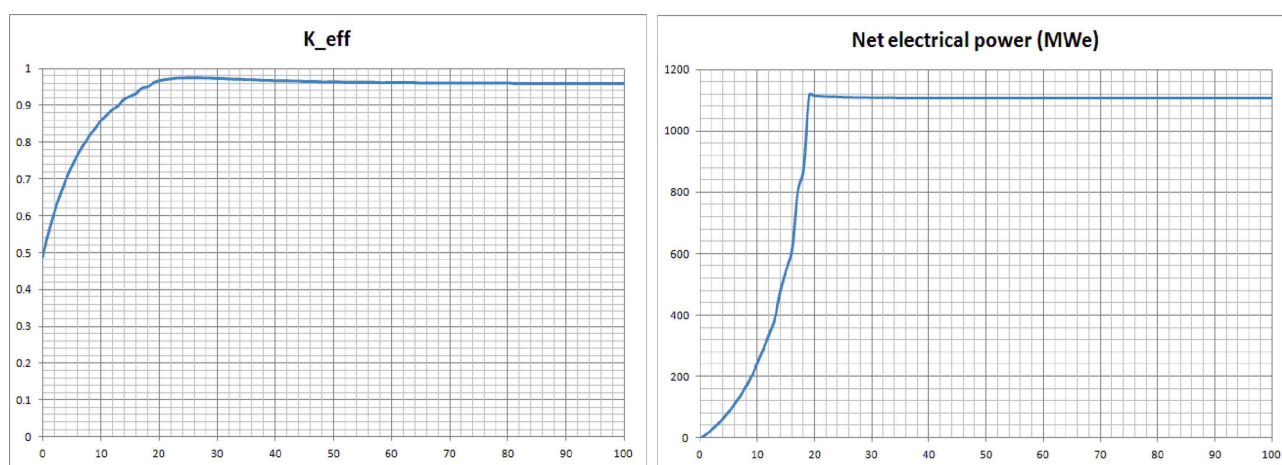


Figure 7. Main results for depleted Uranium with respect to time in years.

About the fissile materials proliferation: the results are about the same as with natural Uranium (§6.1), so the recommendation not to separate Uranium from Plutonium is the same.

6.3. Working with High-Level Radioactive Waste (Stored Spent Fuel) over 100 Years

The composition of the stored spent fuel is taken from [20] page 74 (stored with a fuel burn-up of 47.5 GWd/t). This stored spent fuel is supposed to have been reprocessed. So it has been cleansed of all its minor actinides (Americium, Curium and Neptunium) and of all its fission products (see §5.7). It takes advantage of its abundant fissile materials in it: 0.779% of U235, 0.642% of Pu239 and 0.158 % of Pu241, so a total of 1.579%. An initial relatively high reactivity can be expected.

In **Figure 8** below, the results obtained with the spent fuel are displayed, from 0 to 100 years.

It can be noted that due to the strong initial reactivity, there is no initial rise of K_{eff} . So the net electrical power is immediately high. But contrary to the previous cases, without borated water, the reactor can be critical, with a maximum of $K_{eff} = 1.013$. So probably, a mixture of depleted Uranium (low reactivity) with spent fuel (high reactivity) would give a good fuel. The average net electrical power is

about 1100 MWe, and it has not stabilized at 100 years. But after 255 years, it has stabilized at 1108 MWe, as for natural Uranium (its evolution ends up being almost the same as the natural Uranium one).

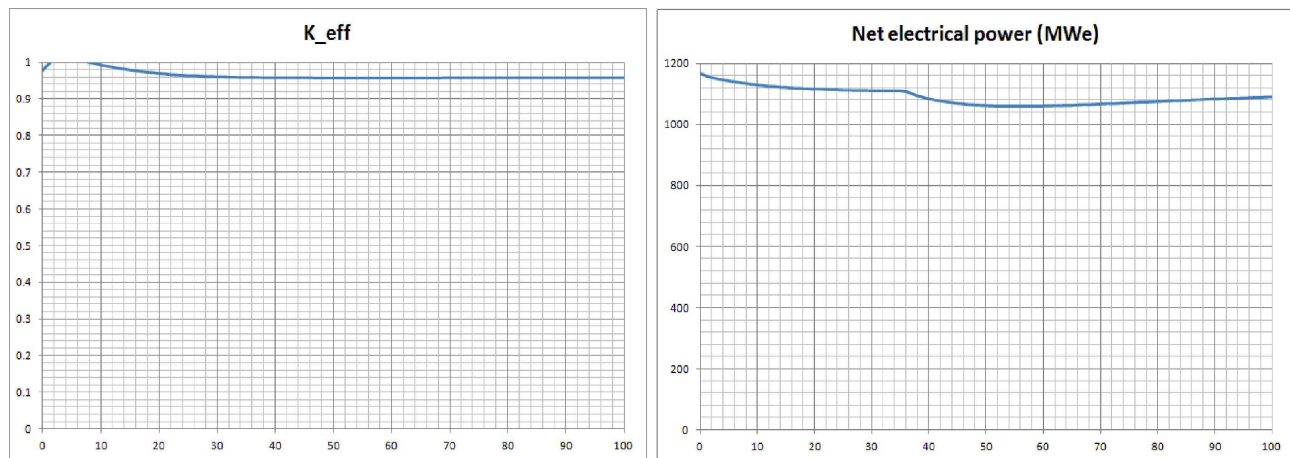


Figure 8. Main results for high-level radioactive waste (stored spent fuel) with respect to time in years.

About the fissile materials proliferation: after stabilization (at 255 years), the results are about the same as with natural Uranium (\$6.1), so the recommendation not to separate Uranium from Plutonium is the same.

6.4. Working with a Mixture of 1.8% of Military Uranium into 98.2% of Natural Thorium over 100 Years

The natural Thorium has no initial reactivity, so a “primer” is necessary to start the reactor. A possible “primer” could be 50% of natural Uranium into 50% of natural Thorium, but the time to reach the nominal net electrical power would be superior to 100 years. A much better “primer” is 1.8% of military Uranium (at 90% U235) mixed into 98.2% of natural Thorium, which is tested here by simply clicking on the “98.2%” button of the D_T_PWR_hybrid_reactor program (\$1.4). The “primer” is introduced once at the start-up of the reactor. Afterwards, the operations of refueling are done with natural Thorium only, so without “primer”.

This explains the provisional losses of K_{eff} and net electrical power, around 14 years.

In **Figure 9** below, the results obtained are displayed, from 0 to 100 years. The small periodic discontinuities on K_{eff} are due to the refueling operations. It can be noted that thanks to the military uranium, the initial reactivity is strong, so there is no initial rise of K_{eff} and the net electrical power is immediately high. With 1.8% of military Uranium, even without borated water, the reactor would remain subcritical ($K_{eff} \text{ max} = 0.998 < 1$). After stabilization, the net electrical power reaches 1222 MWe. The net electrical power is constant because K_{eff} is limited to 0.96.

Note that, once stabilized, the K_{eff} before limitation (first parameter) of natural Thorium is slightly better (≈ 0.98) than the K_{eff} of natural Uranium (≈ 0.96).

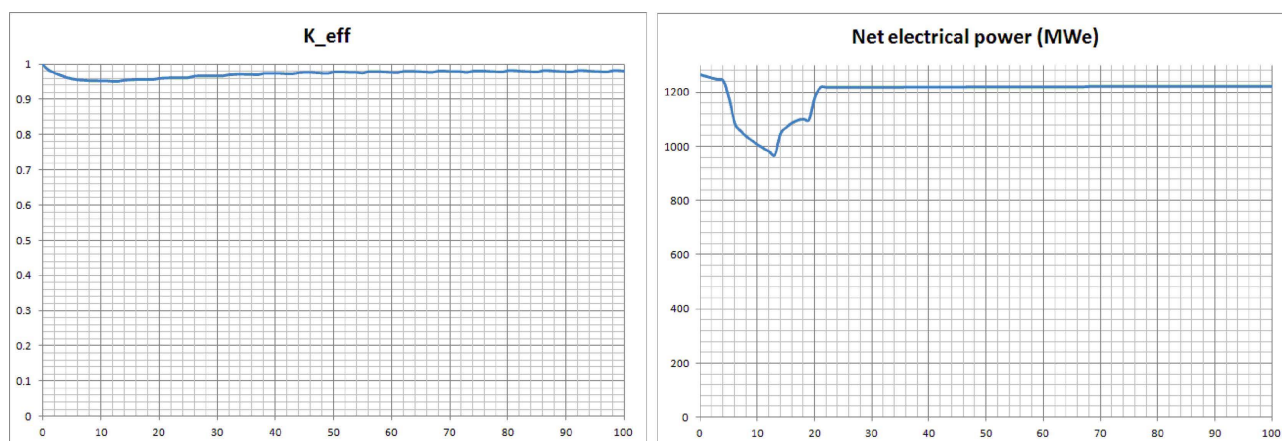


Figure 9. Main results for a mixture of 1.8% of military Uranium and 98.2% of natural Thorium, with respect to time in years.

About the fissile materials proliferation

The final composition shows for Plutonium and Uranium:

Pu238 = 0.013%, Pu239 = 0.006%, Pu240 = 0.002%, Pu241 = 0.003%, Pu242 = 0.005% and Pu243 = 0.000%, U233 = 1.481%, U234 = 0.925%, U235 = 0.243%, U236 = 0.219%, U237 = 0.000%, U238 = 0.138% and U239 = 0.000%.

The content of Plutonium is here very weak, but the content in fissile Uranium (U233 and U235) represents 1.724%. So the recommendation not to separate Uranium, Thorium and Plutonium remains applicable.

The production of minor actinides (Np, Cm and Am) is low. For example, not removing the minor actinides (Np, Cm and Am) for 255 years will reduce the average net power calculated over these 255 years by only 6.6% for natural Thorium versus 90.3% for natural Uranium.

6.5. Points to Deepen

Below is a non-exhaustive list of points to deepen:

- The way to calculate these reactors (fusion and fission) is very simplified because, for both, a punctual model is taken, which means that the uncertainty on the performance of this reactor is important. More accurate calculations would be necessary, especially for the non-leakage probability (Pnl), the rate of leaky neutrons reaching the Li6 blanket (Qnl), the neutron energy spectrum in the reflector and in the Li6 blanket, and the absorptions in the reflector. Moreover, the decay heat, mainly due to fission products but also due to the activation of materials such as steel, which has been ignored in this document, might be studied. It has also been assumed a perfect reprocessing process (without losses). However, the reprocessing process is not perfect, and this might also be taken into account.
- 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.
- The safety of this type of reactor has been quickly approached in §5.3 and in

[5] §5.9. Now, a complete study would be necessary, as this reactor is quite different from a classical PWR:

- o The fission reactor is always subcritical ($K_{\text{eff}} = 0.96$) and, probably, will remain subcritical after any accident. So the fission reactor will always be controlled by the fusion reactor.
- o There is no emergency shutdown rod system. However, a quick stop of the fusion reactor will stop the fission reactor, provided that the fission reactor remains subcritical, as said above. This condition must be checked. Now, as for a PWR, large quantities of boron could quickly be injected.
- o The accidental loss of the fusion plasma heating will lead to a quick drop in plasma temperature, leading to a quick loss of the fusion neutron production and to the almost immediate stopping of the fission reactor.
- o The loss of coolant will be managed as on PWRs, *i.e.* thanks to a high flow rate of borated water provided by the “Safety Injection System”.
- o The boron concentration will be initially low because, in any case, the maximum reactivity K_{eff} will not be much above 0.96 (§5.3). Consequently, in case of a boron dilution, the concentration will slowly decrease, so the reactivity will slowly increase, which will permit a secure detection. Moreover, the fission reactor must be designed to remain subcritical ($K_{\text{eff}} < 1$) even in case of a total dilution (see §6.1 to §6.4).
- o The decay-heat removal will be managed as in a PWR with a dedicated system, called “Residual Heat Removal System”.
- o The Tritium handling has been ignored in this document. But due to the radiological effects, a containment strategy must be applied.

Moreover, hypotheses and simplifications have been taken on the model throughout this article. Some of them have a certain tendency to shift upward or downward the performance of this reactor.

First, the ones for which an accurate calculation will shift the performance upward:

- Taking into account the $\text{He4} +$ ions heating rather than ignoring it (§3).
- Taking into account 5 T for the toroidal magnetic field, rather than 3 T (§4).
- Taking into account the D-D fusion reactions (§2.2).
- The reactivity of the fuel can be improved, as said in a note at the end of §4.3.
- Increasing the Beryllium thickness will certainly increase the leakage neutron multiplication Ptr_{Be} (§4.2) and, consequently, the neutron rate and the reactor power.

Secondly, the ones for which an accurate calculation will shift the performance downward:

- Taking into account the absorptions in the reflector will certainly decrease the performance of the Tritium production, by decreasing the effective rate of leaky neutrons. If the production of Tritium was inferior to the consumption of Tritium, a possible solution would be to invert the position of the Tritium blanket and the reflector (see §4.4).
- The geometry taken for this reactor is simplified (see **Figure 2**), but the com-

plex Stellarator geometry is perhaps not compatible with a hybrid reactor (§1.1 and §4.3).

7. Conclusions

This hybrid reactor has been described in §2. Afterwards, it has been modeled from §3 to §5.

The gross electrical power depends on the rate of neutrons from the fusion reactor and consequently, roughly, depends on the fusion reactor volume. Now, in terms of net electrical power, a fusion reactor with a radius of 0.53 m would supply enough neutrons for a 110 MWe (§4.1) demonstration plant, whereas a fusion reactor with a radius of 1.1 m would supply enough neutrons for a 1000 MWe power plant (§4.1).

The heat flux and the neutron flux on the first wall are small (§4.1) compared to a true fusion reactor. Consequently, such a fusion reactor would be much easier to design and build than a true fusion reactor.

The production of Tritium, thanks to leaky thermal fission neutrons, is about 2.5 times the Tritium consumed; so this reactor can be considered as a small Tritium producer (1.86 kg/year, see §5.9). However, this result is dependent on the true rate of absorptions in the reflector, which is neglected in this document (§4.4).

From §6.1 to §6.4, it has been presented the main results about the working of this reactor with different types of fuel, *i.e.* natural Uranium, depleted Uranium, spent fuel and a mixture of 1.8% of military Uranium into 98.2% of 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 high-level radioactive waste (stored spent fuel) is a relatively reactive fuel which would be favorably mixed with depleted Uranium (§6.3).

The net electrical power with the Thorium fuel, at stabilization, is better than the one with the Uranium fuel, even if the Thorium reactor needs a “primer” to start the reactor (§6.4).

This reactor would be, *a priori*, a safe reactor due to its subcriticality (§5.3 and §6.5). Moreover, the absence of U235 enrichment and the uselessness to separate Uranium, Thorium and Plutonium in the reprocessing operation makes the fuel safe with regard to proliferation. It must also be added that the final waste is constituted by fission products and minor actinides (mainly americium and curium), without any fuel (*i.e.* Uranium, Plutonium or Thorium). Note that natural Thorium is the best fuel tested in this paper (§5.7 and §6.4). Moreover, it produces fewer minor actinides than natural Uranium (§6.4).

Note also that Neptunium could be included in the natural Uranium fuel and not be considered as a minor actinide waste, as the disadvantage of including it in the fuel is negligible (§5.7).

The points to deepen in §6.5 complete the analysis of this hybrid reactor.

The presented results are orders of magnitude due to simplified modeling, but

they are sufficient to consider the possibility of a functional D-T hybrid reactor.

Conflicts of Interest

The author declares no conflicts of interest regarding the publication of this paper.

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