Summarized proposal of a Deuterium-Deuterium fusion / PWR fission hybrid reactor

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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.

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1. Introduction and presentation

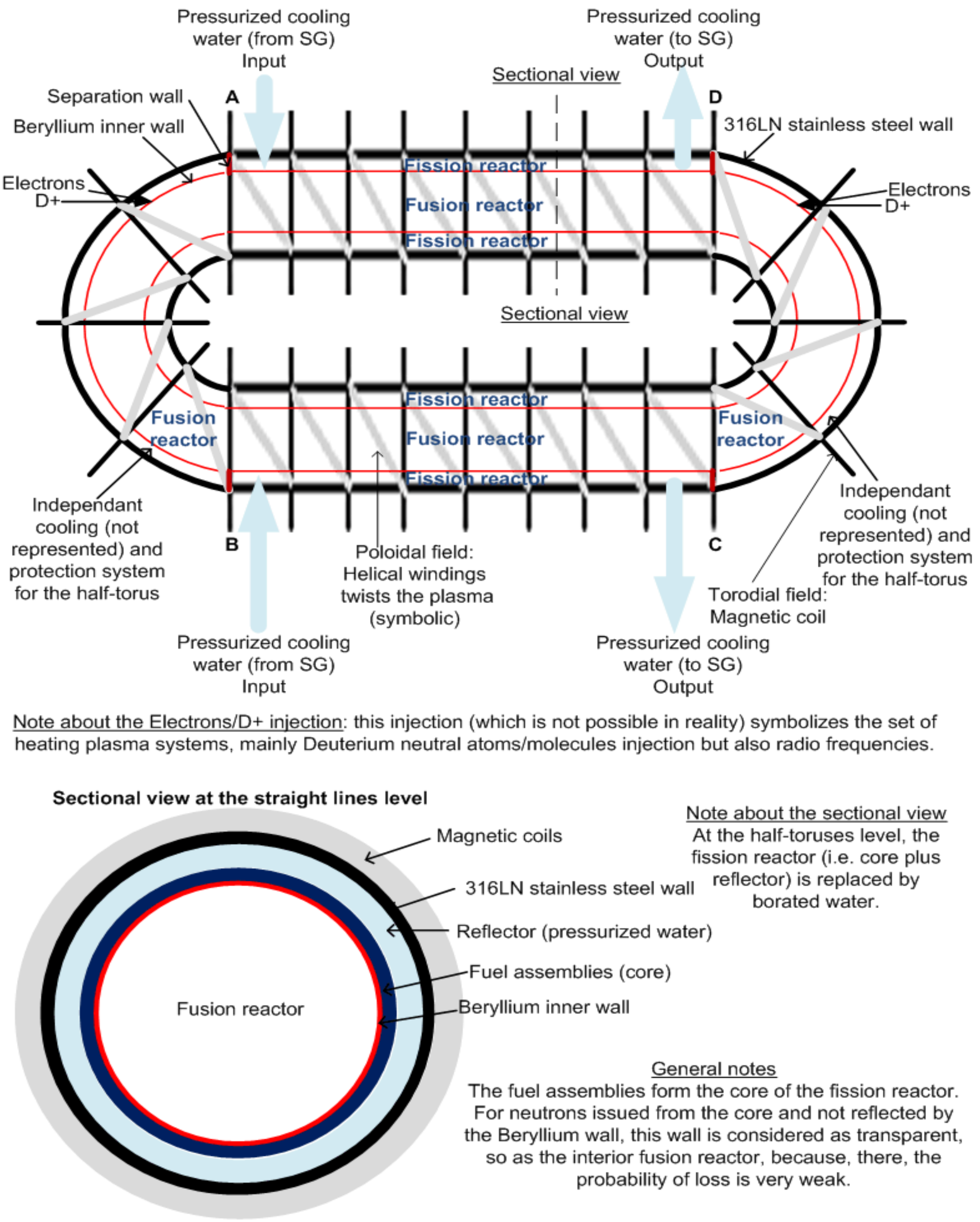
The goal of this presentation is to briefly describe a hybrid power plant formed by (see figure 1):

- A Deuterium-Deuterium (D-D) fusion reactor yet proposed by the author in a previous article. 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.

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. It will be produced nuclear fusions with a mechanical gain (Q), i.e. fusion power / mechanical injection power, depending on the pipe radius (Rp) but inferior to 1 for this hybrid reactor. For example, at Rp=2 m, Q=0.184.

- 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 neutrons 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 before becoming a powerful electricity generator.

This paper is an abstract (6 pages) of the original article (39 pages) available here: http://f6cte.free.fr/Proposal_of_a_Deuterium-Deuterium_fusion_PWR_fission_hybrid_reactor.pdf



Note about the Electrons/D+ injection: this injection (which is not possible in reality) symbolizes the set of heating plasma systems, mainly Deuterium neutral atoms/molecules injection but also radio frequencies.

Sectional view at the straight lines level

Note about the sectional view
At the half-toruses level, the fission reactor (i.e. core plus reflector) is replaced by borated water.

General notes
The fuel assemblies form the core of the fission reactor. For neutrons issued from the core and not reflected by the Beryllium wall, this wall is considered as transparent, so as the interior fusion reactor, because, there, the probability of loss is very weak.

Figure 1. D-D/PWR fusion reactor principle diagram

2. Why this type of fusion reactor

It could be proposed a D-T reactor, 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 works in a reasonable domain, i.e. with a Beta factor equal or inferior to 0.05 which is the present limit for Stellarators. 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, as no Tritium must be supplied and the thermal and neutrons constraints are minimal.

3. 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.

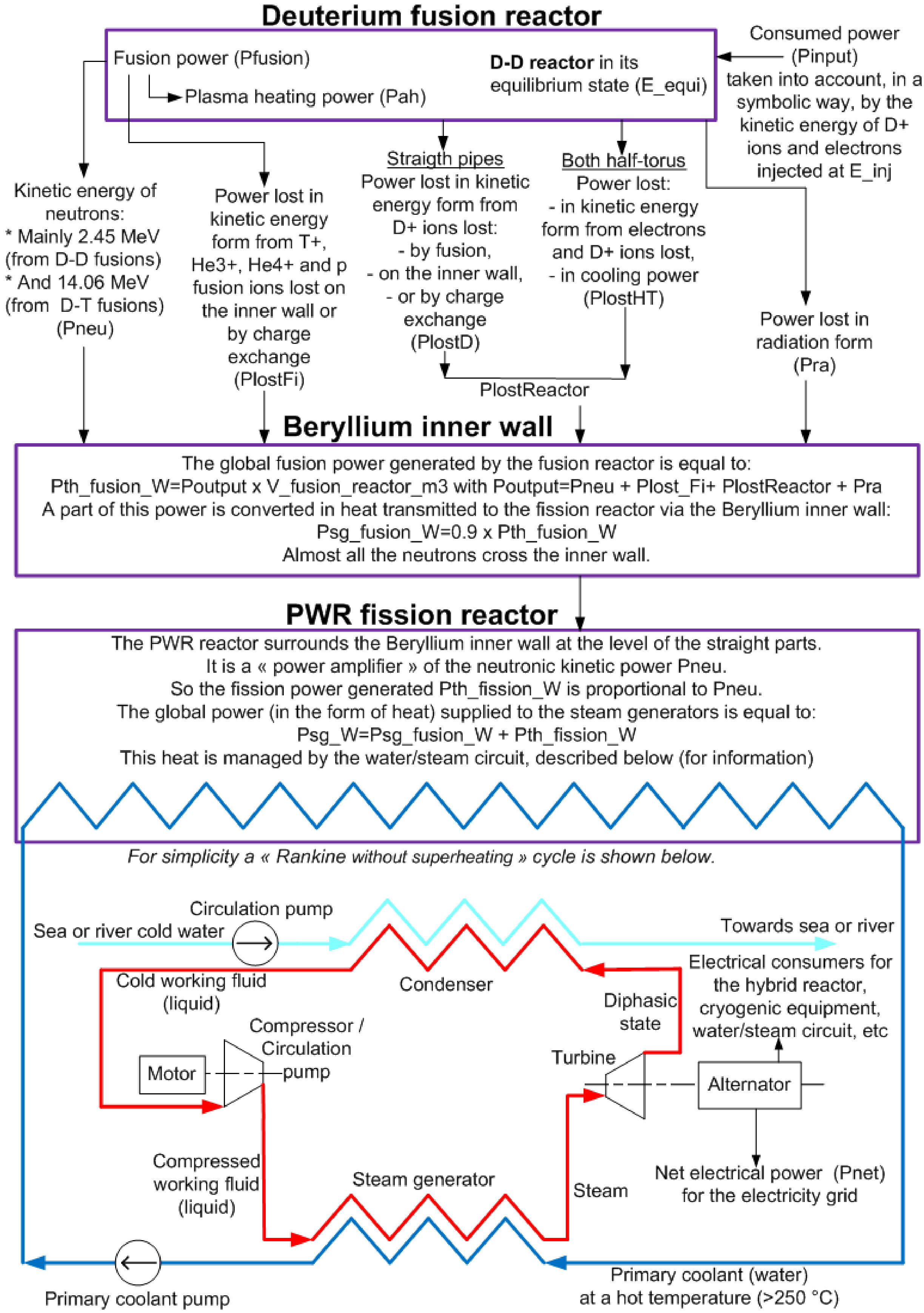
The BWR (Boiling Water Reactor) uses light water as PWR but it is larger than the PWR due to a lower moderation power of the boiling water.

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.

4. Hybrid reactor energy balance

The figure 2 shows, in a general way, how the hybrid reactor energy balance is taken into account, relatively to the power plant.



The global fusion power generated by the fusion reactor is equal to:
Pth_fusion_W=Poutput x V_fusion_reactor_m3 with Poutput=Pneu + Plost_Fi+ PlostReactor + Pra
A part of this power is converted in heat transmitted to the fission reactor via the Beryllium inner wall:
Psg_fusion_W=0.9 x Pth_fusion_W
Almost all the neutrons cross the inner wall.

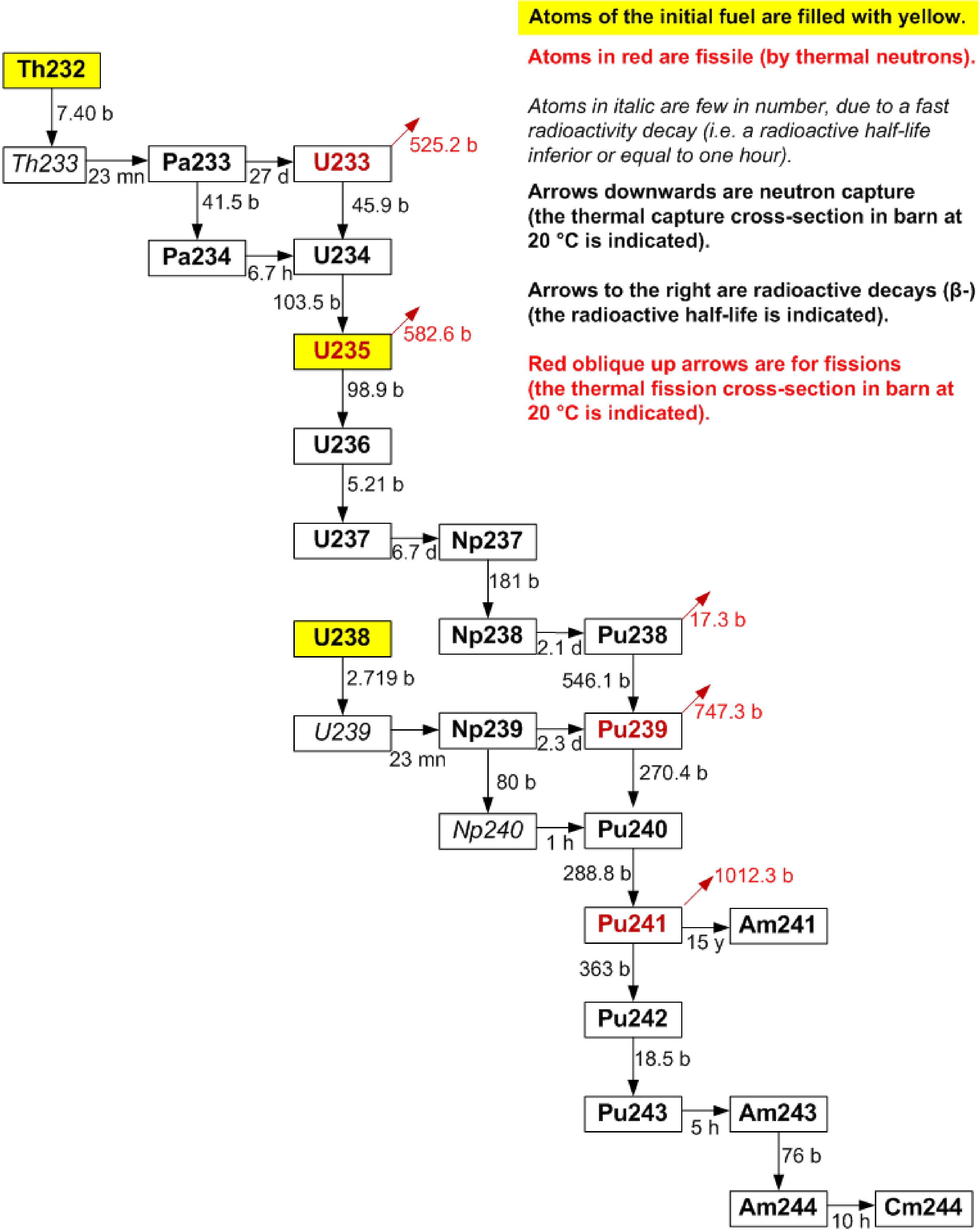
The PWR reactor surrounds the Beryllium inner wall at the level of the straight parts.
It is a « power amplifier » of the neutronic kinetic power Pneu.
So the fission power generated Pth_fission_W is proportional to Pneu.
The global power (in the form of heat) supplied to the steam generators is equal to:
Psg_W=Psg_fusion_W + Pth_fission_W
This heat is managed by the water/steam circuit, described below (for information)

For simplicity a « Rankine without superheating » cycle is shown below.

Figure 2 D-D/PWR hybrid reactor energy balance

5. Fuel evolution taken into account

In the figure 3 below, it will be found the simplified chain of the managed fuel (Thorium+Uranium).



Atoms of the initial fuel are filled with yellow.

Atoms in red are fissile (by thermal neutrons).

Atoms in italic are few in number, due to a fast radioactivity decay (i.e. a radioactive half-life inferior or equal to one hour).

Arrows downwards are neutron capture (the thermal capture cross-section in barn at 20 °C is indicated).

Arrows to the right are radioactive decays (β-) (the radioactive half-life is indicated).

Red oblique up arrows are for fissions (the thermal fission cross-section in barn at 20 °C is indicated).

Figure 3. Simplified evolution chain of a Thorium (Th232) + Uranium (U238+U235) fuel

6. Example of working with a 50/50 mixture of natural Uranium and natural Thorium over 100 years

On the figure 4 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 discontinuities on the effective multiplication factor K_eff, before limitation, are due to the refueling operations. 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). The net electrical power is constant because the K_eff is, in fact, limited to 0.9 with borated water. Note that without borated water, the reactor would remain sub-critical (K_eff<0.96).

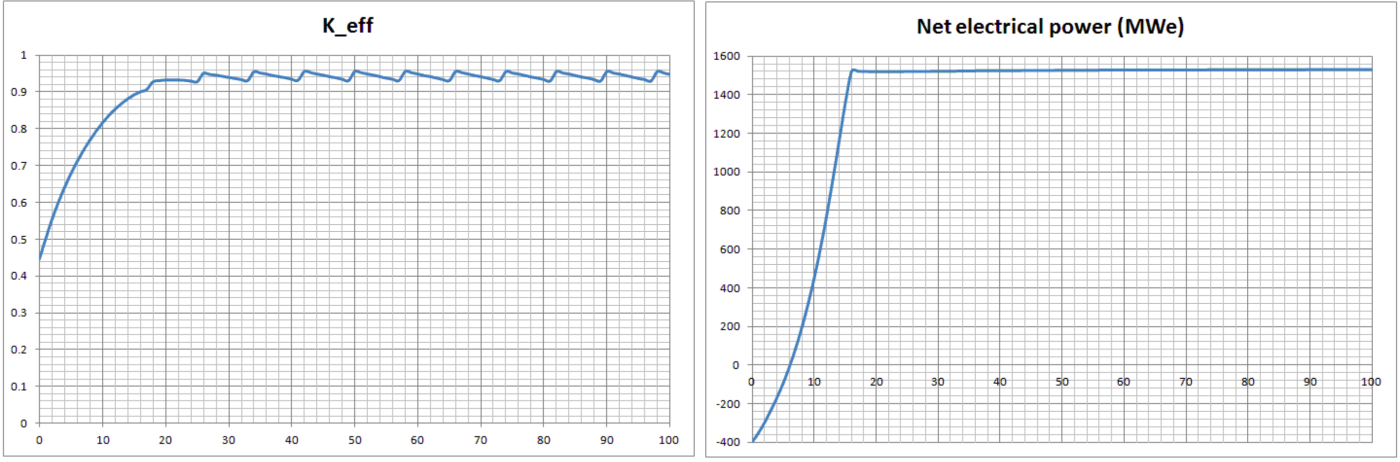


Figure 4. Main results for a 50/50 mixture of natural Uranium and natural Thorium with respect to time in years

7. Conclusion

This reactor is able to successfully incinerate any natural or depleted nuclear fuel, for thousands of years of world consumption.

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.

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 if possible or, more slightly, the maximum K_eff.

The expected life 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. 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).