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An Inertial-Fusion Z-Pinch Power Plant Concept

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Abstract

With the promising new results of fast z-pinch technology developed at Sandia National Laboratories, we are investigating using z-pinch driven high-yield Inertial Confinement Fusion (ICF) as a fusion power plant energy source. These investigations have led to a novel fusion system concept based on an attempt to separate many of the difficult fusion engineering issues and a strict reliance on existing technology, or a reasonable extrapolation of existing technology, wherever possible. In this paper, we describe the main components of such a system with a focus on the fusion chamber dynamics. The concept works with all of the electrically-coupled ICF proposed fusion designs. It is proposed that a z-pinch driven ICF power system can be feasibly operated at high yields (1 to 30 GJ) with a relatively low pulse rate (0.01 – 0.1 Hz). To deliver the required current from the rep-rated pulse power driver to the z-pinch diode, a Recyclable Transmission Line (RTL) and the integrated target hardware are fabricated, vacuum pumped, and aligned prior to loading for each power pulse. In this z-pinch driven system, no laser or ion beams propagate in the chamber such that the portion of the chamber outside the RTL does not

need to be under vacuum. Additionally, by utilizing a graded-density solid lithium or fluorine/lithium/beryllium eutectic (FLiBe) blanket between the source and the first-wall the system can breed its own fuel absorb a large majority of the fusion energy released from each capsule and shield the first-wall from a damaging neutron flux. This neutron shielding significantly reduces the neutron energy fluence at the first-wall such that radiation damage should be minimal and will not limit the first-wall lifetime. Assuming a 4 m radius, 8 m tall cylindrical chamber design with an 80 cm thick spherical FLiBe blanket, our calculations suggest that a 20 cm thick 6061-T6 Al chamber wall will reach the equivalent uranium ore radioactivity level within 100 years after a 30 year plant operation. The implication of this low radioactivity is that a z-pinch driven power plant may not require deep geologic waste storage.

Introduction

Obtaining controlled fusion reactions as a means for generating power has proven to be a very difficult problem. Of the many proposed ways to create fusion conditions, only magnetic confinement fusion (MCF) and inertial confinement fusion (ICF) have gained widespread attention and significant financial support. Within these two divisions, multiple fusion reactor designs have been developed for the many different ways of confining or driving the reactions (Bolton, H. R. et. at. 1989b;Call, C. J. and Moir, R. W. 1990;Kulcinski, G. L. et. at. 1994;Moir, R. W. et. at. 1994;Turchi, P. 1984). Typically whenever a new potential new fusion technology is conceived or tested, a new reactor design is based around that scheme. Fast z-pinches developed as Sandia National Laboratories (SNL) have demonstrated the ability to efficiently produce thermal x-rays with temperatures and time scales nearly appropriate for driving ICF capsules. Keeping this source in mind, we are investigating one way to build a fusion power system with mechanical electrodes that are destroyed and rebuilt after each power pulse.(Slutz, S. et. at.)Utilizing a recyclable direct mechanical contact between the driver and the target has a number of advantages. These include standoff (separating the expensive, repetitively operated primary power source from the damage of the target emissions), precise target alignment, and eliminating the need for a direct

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line of sight between the driver and the target. In addition, because the small gap of the current-carrying transmission line is the only space required to be under vacuum, the remainder of the target chamber can be kept at atmosphere. In this paper, we illustrate the main functional engineering issues of a z-pinch driven ICF fusion power plant and contrast some of these issues against other reactor concepts.

Before beginning a detailed discussion of a z-pinch driven ICF reactor, we give a brief introduction to fast z-pinches.(Liberman et. al. 1998;Ryutov, D. D., Derzon, M. S., and Matzen, M. K. 2000) A z-pinch is the radial implosion of a cylindrical or annular plasma under the influence of a strong magnetic field produced by a current flowing down the length of the plasma. This magnetic field concept originated in the 1930'swhen Tonks suggested the term z-pinch.(Tonks, L 1937) Usually, it involves the ionization and subsequent implosion of a gas for time-scales on the order of microseconds. At Sandia National Laboratories, z-pinches are driven by the Z machine which typically delivers 20 MA of current through more than 300, ~7 μm diameter tungsten wires arranged in a 2 cm radius, 1 cm tall cylindrical ring.(Spielman, R. B. et. at. 1998) The wires vaporize forming a very-uniform plasma sheath that implodes under the force of its own radial magnetic field onto a low-density foam or annular foil. This compression heats the interior of the foam to temperatures as high as 230 eV (2.7 million degrees centigrade).(Nash, T. J. et. at. 1999;Peterson, D. L. et. at. 1999) The thermal x-rays emitted during the course of the implosion contain up to 1.8 MJ of total energy and radiate for about 10 ns.(Spielman, R. B. et. at. 1997) This technology therefore differs from the classical z-pinch in that fast z-pinches can create high-level radiation environments on time scales similar to those created in indirect-drive laser hohlraums or ion-beam ICF drivers.

Fig. 1(a) shows the top-level current transmission line on Z and Fig. 1(b) shows a schematic of an internal z-pinch driven ICF target scheme called a dynamic hohlraum. The transmission lines on Z are made of aluminum at large radii and stainless steel near the target. The dynamic hohlraum configuration in Fig. 1(b) is composed of a tungsten sheath to contain the radiation and a ~10 mg/cc, ~5 mm radius foam to contain the capsule, shape the radiation pulse, and hold-off the imploding plasma. Fig. 2 shows the x-ray emission time-histories inside a dynamic hohlraum on Z as tailored for driving ICF targets. This plot shows the degree of pulse-shaping which has been developed in the z-pinch program at SNL. The next step in verifying the applicability of this concept is to measure the radiation symmetry that exists at the ICF capsule location within the dynamic hohlraum configuration. There are other proposed target configurations and we refer the reader to the open literature for a more complete description.(Hammer, J. H. et. at. 1999;Leeper, R. J. et. at. 1999) Simple scaling from current and previous pulsed power z-pinch machines indicate that a next generation z-pinch driver which generates ~60 MA of load current could produce almost 10 MJ of x-ray energy for driving an ICF capsule.(Olson, R. E. et. at. 1999a) There are a number of power feed configurations target configurations and one of them uses two of these high current power feeds.(Hammer, J. H., Tabak, M., Wilks, S. C., Lindl, J. D., Bailey, D. S., Rambo, P. W., Toor, A., Zimmerman, G. B., and Porter, J. L. 1999) We believe that the final yield will be in the 3-12 GJ range but for this paper we consider the range 1-30 GJ and this is what is used in the examples. In addition, we envision that the individual chamber rep-rate will need to be less than once per 10 seconds. Multiple chambers may be required, in fact desired, from a perspective of plant maintenance and operation.

There are many fusion target designs containing (Lindl, John 1998;Linhart, J. G. 1998) (Lindemuth, I. R. and Kirkpatrick, R. C. 1983a) and reactor configurations that can be found in the literature (Bolton, H. R., Choi, P., Dangor, A. E., Goddard, A. J. H., Haines, M. G., Peerless, S. J., Power, A., and Walker, S. P. 1989a) In addition, there are a number of other electrically-coupled fusion concepts which do not rely on inertial fusion. (Bolton, H. R., Choi, P., Dangor, A. E., Goddard, A. J. H., Haines, M. G., Peerless, S. J., Power, A., and Walker, S. P. 1989b) (Lindemuth, I. R. and Kirkpatrick, R. C. 1983b)The power generating system described in this paper can be modified to fit many of these different driver and target concepts. (Hartman, C. W. Carlson G. Hoffman M. R. Werner 1977;Turchi, P.)

Overview

The concept described below has been named the Z-Pinch Power Plant or ZP-3. The philosophy around which the ZP-3 concept has been developed is to attempt a design that can be built using present day technologies and materials (or a reasonable extrapolation of these technologies) whenever possible while separating difficult problems.(Derzon, M. S. and et al. 1999) We attempt to separate the difficult problems in a fusion chamber system and solve each independently. This means that the first-wall is separate from the breeder, moderator, and coolant, and the driver, target alignment , and vacuum punping are decoupled from all the above. This can provide a significant design advantage and has been discussed in some detail by others.(Avci, H. I. and Kulcinski, G. L. 1979;Moir, R. W. 1995) In addition, intentions at the onset of this study were to minimize the overall structural activation and damage by utilizing large amounts of low Z material between the target and the first-wall. Many fusion system first-walls must be replaced every few years whereas this one is intended to last the entire system lifetime.(Moir, R. W. 1996) The rather stringent requirements laid out above lead to some simple, inescapable conclusions. The first is that, because today's technology requires hardware contact between the target and the machine, then ZP-3 must have a current carrying structure that is replaceable every power pulse. This structure is termed the Recyclable Transmission Line (RTL) and is presently envisioned to be made of either the same material as the moderator or aluminum.(Slutz, S. and et al) In order to protect the permanent structure of the chamber from damage, the RTL must provide standoff from the target and may be up to 270 kg (Derzon, M. S, et al. 1999) in worst case conditions or may be only a few kg under optimal conditions (Slutz, S., Olson, C. L., Rochau, G. E., Derzon, M. S., Peterson, P. F., Degroot, J. S., Jensen, N., and Miller, G.). The pulse rate must be reasonably low to allow for extraction and replacement of both the RTL and the moderator/coolant material between power pulses. In addition, the yield must be high to accommodate the economics of the target and RTL. These costs are expected to be much higher than the cost per pulse for conventional ICF system targets. Fig. 3 shows the relationship between target yield, RTL/target cost, and pulse rate.(Call, C. J. and Moir, R. W. 1990;Logan, B. G. 1993) This assumes a 1

GW_e plant operating with a 33% efficient conversion from thermal to electrical energy and a RTL/target cost determined from:

$$\text{RTL/Target Cost} = 0.1\left(\frac{\$0.05}{\text{kWh}}\right)\left(\frac{1\text{ kWh}}{3.6\times10^{6}\text{ J}}\right)(0.33)\text{Y}, \qquad (1)$$

where electricity is assumed to cost \$0.05 per kWh, Y is the target yield in J, and the RTL/target cost makes up 10% of the overall cost of electricity. For these parameters, a 1 GW_e plant with a target yield of 1 GJ must shoot 3 times a second at a RTL/target cost of about \$0.5 per pulse. Due to energy conversion and thermal cycle considerations, this is considered the lower limit on the target yield. It is reasonable to consider a complete power plant made up of multiple chambers or modules, each of which have a 1 GJ or greater target yield and are operated at less than 1 Hz. If ZP-3 contains only 1 module, it is assumed that a 30 GJ target yield is the required upper limit. This indicates a 0.1 Hz pulse rate with a combined target and RTL cost of about \$14 per pulse. In contrast, a 3 GJ yield and the associated target costs of \$1.4 per pulse would be acceptable. If the cost for a given module is low enough to allow for a ten module power plant, then each module would only need to fire a 3 GJ target once every ten seconds. Consider this in conjunction with the use of multiple chambers; the pulse rate per chamber is reduced as well as the mass flow rate per chamber down the whole scheme becomes more viable in terms of the difficulties with filling and moving materials through the chamber. Unlike systems with nominally 8-shots/second that require reaching high vacuum in one big chamber this design promises less mass, less stringent vacuum pumping requirements and a more reasonable length of time to replace the hot moderator and RTL/target assembly.

A cost estimate has been conducted for the manufacturing of the RTL and target structures for the ZP-3 system.(Zamora, A.) Assuming a 270 kg (600 lb) RTL plate structure, the manufacturing costs were estimated at \$0.7 per RTL plate, where each RTL uses two plates. This estimate was made assuming technology and manufacturing techniques that exist today. This does not include the cost for equipment, materials, or the elemental separation that will be required after each pulse. However, because the material will be reprocessed and reused, the material costs should be only a small fraction of the manufacturing costs. In addition, the elemental separation process should be straightforward and therefore impart an operational cost that is similar to or less than that for the RTL manufacturing. Calculations conducted by Slutz((Slutz, S., Olson, C. L., Rochau, G. E., Derzon, M. S., Peterson, P. F., Degroot, J. S., Jensen, N., and Miller, G.)) indicate a theoretical minimum mass for the RTL of on the order of 1 kg in order to achieve acceptable electrical performance. This would lead to very low RTL costs,as compared to the estimates for massive RTLs of 0.7$ per. Besides the RTL structure, the target costs for the ZP-3 system (assuming the same z-pinch wire array configuration currently used on the Z machine) have been estimated at $0.4 per assembly .(Zamora, A.) This estimate does include the cost of the tungsten wires. By collecting and recycling the vaporized wire array materials these costs could be even lower than reported. These initial estimates give some indication that the economics of a z-pinch driven ICF power plant may be acceptable.

The very high yield in the ZP-3 system feeds directly into the other requirement that a massive amount of low Z material is placed between the target and first-wall. A 1-30 GJ yield can vaporize and melt a reasonable amount of material and therefore lends itself to a system design with a thick blanket. More specifically because the system must also breed tritium, shield the first-wall from neutrons, and absorb the released fusion energy, we intend to design an integrated blanket of low Z breeder material that does all of the above as well as mitigate the target induced mechanical shock. Finally, because of the thick blanket structure that exists around the target, the first-wall can possibly be made to survive for the entire plant lifetime and require no geologic storage for the relatively short cooling period. Fig. 4 is a chart showing the relationship between components of the power plant described above.

In Fig. 5 we show a schematic of a single ZP-3 system module (Derzon, M. S. 2000) and Fig. 6 shows the overall vision of a ZP-3 power plant with 12 modules. The target carousel in Fig. 5 is used to manipulate targets, transmission line and potentially the moderator into the fusion chamber. After the lid is closed and the moderator is in place then the event occurs. After the fusion event the hot material is flushed out through the bottom of the chamber and the chamber can be reloaded and another target fired. After being flushed out of the chamber the hot moderator/coolant/RTL material is processed through a heat exchanger and refabricated for more events. The large building at the center of the ZP-3 power plant complex, shown in Fig. 6, is the material collection and re-manufacturing center. The pre-pumped and pre-aligned cartridges (integrated blanket, RTL, and target hardware assemblies) are distributed to the individual modules while the post-pulse material is pumped back to the manufacturing center for recycling. The concept as discussed in this paper does not preclude the possibility of directly converting the fusion energy into electricity, but direct conversion is not discussed in order to reduce complexity .

The overall vision of a ZP-3 system module is a rep-rated pulse power driver which, every few seconds, couples current through a pre-pumped, pre-aligned RTL to a z-pinch load. The load utilizes fast z-pinch technology to convert the electrical energy to thermal radiation which drives an ICF target up to a 30 GJ yield. The neutron and photon flux from the high-yield target vaporizes much of a low Z blanket and liquefies the rest. In the process, the neutron flux through the blanket breeds tritium, deposits the bulk of the fusion energy, and allows only multiply scattered neutrons to reach the first-wall. These low energy neutrons do less damage to the wall, comparable to a LWR pressure vessel, and cause little activation possibly allowing for a 30 year lifetime based on neutron damage alone. The vaporized/liquefied blanket material is pumped from the target chamber through a heat exchanger and tritium extractor while the leftover blanket slag is collected for recycling. In the end, the concept version of ZP-3 is a 1 GW_e power plant which produces economical power with no proliferation concerns, little greenhouse emission, and little long-lived radioactive waste. In a more complete analysis the thermal cycle, the tritium breeding ratio, and the chamber shock and damage considerations will determine the optimal yield. These are not specified or chosen here and therefore no attempt is made to determine the optimal yield per shot or other characteristics of this power plant concept.

Chamber Neutronics

A complete design of a fusion system (based on the chamber neutronics requires a target calculation that defines the neutron spectra and time-dependant fluence. To avoid the complicated design of a target with a fusion yield up to 30 GJ and still describe the important issues for the power plant concept we have relied on some simple observations of a 320 MJ target designed for the z-pinch driven X-1 high-yield test facility. (Olson, R. E., Chandler, G. A., Derzon, M. S., Hebron, D. E., Lash, J. S., Leeper, R. J., Nash, T. J., Rochau, G. E., Sanford, T. W. L., Alexander, N. B., and Gibson, C. R. 1999b;Rochau, G. E., Hands, J. A., Raglin, P. S., and Ramirez, J. J. 1998) Fig. 7 shows the radial profile and neutron spectrum of this X-1 target as designed by the 1-D Lagrangian radiation magneto-hydrodynamics code BUCKY.(Peterson, R. R., Macfarlane, J. J., Santarius, J. F., Wang, P., and Moses, G. A. 1996). The neutron spectrum is calculated using a 1-D deterministic code called ONEDANT (Alcouffe, R. E. 1990) . This code accepts the neutron production and target conditions as input from BUCKY and calculates the associated detailed neutron transport. The characteristic 14.1 MeV D-T neutrons make up about 70% of the total neutron yield while scattered neutrons make up the other 30%. Although the average neutron energy is reduced in the target, we will assume the neutron source to be purely 14.1 MeV neutrons for the duration of the paper. This faster neutron spectrum generally leads to lower overall tritium breeding in the blanket, and higher activation levels in the chamber wall providing less target dependant, but more conservative calculations of the chamber neutronics. It should also be noted that the effects of higher energy neutrons, those generated by slowing reaction products in the fuel, are ignored. X-ray deposition is not accounted for in this discussion because it dies not contribute significantly to the activation or coolant heating at this level of detail. The ion heating from the target will be very relevant to thermal cycle and chamber damage calculations that we deem outside the scope of this paper.

Energy Conversion and Tritium Breeding

In order to calculate the required blanket thickness for proper energy conversion and tritium breeding, we developed a simple 1-D spherical model similar to that developed and studied by Sahin et. al. (S.Sahin, R.W. Moir, J.D. Lee, and S. Unalan 1994) The model consists of a spherical shell of breeder material with a 5 cm inner radius and a shell thickness, ΔR, up to 200 cm. This geometry is shown schematically in Fig. 8. The blanket materials studied were lithium

(Li), a lithium-lead mixture of 83% lead and 17% lithium by mass (Pb-17Li), and a molten salt with a chemical symbol defined by $(LiF)_2(BeF_2)$ called FliBe.

The neutron scattering and tritium breeding calculations were conducted in COG (Wilcox, T. and Lent, E. 1910) an arbitrary geometry Monte-Carlo neutron and photon particle transport code developed at Livermore National Laboratories for deep neutron penetration studies. COG uses the ENDL and EPDL cross-section libraries for all neutron and photon interaction calculations and provides an output for the frequency and type of interactions that occur throughout the geometry. The target was modeled as a 14.1 MeV monoenergetic point source at the center of the spherical shell and a delta function in time. Tritium breeding occurs in the blanket through the two lithium nuclear reactions described by:

$$\begin{aligned} {}^{6}Li + n &\rightarrow {}^{4}He + {}^{3}H \\ {}^{7}Li + n &\rightarrow {}^{4}He + n + {}^{3}H. \end{aligned} \qquad (2)$$

Summing the ${}^{6}Li(n,\alpha)T$ and ${}^{7}Li(n,n'\alpha)T$ reactions and taking the ratio of the sum to the total number of input particles in the Monte-Carlo calculation, the tritium breeding ratio (TBR) can be calculated for each simulation. By setting COG to calculate the energy deposition in the blanket volume from both neutron and neutron induced photon (n,γ) interactions, a more accurate calculation of the energy conversion can be determined. COG quotes detector results in units normalized per source particle. Thus, the energy conversion ratio is determined by dividing the sum of the energy deposition and the 3.5 MeV assumed charged particle conversion by the total 17.6 MeV produced in a single fusion reaction. Finally, the effective shielding factor is determined for each pulse by multiplying the attenuated neutron flux at the edge of the blanket by $4\pi(5 + DR)^2$.

For blanket thicknesses, ΔR up to 200 cm, are shown in Fig. 9. It is noted that Pb-17Li is not a reasonable alternative simply because the (n,2n) reactions increase the neutron flux at the blanket edge, the TBR is the lowest for all blanket thicknesses, and the energy conversion ratio is only slightly better than natural Li. The best alternative in all the categories of Fig. 9 is FLiBe, where an 80 cm blanket has a TBR of 1.2, an energy conversion ratio of 1.08, and a flux reduction factor of 0.08. It is of particular interest that the energy conversion ratio is over 1 in this case. This is due to the positive Q (n,T) and (n,n'T) neutron interactions which occur in the FLiBe.

The obvious temptation is to consider only FLiBe as the blanket material not only because it has favorable neutronics parameters, but also because the fluorine makes it less reactive with the structural materials. FLiBe has been manufactured and tested in small quantities, but the overall properties (S.J. Zinkle) are not as well understood as those of Li metal. Li should also be considered because of the additional safety complications associated with the Be in FLiBe, and because the Li TBR can be raised far above 1.2. This may be required by other applications such as the transmutation of fission reactor waste. Burning fission reactor waste becomes feasible with high tritium breeding ratios because the excess tritium can be used in additional chambers without the need to breed fuel. Instead of breeding more tritium the neutrons can be used to transmut e fission reactor wastes. The bottom line here is that it is too early to tell which material will prove to have better characteristics in a power plant. Since it is too early in the design of the ZP-3 system to choose one material over the other, both the natural Li and FLiBe blankets will be considered .

Neutron Damage Impact on Wall Survivability

In addition to the important blanket factors described above, the wall survivability of ZP-3 is an important advantage of this high yield system. The major factor in wall survivability for a fusion system is the neutron damage at the first-wall. The ionizing radiation emitted from a fusion target causes lattice damage to the crystal structure that eventually weakens the material thereby necessitating its replacement. A common rule of thumb for fusion system wall survivability is that a steel first-wall can withstand 10 MW/m^2 for a one year lifetime.(Duderstadt, J. J. and Moses, G. A. 1982)

Because of the thick blanket structure between the ZP-3 target and the first-wall, x-ray or gamma interactions with the first-wall are considered to have a negligible effect. However, as dictated in the previous section, the penetrating neutron flux can be significant. Thus, we have developed a simple methodology to determine the lifetime of the first-wall due to the interaction of radiation with matter given a neutron flux and spectrum at the outer edge of the neutron blanket. The 10 MW/m^2 threshold for a 1 year operation corresponds to a total lifetime energy fluence, Φ_T, of 2.0*10^{27} MeV/cm^2. This is assumed to be the total energy fluence that the first-wall can sustain before it needs to be replaced. The number of operational years which the ZP-3 wall will last can be calculated by re-analyzing the neutronics models discussed in section 2.1. By binning the neutron spectrum at the edge of the blanket into multiple energy groups, we calculate the energy fluence at the first-wall as normalized per target neutron by:

$$\Phi_E = \frac{(5+DR)^2}{R_c^2}\int E\frac{d\Phi}{dE}dE \quad \left\{\frac{\text{MeV}}{cm^2}\right\}, \tag{3}$$

where $d\Phi/dE$ is the differential particle fluence per target neutron as calculated by COG and R_c is the radius of the target chamber in cm. The lifetime of the system wall due to neutron damage is then calculated from:

$$\text{Lifetime} = \frac{\Phi_T E_f}{\Phi_E Y f}, \tag{4}$$

where E_f is the energy release per fusion, Y is the total target energy yield, f is the pulse frequency, and no consideration has been given to system maintenance time.

Equations (3) and (4) were calculated for natural lithium and FLiBe blankets with thicknesses up to 200 cm and for chamber radii ranging from 100 cm to 800 cm. These calculations assume the worst case scenario of a single ZP-3 module yield with a 14.1 MeV monoenergetic neutron spectrum. The lifetime values for these parameters are summarized in Fig. 10. As was mentioned above, the natural Li blanket must be at least 95 cm thick to achieve proper neutronics performance. Assuming a 400 cm radius chamber, this corresponds to a neutron damage-limited lifetime of only about 12 years. For the first-wall to last a full 30 years either the chamber radius must be about 600 cm or the Li blanket must be equivalent to 120 cm thick. However, one must keep in mind that this is for containment of the highest possible yield for a ZP-3 module. If there are 10 chambers each containing a 3 GJ target every 10 seconds, then the wall lifetime for a 95 cm thick Li blanket would be about 120 years for each module. The FLiBe blanket serves the purpose much better than Li. For the required FLiBe blanket thickness of 80 cm discussed in section 2.1, a 400 cm radius chamber would have a neutron damage-limited lifetime of over 300 years. In the FLiBe case, the minimum chamber radius allows for a full 30 years of operation before reaching this damage limit is nominally 110 cm.

Activation

The activation of the chamber structure is another problem that plagues many ICF and MCF systems. Because of the relatively low energy release per fusion reaction, fusion systems based on DT or DD reactions will have a neutron fluence higher than that in present day fission reactors. This means that, although pure fusion power plants may have less long-term radioactive waste than a fission power plant, they still require waste storage. Many initial studies of fusion system systems ignore this important issue of activity in favor of more lengthy discussions of economics and/or systematics. In today's political climate, the activation issue is as important as any of the neutronics parameters and should be addressed from the onset of any new system design.

The ultimate determination of activity as a function of time after system shutdown is a strong function of the details in how the system is configured. The permanent structure of a fusion system may be activated to much higher levels and often require storage for thousands of years after operation. The ZP-3 utilizes a large amount of low Z material placed close to the target such that the energy fluence at the first-wall is very low. The result is that this power plant design has the potential of not requiring geologic time scale disposal of waste (which we define as less than 1000 years). The threshold we have set for the storage of an activated material is the activity level of natural uranium ore. This is arbitrary, but it is the lowest or most conservative estimate that we consider reasonable as a regulatory treatment of radioactive waste from a fusion plant.

The activation of the chamber and the moderator materials were estimated by again utilizing the simple model introduced given earlier. In order to parameterize the activation of the ZP-3 chamber, these blanket models were enclosed by a 6061 - T6 Al chamber. The chamber was modeled as a 20 cm thick, 800 cm tall, and 400 cm radius cylindrical chamber with 20 cm thick Al endcaps. This geometry is shown in Fig. 11. For simplicity, these models do not include any RTL structure or the details of the RTL insulator stack at the chamber wall.

Activation calculations for this chamber design were conducted with the ALARA activation code developed at the Fusion Technology Institute .(P.P.H. Wilson and D.L. Henderson) This code has been developed specifically for calculating the activation of fusion system materials subject to an arbitrary pulsing schedule with a known neutron and/or photon flux. The required 140 group neutron spectra in ALARA is calculated by a COG model with the geometry discussed above. The upper half space of the system wall in this COG model is divided into 10 segments of equal volume and the fluence through each region is averaged over the wall thickness. This COG generated neutron fluence is normalized to a total neutron flux for 3 $GW_{thermal}$. We then calculate all the nuclear interactions occurring in these wall and moderator volumes for any specified material and keep track of the multiple generations of daughter nucleides in ALARA.

The results of this activation scan are shown in Fig. 12. Calculations for select Li blanket thicknesses up to 200 cm are given and the activity of the chamber is reported in Ci/GW_e. Background natural uranium activity is assumed to be 210 Ci/GW_e and is also shown in Fig. 12 for reference. The results of this scan indicate that the chamber wall can reach natural uranium levels in less than 1000 years for Li blanket thicknesses greater than 70 cm. It should be noted that 6061 - T6 Al is a low activation material such that if the chamber wall was made of steel, these levels would be quite a bit higher. This activity scan was not conducted for FLiBe, but because of the significantly higher density and the degree of neutron moderation, the activation levels are expected to be much lower.

To develop an idea of what the activation of the ZP-3 chamber structure might be for other materials, COG and ALARA calculations were conducted for ZP-3 chambers made of SS318, 2.25 Cr - 1 Mo steel, and 6061 - T6 Al. The results are given in Fig. 13 (a) for a 95 cm thick Li blanket and in Fig. 13 (b) for an 80 cm thick FLiBe blanket. The activation of the ZP-3 chamber wall is plotted as a function of time after shutdown for each of the structural materials after a 30 year continuous operation. Also plotted in Fig. 13 (a) and Fig. 13 (b) are the total activation of a once-through LWR, the activation of the ZP-3 chamber assuming no blanket, and the activity of natural uranium ore for reference. Note that without a Li moderator, similar in many respects to a MFE system, there is no advantage in the level of activity for the ZP-3 chamber over that of an LWR up to 10,000 years after shutdown. However, using low-activation materials such as the 2.25 Cr - 1Mo steel and 6061 - T6 Al, the activity level is much less over time. The 6061 - T6 Al chamber reaches natural uranium activity levels within 700 years for the Li blanket and within 100 years for the FLiBe blanket. This is a distinct advantage that ZP-3 has overother fusion system designs and all fission reactors and fuel.

Obviously, there are other waste products beside the system wall to be considered. The target hardware will likely require some high Z materials that will vaporize in the explosion and redeposit somewhere in the coolant blanket as it is pumped out of the chamber. However, by recycling these materials through elemental separation and recasting them in more target structures, the overall volume of this waste can be kept to a minimum. The same situation exists for the RTLs that will likely be made of either low activation Al or be cast out of the same material as the blanket and coated with a conductor.

Mechanical Considerations (the RTL, Stress at the wall, Thermal Cycle and Mass Flow)

The previous sections have outlined the neutronic characteristics of ZP-3. The system can breed its own fuel, convert the fusion energy into thermal energy, and reduce the neutron energy fluence to the point where the first-wall can potentially survive this ionizing radiation for the entire system lifetime. What has yet to be considered are some basic issues of the RTL, mechanical stresses and shock from the loading of up to a 30 GJ nuclear energy release inside the chamber, as well as a simple conceptualization of the thermal cycle and mass flow through the chamber.

The RTL

The Recyclable Transmission Line (RTL) concept emerged at a workshop (*Z- Pinch Fusion for Energy Applications* at Sandia National Laboratories, Albuquerque, NM (Spielman, R. B. 1999)) and was developed further at the Snowmass workshop on Fusion Energy (Hawryluk, R., Logan, G., and Mauel, M., 1999(Slutz, S. and et al). The idea is to construct the final portion of the transmission lines out of material that can be recycled inexpensively . Materials such as lead, tin, carbon, and aluminium can easily be separated from FLiBe and thus are good candidate RTL materials. In addition it may be possible to use FliBe or Li in their solid states as a portion of the RTL. However, since solid FLiBe is an insulator, a conductive coating will probably be required.

A detailed sketch of how an RTL may appear is shown in Fig. 14. The connection between the recyclable and the permanent part of the transmission line is at the top of the reactor chamber. Note that the RTL has an advantage over all other existing approaches to inertial fusion, which is that the RTL does not have to go in a straight line. For example, as shown in Fig. 14, the RTL can have a right angle bend, that allows for shielding the x-rays and the blast wave from the fusion explosion from the delicate parts of the driver. These are the convolute , the vacuum interface and the permanent connection hardware. In addition, the RTL can be at vacuum before being installed in the chamber. In contrast, a laser or ion driver always has the problem of the last optic element and pumpout is after installation. A coaxial RTL can be used with a dynamic hohlraum (Brownell, J. H. R. L. Bowers K. D. McLenithan D. L. Peterson 1998) (capsule as indicated in the figure. The use of doubled ended z-pinch driven hohlraum (Hammer, J. H., Tabak, M., Wilks, S. C., Lindl, J. D., Bailey, D. S., Rambo, P. W., Toor, A., Zimmerman, G. B., and Porter, J. L. 1999) would require the use of a triaxial feed. The labelled RTL portion of the transmission line will be blown up with each detonation of the capsule located within the z-pinch. This material will be recycled to form a new RTL for subsequent detonations. A particularly attractive option (Peterson, P. F., Cole, C., Donelli, A., and Olander, D. R. 1999) is to use a lithium compound such as FLiBe, since lithium will be in the reactor anyway to provide cooling and tritium breeding. The FLiBe need a coating of a chemically compatible metal conductor, since solid FLiBe is an insulator. This material must be readily separated from FLiBe. Tin and lead are two good candidate materials.

The RTL must have the proper dimensions so that it operates as a self magnetically insulated transmission line. Efficient power flow has been demonstrated with a mm gap between the anode and cathode near the z-pinch. To maintain magnetic insulation the gap must increase with distance from the pinch. The required gap is approximately

$d(z)=d_0\exp(\beta z)$, where $\beta = \dfrac{1}{ct_r\sqrt{F_L}}$, F_L is the fraction of the peak current that is needed before the RTL will be self magnetically insulated, and t_r is the current rise time. Note that some current is always lost at the beginning of the pulse. The inductance and the required driving voltage can then be calculated for a given current and pulse risetime. As an example, a 4 meter long RTL carrying a current of 100 MA with a pulse risetime of 150 ns would require a driving current of approximately 5 MV. This is a nominal voltage increase above the 3 MV delivered by the Z machine. However, the RTL will probably have to be run over insulated as is done on the Z machine. This would raise the required voltage. We need to determine if transmission lines constructed from these materials will efficiently transport electrical power at the large currents required of a fusion driver. An initial attempt was made to explore the RTL concept.

Experiments were performed recently at the Saturn facility at Sandia in order to test the issue of current loss and power transmission in stainless steel, aluminum, and aluminum coated with 100 μm carbon. These experiments were a preliminary attempt to observe the effect of material on transmission line performance in a configuration crudely scaled to reactor power plant conditions. Resources limitations constricted the experiments to a test of steel, aluminum, and carbon coated aluminum transmission lines. There was no indication of current loss in any of the RTL experiments. This is good news in that power flow is apparently not extremely sensitive to the electrode material. However, several things should be kept in mind. First, the RTLs in this experiment were fairly short (30 cm) when compared to a reactor sizes RTL that will be several m. Second, the bottom end of the RTLs in this experiment were shorted. A reactor scale RTL will have a z-pinch load. Since the total inductance of a reactor RTL will determine the machine driving voltage that is required, this inductance should be minimized. This means that the RTL will operate near the magnetic insulation limit. This may drastically increase the sensitivity of power flow to the choice of electrode material. Subsequent experiments we will be designed to be sensitive to these features.

Stress at the wall

Analyzing the stress and shock at the chamber wall is a difficult problem and requires a detailed knowledge of the blanket and chamber wall design. In the absence of a detailed analysis we will extrapolate from prior work. The Fusion Technology Institute at the University of Wisconsin Madison has conducted a structural fatigue analysis. This analysis was for a pressure loading due to a 1 GJ fusion target on the chamber wall of the Light Ion Fusion Laboratory Microfusion Facility (LMF).(Badger, B. and et. al.) These structural fatigue calculations considered a pressure impulse on a 150 cm radius cylindrical system wall made of either 2.25 Cr - 1 Mo steel or 6061 - T6 Al, each with a 2 cm thick graphite inner lining. It was determined that a 5 cm thick steel wall would survive the periodic loading of a 1 GJ pressure impulse for over 10^8 pulses. These calculations also indicate that the Al chamber must be roughly twice as thick as the steel chamber for the same number of pulses.

A considerable difference between the ZP-3 and LMF target chambers is the amount of material between the target and first-wall. The LMF chamber only contains a low pressure Xe fill gas while the ZP-3 chamber contains the entire RTL structure and at least 95 cm of Li or 80 cm of FLiBe. This large amount of mass could impart a significant momentum on the first-wall if the shock propagation is not properly mitigated. The LMF results give hope that the impulse loading will not be a limiting factor in the wall lifetime. However, detailed calculations and the associated experimental verifications need to be conducted to parameterize the expected mechanical response for a given blanket structure.

It is understood that a completely solid blanket is likely not the optimal configuration to mitigate shock. Some thought has been given to the advantages of using a chamber fill which contains either a foam, or many hollow Li or FLiBe crushable spheres (bubbles) which can mitigate shock by removing propagation energy through mechanical disassembly. The breeder blanket material could be pumped into the chamber as solid density bubbles or a low-density foam with a carrier gas such as helium. Bubbles are a good alternative because they can be pumped into the chamber at a high rate through large apertures. Shock propagation can then be mitigated simply and crudely by adjustments in bubble characteristics (such as wall thickness and diameter) and by grading their collective density as a function of radius from the target. This is a particularly applicable concept for the ZP-3 chamber in that the lack of a line-of-sight requirement makes it possible. If either this type of fill or the solid spherical blanket shells do not prove feasible, then the more typical IFE flowing metal wall could be used.(Moir, R. W. 1995)

2.4 *The Thermal Cycle*

The ability of ZP-3 to produce power relies on the conversion of fusion energy to thermal energy and the subsequent conversion of thermal energy to electrical energy. For the latter, an exchange of energy between the blanket and a secondary coolant must be completed in a loop outside of the system chamber, or a direct energy conversion scheme or Rankine cycle (Logan, B. G. 1993) must be applied. Each of these will require that the blanket material be in either a liquid or gaseous form. Thus, a simple calculation of the vaporization and liquefication for spherical Li and FLiBe blankets was conducted using an iterative temperature model. COG was again utilized to calculate the neutron energy deposition as a function of radius for the two materials. The blankets were divided into 5 cm thick zones and the total neutron and neutron induced photon energy deposition was tabulated as normalized per fusion neutron. This energy deposition was coupled with experimental values of the heat capacity(S.J. Zinkle) in both solid and liquid form and the temperature rise was iteratively calculated. The calculations included both the heat of fusion and the heat of vaporization in order to determine the state of each blanket as a function of radius. The calculation did not include the effect of heating at the inside of the moderator sphere due to the charged particle fusion products. These products carry approximately 25% of the energy and they will be important in determining the appropriate thermal cycle and more accurately assessing the role of shock dynamics in the chamber design, however, we will not attempt to account for their effect in this analysis. Fig. 15(a) shows the Li temperature, given the neutron deposition only and it will only be accurate until the thermal effects of the ions diffuses away from the inner moderator surface, as a function of radius assuming that the initial temperature before the pulse was just 50 K below the Li melting point of 453.69 K (0.039 eV). This was done for both a 1 GJ and a 30 GJ target yield. These yields were the extremes of what we felt were reasonable yields per shot. For reference, the TBR for a Li blanket is also plotted in Fig. 15(a). This indicates that, for a 30 GJ target yield, the blanket is vaporized up to a radius of about 40 cm and liquefied past a radius of 125 cm. For a 1 GJ target yield, the blanket is only vaporized out to 10 cm and liquefied out to 50 cm. Outside this radius, the Li is at the melting point where it is part solid and part liquid. Clearly, if the entire blanket must be circulated through a heat exchanger and the neutronics require a 95 cm thick Li blanket, then the target yield must be closer to 30 GJ than 1 GJ for this initially solid Li blanket.

The FLiBe blanket has a very different temperature profile due to its much higher density and much lower heat capacity. This temperature profile is plotted in Fig. 15 (b) for both a 30 GJ and 1 GJ target yield and the TBR for FLiBe is included for reference. In these calculations, the initial FLiBe temperature is again 50 K less than the melting point of 742 K (0.0638 eV). Fig. 15(b) indicates that even at a yield of 30 GJ, the FLiBe blanket is part solid for radii greater than about 70 cm. For a 1 GJ target yield, only about the first 30 cm are vaporized or liquefied and the remainder of the blanket remains solid. This implies that, if a FLiBe blanket is to be used as the breeder and moderator in a ZP-3 module, than either the solid pieces must be melted off-line to extract the tritium, or the blanket must be initially liquid. According to Fig. 9(b), the energy conversion ratio at a FLiBe thickness of 35 cm is about 80% such that there might be enough vaporized/melted FLiBe to heat exchange with an adequate efficiency. At any rate, as in the case of Li, if an initially solid blanket is to be used near the target as the breeder and moderator then the target yield should be closer to 30 GJ than 1 GJ for highest efficiency and ease of use. Of course, a more detailed analysis of the thermal cycle including the efficiency of heat exchange and the energy required to liquefy any unmelted material needs to be completed, but that is left for another study.

Discussion

This paper does not intend to describe a complete operational fusion power system from the pulsed power flow to energy conversion to a complete waste stream analysis. Instead, the scope of this study is to present a set of macroscopic scaling parameters for the target chamber of a rather new fusion power plant concept. This concept does not require the rapid vacuum pumping of a large chamber volume in which to deliver energy to the target. Instead, this energy is delivered through a pre-pumped RTL with out a need for a direct line-of-sight between the target and driver. The fusion targets are mounted directly to this RTL eliminating the in-flight accurate aiming and firing of the primary target driver often required in laser of ion-beam ICF systems.

This concept uses an initially solid blanket that is vaporized or melted by the fusion yield, and then flushed out with the liquefied or vaporized parts of the RTL and target. This technology allows for nearly minimal mass flow per unit energy over other fusion concepts. In turn this means potentially advantages in pumping cost over other concepts. Due to the graded density or solid density blanket structure placed close to the target, neutron damage should not be the limiting factor in the lifetime of the ZP-3 chamber wall. Most important, however, is that long-lived radioactivity in the chamber wall may not have the waste disposal problems which have plagued the fission nuclear power industry. In this concept, there is the potential for as little as a 100 year storage time before the chamber wall reaches natural uranium activity levels. One advantage of the ZP-3 system which has yet to be mentioned is that z-pinch pulsed power technology is more resistant to shock and debris than other fusion systems. It is thus a very suitable technology for applications to power plant or industrial activity. The combination of each of these advantages listed above may be the difference between an interesting technical study and a workable power plant.

To expand more on a couple of these benefits, we note that the issue of vacuum-pumping and machine robustness is non-trivial. It may actually be a critical advantage of the ZP-3 system. Experience on the Z facility (where only approximately 2 MJ of total energy is released) illustrates how difficult this pumping and long term chamber survival will be. This experience shows that a large amount of gas is released in the chamber creating high pressures immediately after a pulse. In addition, vaporized target material and unvaporized target debris leave dark soot on all surfaces and can impart significant damage to thick shielding materials. By flushing the blanket, RTL, and target materials out of the chamber between pulses, this damage could be less relevant because the mounting structures for the RTLs and associated hardware may be protected. By not requiring a line-of-sight access between the driver and the target, both the chamber wall and the RTL mounting structures can also be protected against neutron damage. In addition, by not requiring high vacuum in the target chamber and keeping to a fairly low pulse rate (< 1 Hz), the residual gases produced in the chamber should not be a problem.

These theoretical arguments should at some point be addressed experimentally. Some critical aspects of this z-pinch driven fusion power plant are already being shown in other venues. For example, the RHEPP facility is a pulsed power machine at Sandia National Laboratories which has operated for millions of pulses at kHz rates (albeit at much lower currents). In addition, the Z facility at SNL is demonstrating remarkable progress in driver performance.

While we have been thorough in our discussion of the benefits of a z-pinch driven ICF power plant, there are a number of significant issues regarding the ZP-3 system that have been overlooked or not seriously treated. This has been done in order to allow a discussion of more macroscopic problems without the complexities of the engineering details. Four of these issues which will require further attention are shock propagation, debris, economics, and recycled waste stream analysis.

The most important of these issues that is significantly different than other ICF system schemes is that of the mechanical shock propagation to the wall and the damage associated with fast-moving debris. Some attention has been paid to comparing the ZP-3 configuration to that of the LMF target chamber. To avoid making detailed calculations of this mechanical stresses and shock loading, the ZP-3 chamber has simply been scaled far beyond the requirements of the LMF mechanical analysis. This over-engineering is used to outline the feasibility of the ZP-3 system without requiring a detailed structural analysis. If this is not enough, it should also be noted that the use of small hollow spheres, or bubbles, of breeder material may be used to mitigate the shock propagation and debris. The trade-off is that it will require more material than the simple spherical blanket model discussed in this paper thereby increasing the required material flow rate. In addition to the graded chamber fill, direct conversion may reduce the debris problems by converting the associated kinetic energy into electrical energy. Each of these is a distinct possibility for the ZP-3 system and should be examined in the future.

Conclusions

This paper is an attempt to outline some of the basic physics and engineering issues within the target chamber of a z-pinch driven ICF power plant. We have sketched the elements and design considerations for such a facility, but this needs to be followed by more detailed analysis of each of the main elements discussed. Within those limitations we have described a power plant scenario with numerous advantages and possibilities. Evacuation of the chamber is not required. Target insertion and alignment constraints are minimal and can be met with present day technology. The solid blanket structure or possible chamber fill is potentially more robust than the liquid jet technology routinely considered for ICF. Neutron induced wall damage may be minimal, and there may be no need for geologic disposal of the permanent chamber structures. In addition, typical pulsed power environments are shock tolerant and do not require high cleanliness, a much more forgiving environment than that of either laser or ion-beam driven systems. Each of the beneficial issues introduced in this paper are sufficient justification for a more detailed study of this different type of fusion power plant. Although there are a great deal of improvements yet to be made, this inertial fusion energy approach exhibits great promise as a potential fusion power concept and should be vigorously pursued and supported.

Acknowledgements

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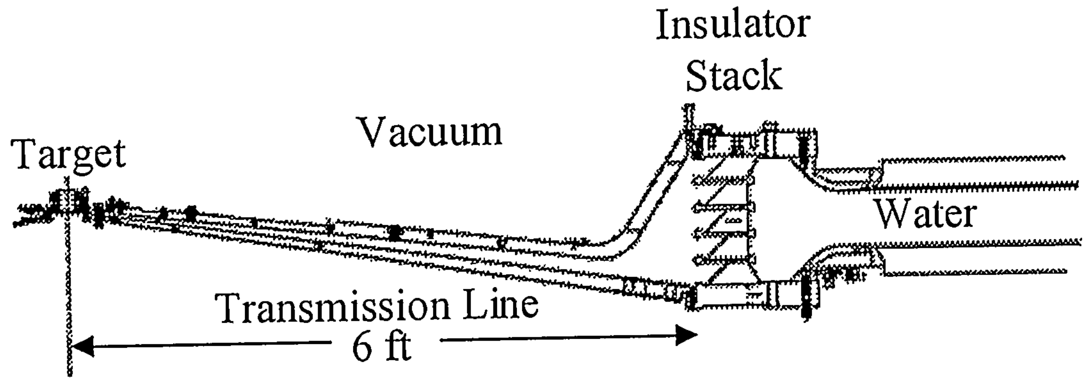
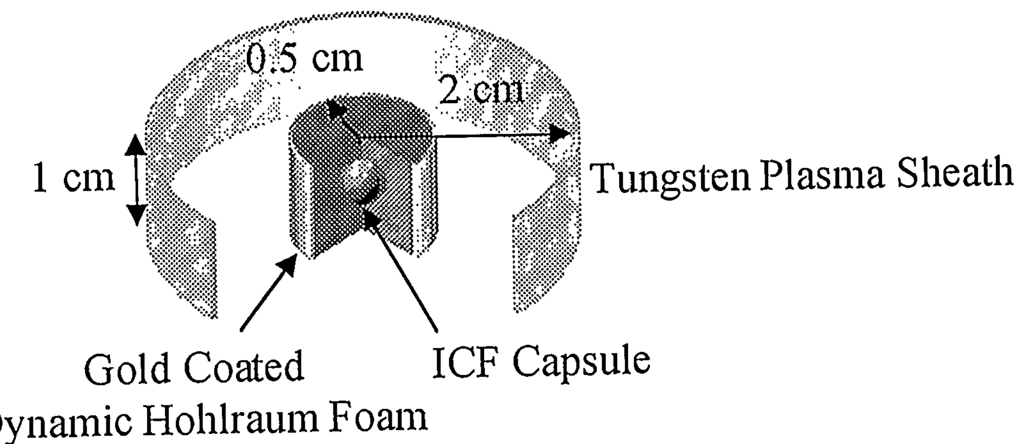


(a)



(b)

Fig. 1: (a) Cross-sectional view of a portion of the transmission line on the Z machine. The transmission line is radially symmetric with an outer radius of 6 ft. (b) A schematic of a dynamic hohlraum. The tungsten plasma sheath collides with the foam and heats the interior to form a hohlraum with a dynamically changing radius.

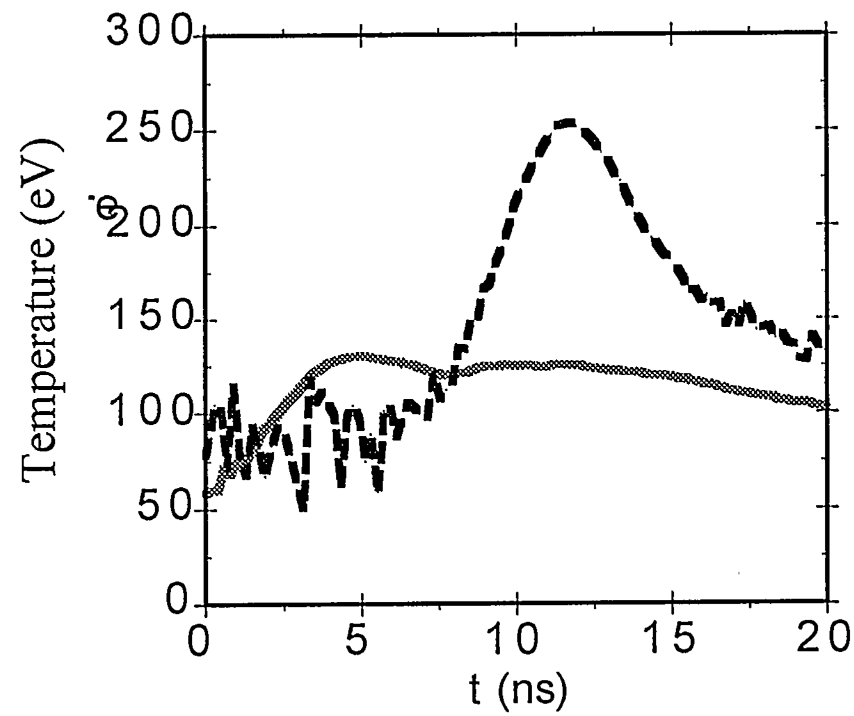


Fig. 2: X-ray emission versus time inside two target types on Z. By combining the two loads, a temporally-shaped temperature profile may be applied to a fuel capsule. The low-temperature foot pulse (solid) is created with a high-Z annular foil while the main pulse (dotted) is created by a z-pinch colliding with an on-axis foam.

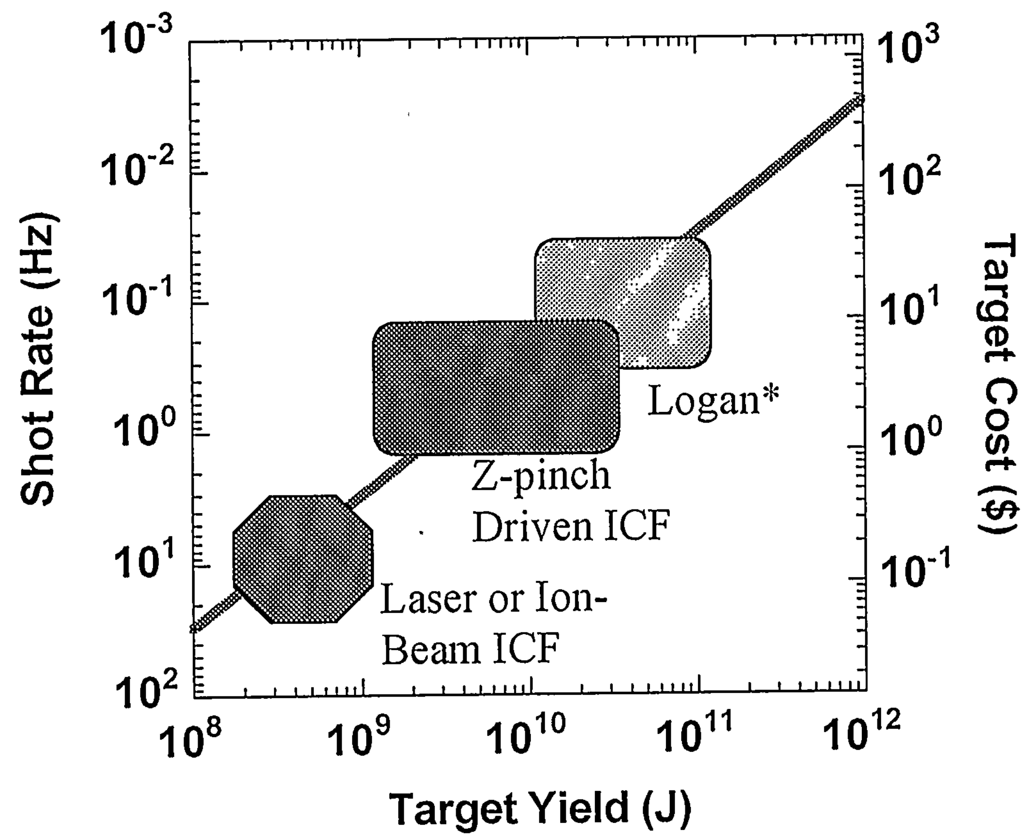


Fig. 3: Shot rate and target cost for a 1 GW_e pulsed fusion reactor. The shot rate is based on a 33% thermal to electrical energy conversion. The target cost is based on a $0.05/kWh energy cost and the assumption that target expenditures make up 10% of the overall cost of electricity.

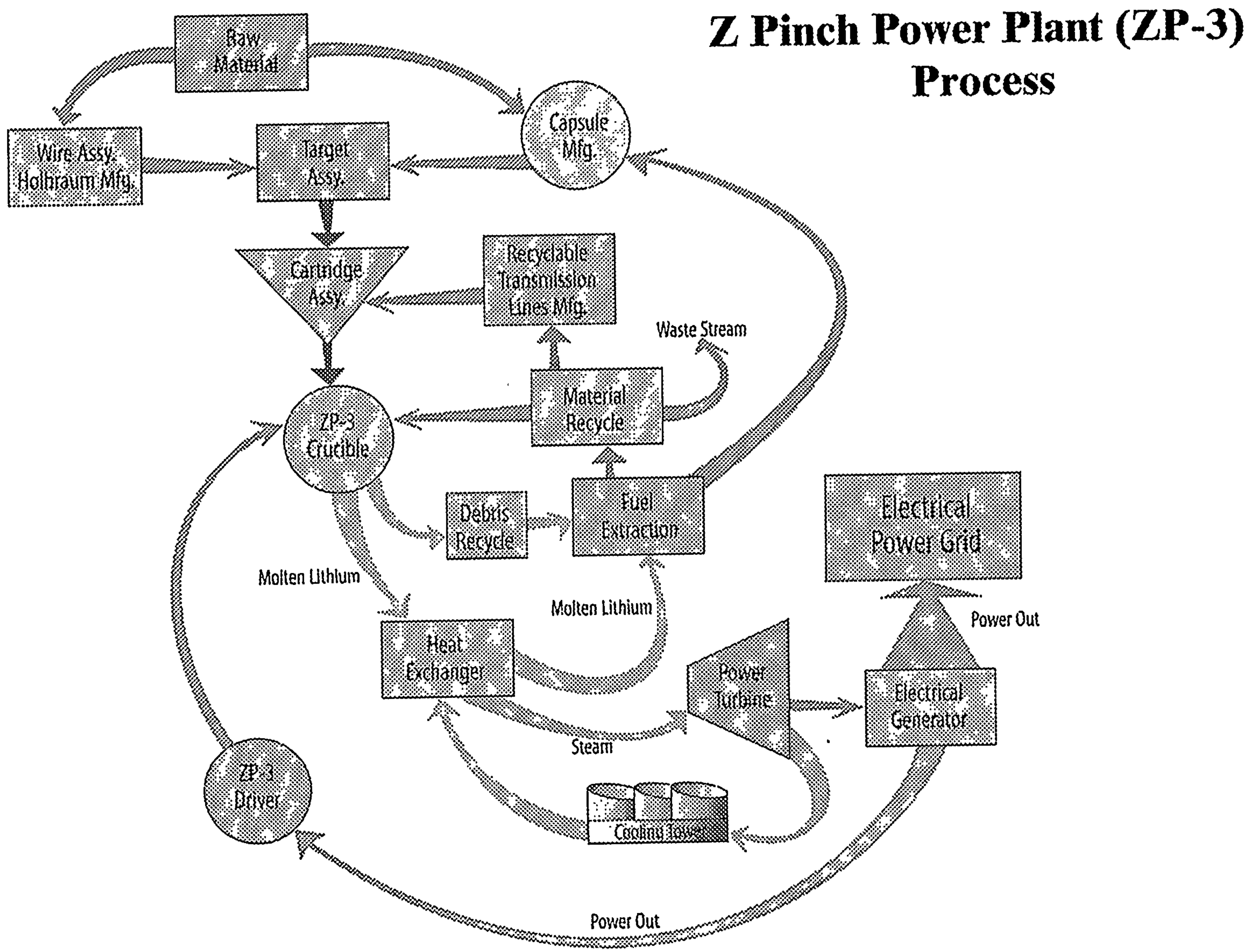


Fig. 4. Conceptualization of the various components and flows in the Z Pinch Power Plant (ZP-3) at steady-state operation.

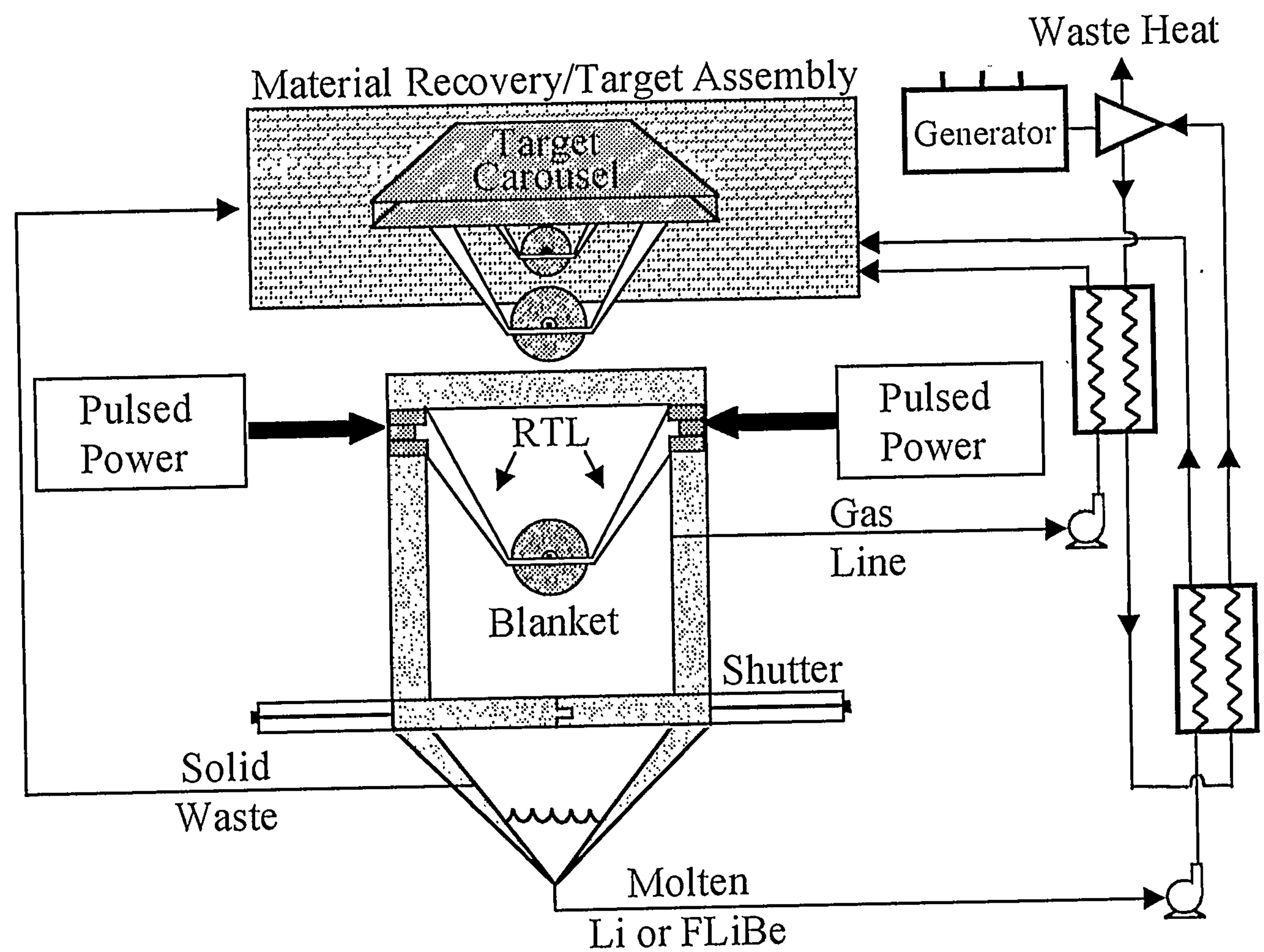


Fig. 5: Schematic of a single ZP-3 module. The pre-pumped, pre-aligned RTL and integrated target hardware is lowered into the chamber before each shot along with the blanket structure. The target energy yield vaporizes or liquefies part of the RTL and blanket which is pumped out of the chamber and circulated through a water or gas heat exchanger. After heat exchange, the material is sent through a tritium extractor and material separator and then recast into a blanket or RTL for a future shot.

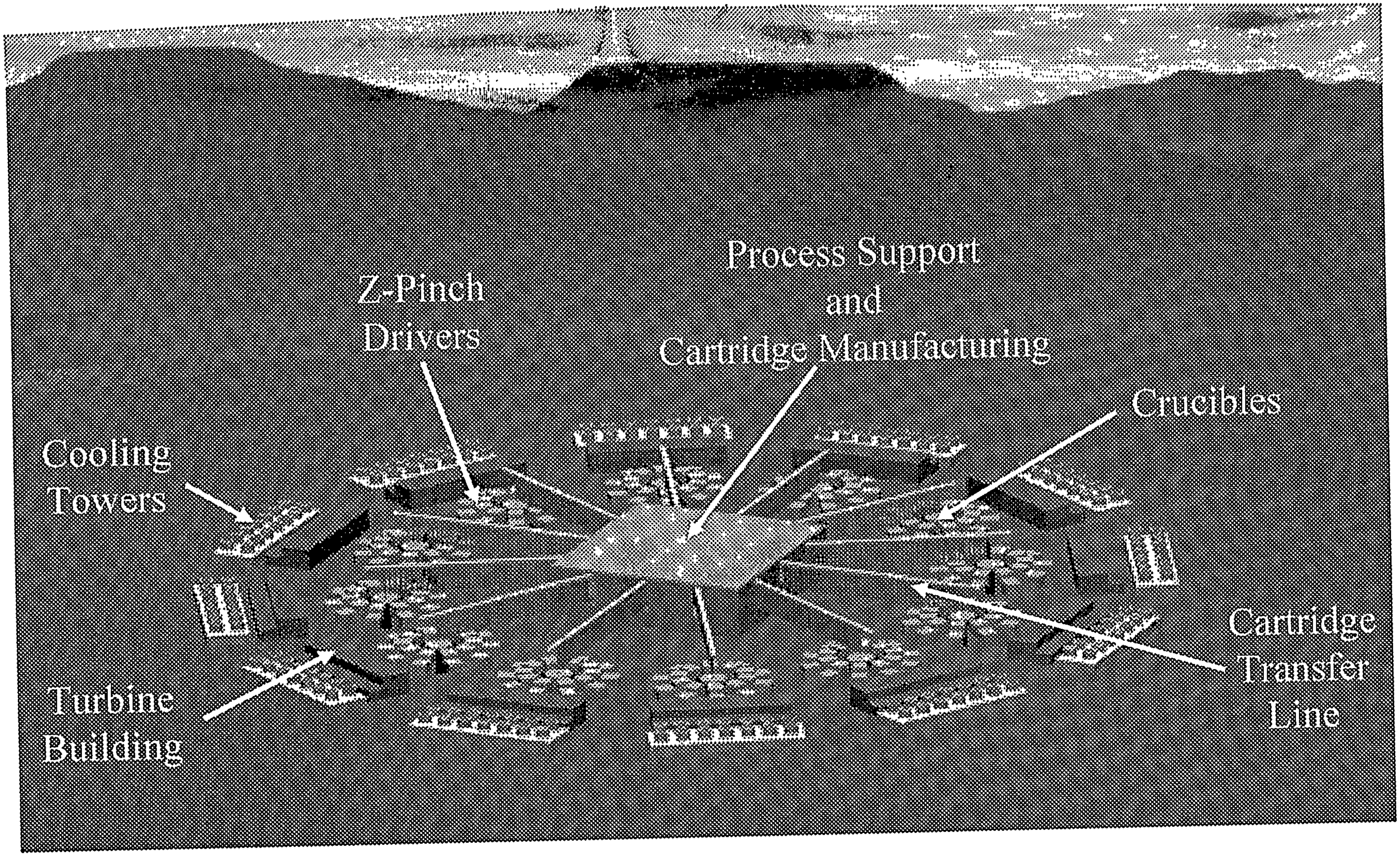
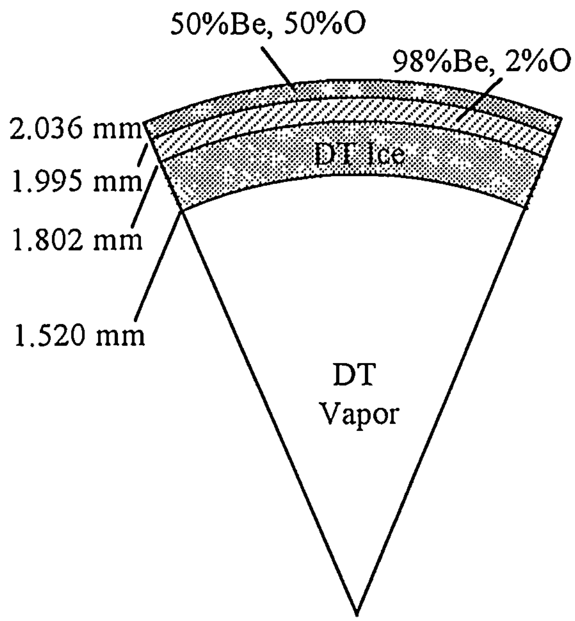
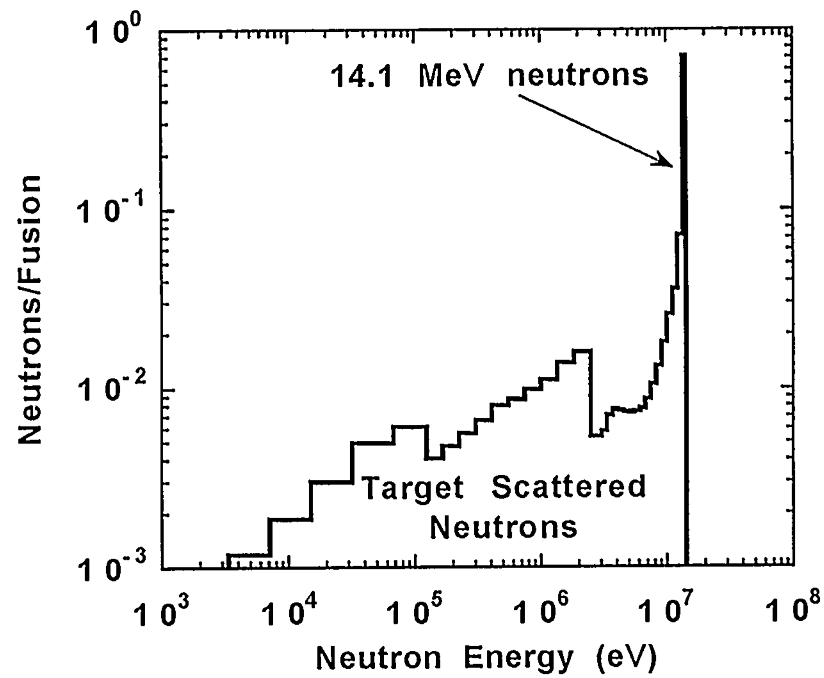


Fig. 6: An artist's rendition of the Z-Pinch Power Plant (ZP-3). In this version, the complex contains 12 modules which all share a single material collection and re-manufacturing center. Cartridges (blanket, RTL, and target assemblies) are cast from recycled material and distributed to each module while the post-shot material is pumped back to the manufacturing center for recycling.



(a)



(b)

Fig. 7: (a) Radial build of a 320 MJ ICF target for the X-1 advanced radiation facility as designed by the BUCKY 1-D radiation-hydrodynamics code. (b) Target softened neutron spectrum as modeled by the ONEDANT 1-D deterministic neutron transport code given the fusion neutron production rate calculated by BUCKY. 14.1 MeV characteristic neutrons make up ~70% of the total neutron spectrum while collided neutrons make up the additional ~30%.

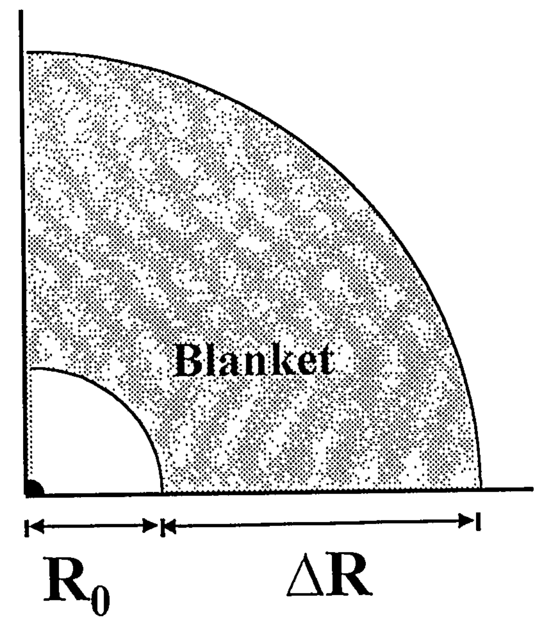


Fig. 8: Cross-sectional view of the spherical blanket models used in COG. Blanket materials which were modeled include natural Li, Pb-17Li, and FLiBe.

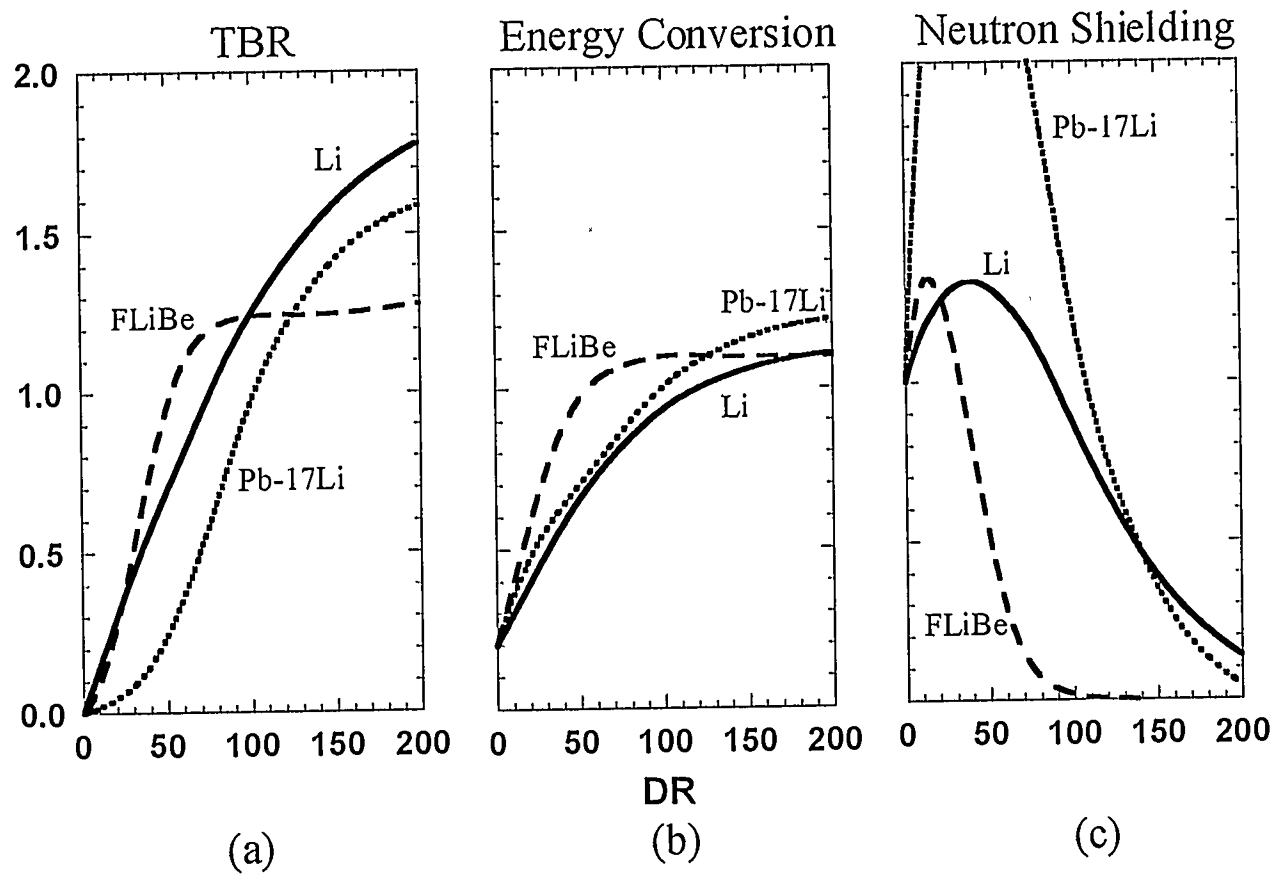


Fig. 9: COG calculations of the (a) tritium breeding ratio, (b) energy conversion ratio, and (c) neutron shielding effectiveness for natural lithium (solid), FLiBe (dashed), and Pb-17Li (dotted). The energy conversion ratio is calculated by taking the ratio of the energy deposited in the blanket to the total released fusion energy while the shielding effectiveness is determined by taking the ratio of the COG calculated fluence at the blanket edge to the fluence assuming no blanket structure

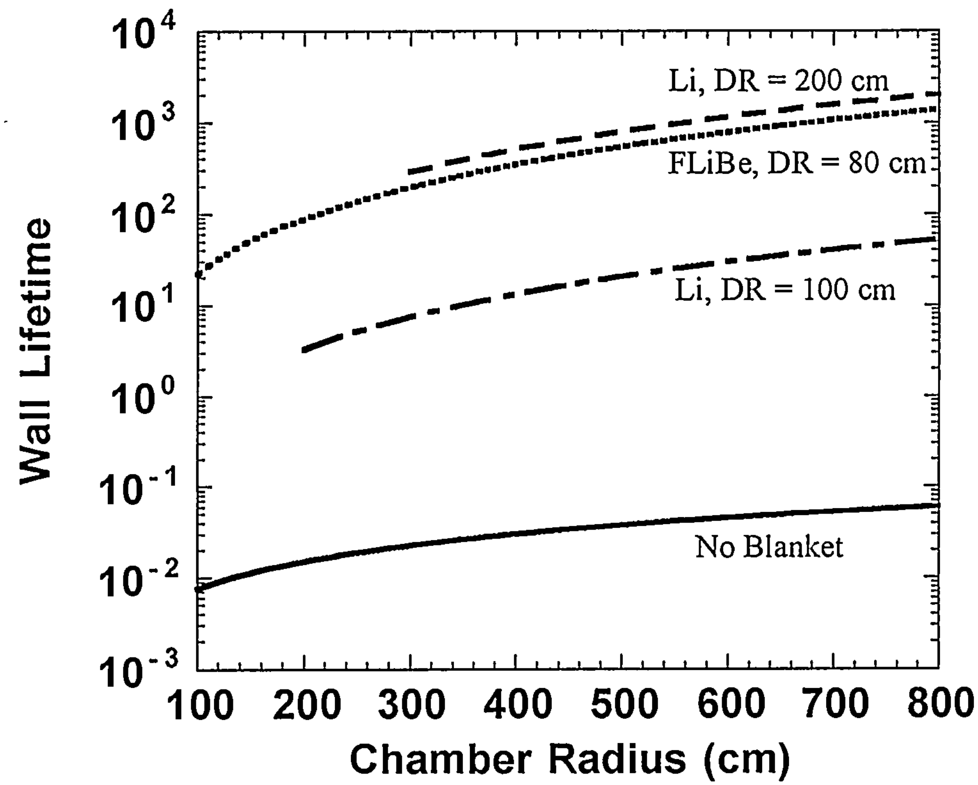


Fig. 10: Radiation damage induced chamber wall lifetimes as a function of chamber radius for no blanket (solid), Li blankets with shell thicknesses of 100 cm (dot dashed) and 200 cm (dashed), and a FLiBe blanket with a thickness of 80 cm (dotted). The wall lifetimes are calculated assuming wall-vaporization due to x-ray interaction is not significant and that the wall lifetime is limited by an overall neutron fluence of $2.0*10^{27}$ MeV/cm^2.

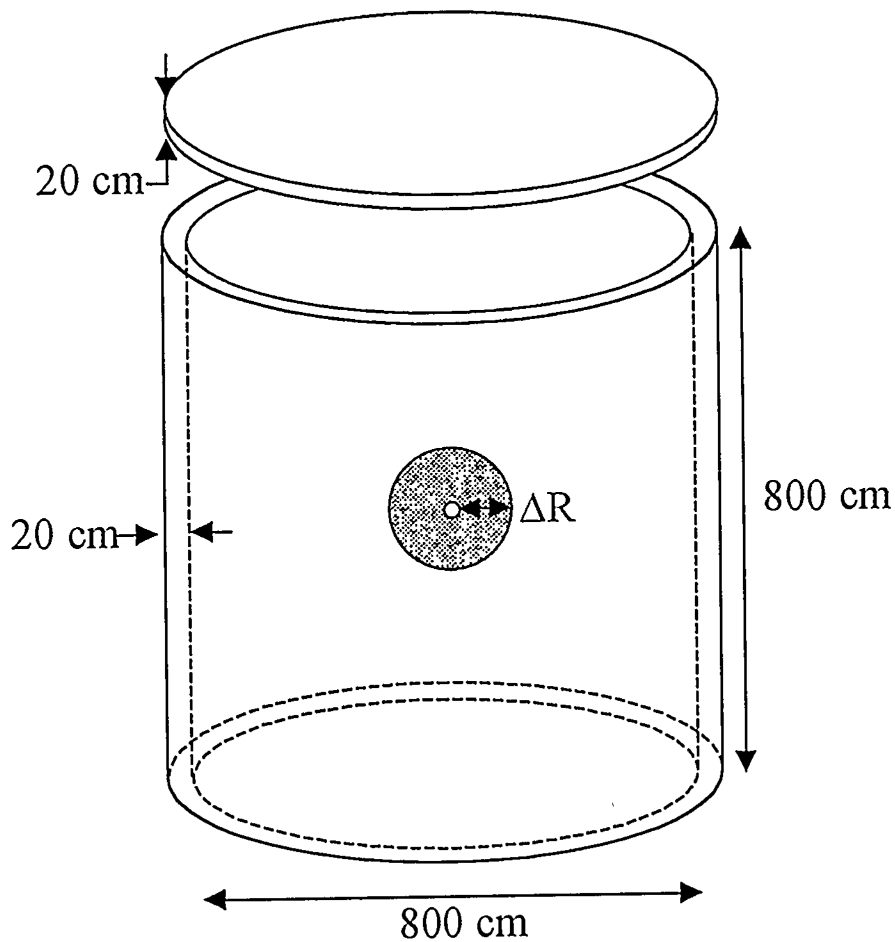


Fig. 11: Chamber geometry in COG and ALARA models for determining the chamber wall activation. SS316, 6061-T6 Al, and 2.25Cr-1Mo steel were considered as wall materials.

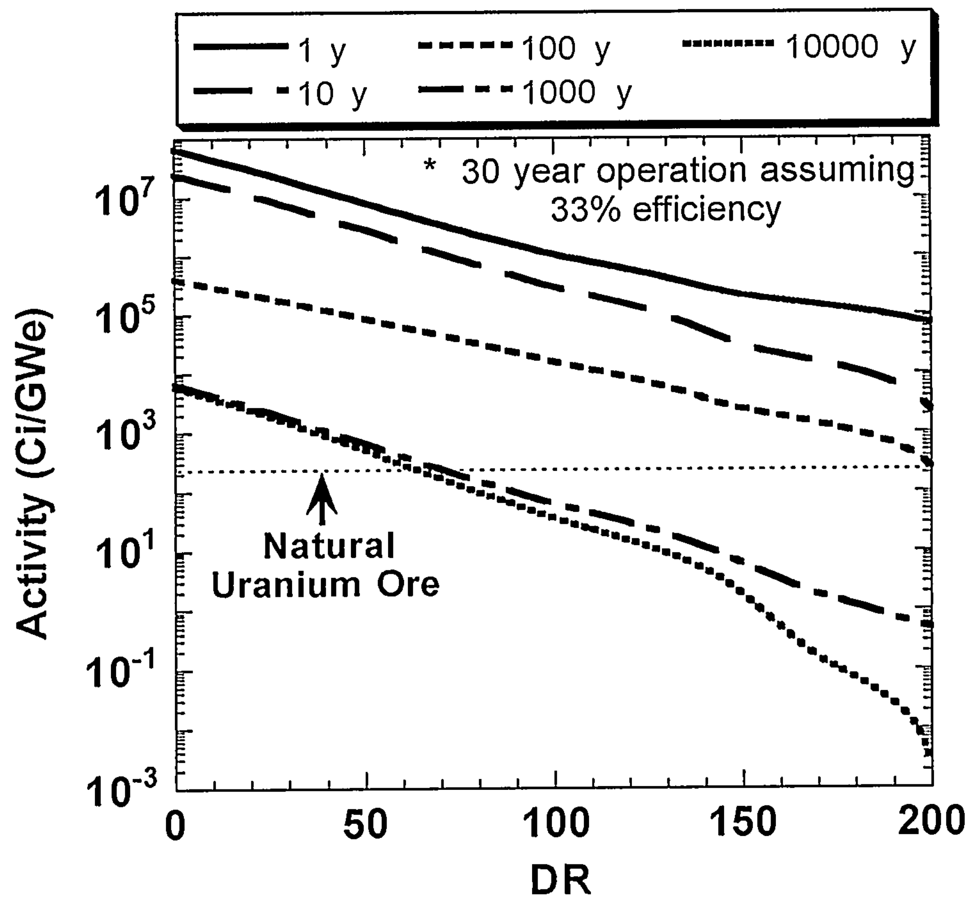


Fig. 12: ALARA calculated chamber wall activation as a function of blanket thickness assuming a cylindrical 6061-T6 Al chamber with a 20 cm wall thickness, a radius of 400 cm, a total height of 800 cm, and 20 cm thick Al end caps. The chamber activity is shown for cool-down periods of 1 year (solid), 10 years (large dashed), 100 years (small dashed), 1000 years (dot dashed), and 10000 years (dotted). These calculations do not include the shielding effect of the RTL structure.

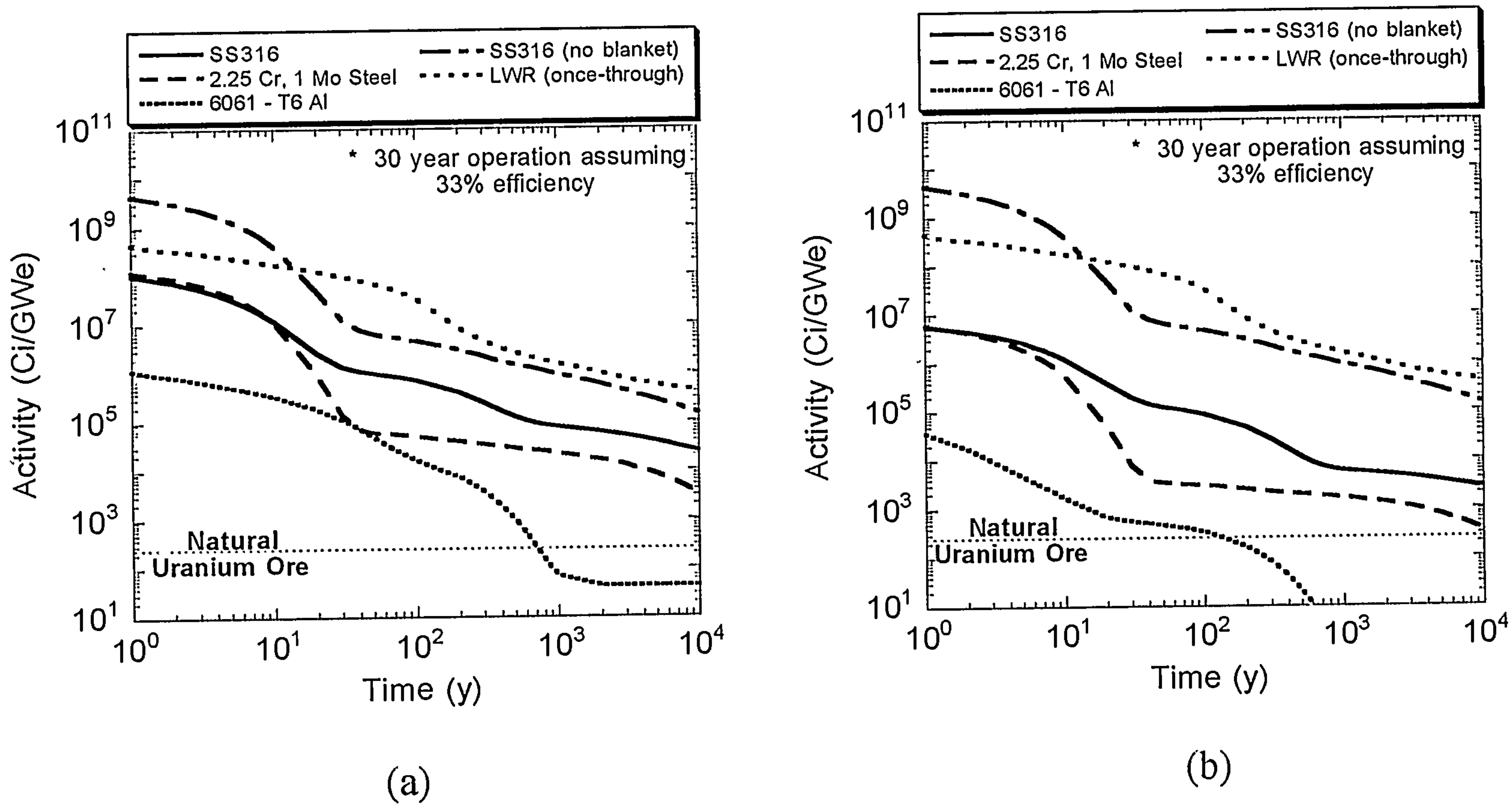


Fig. 13: Chamber wall activation as function of time after shutdown following a 30 year ZP-3 lifetime assuming a 30 GJ target yield and a 0.1 Hz shot rate for an (a) 95 cm thick Li blanket and a (b) 80 cm thick FLiBe blanket. Chamber wall materials which were studied include SS316 (solid), 2.25 Cr -1 Mo Steel (dashed), and 6061 - T6 Al (small dotted). Also plotted are the activity of an SS316 chamber with no blanket (dot-dashed), and a once-through LWR (large dotted). These calculations were conducted with the ALARA activation code assuming continuous operation with no downtime for maintenance.

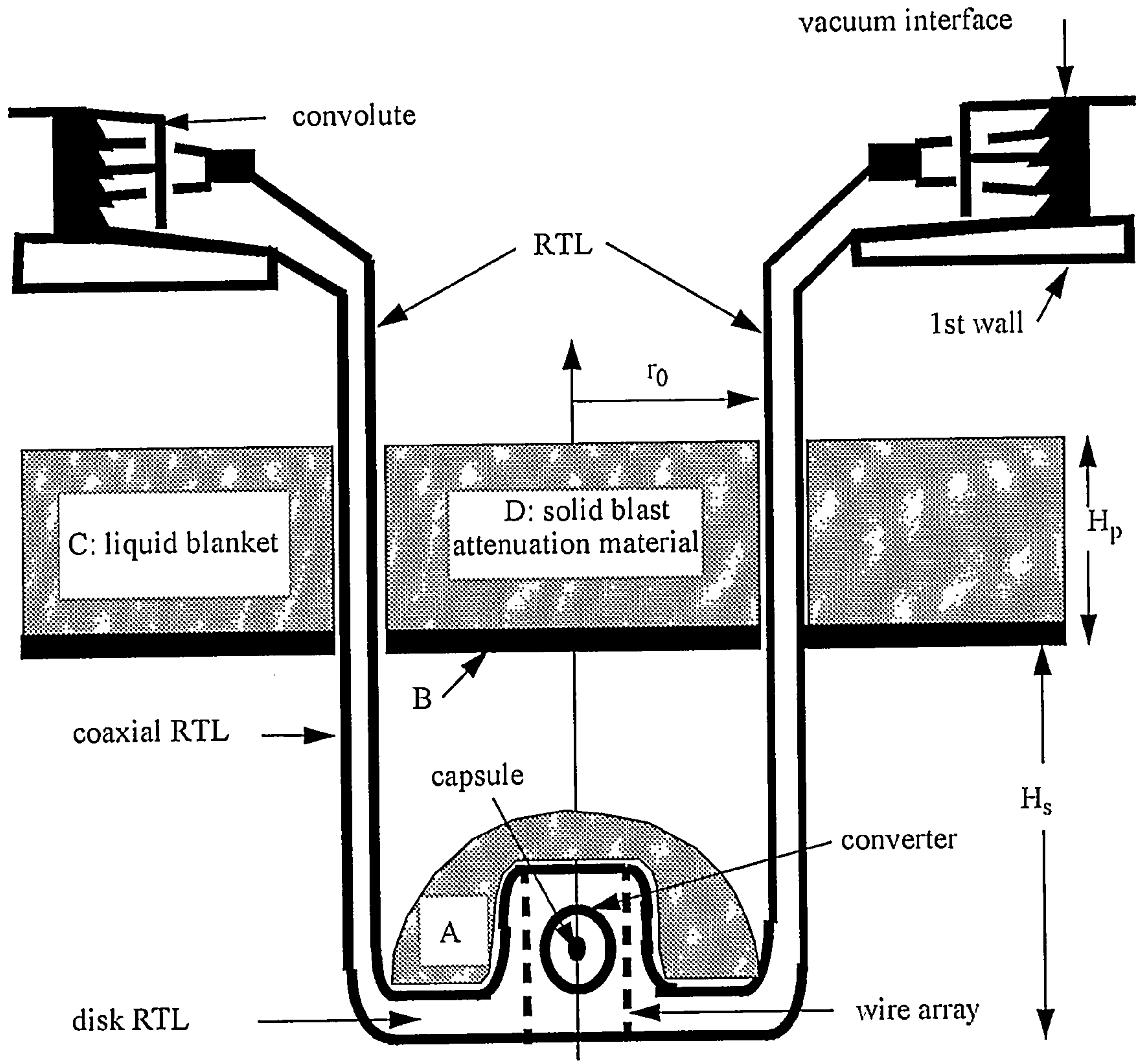


Fig. 14. Sketch of RTL, vacuum interface and liquid or solid blanket materials.

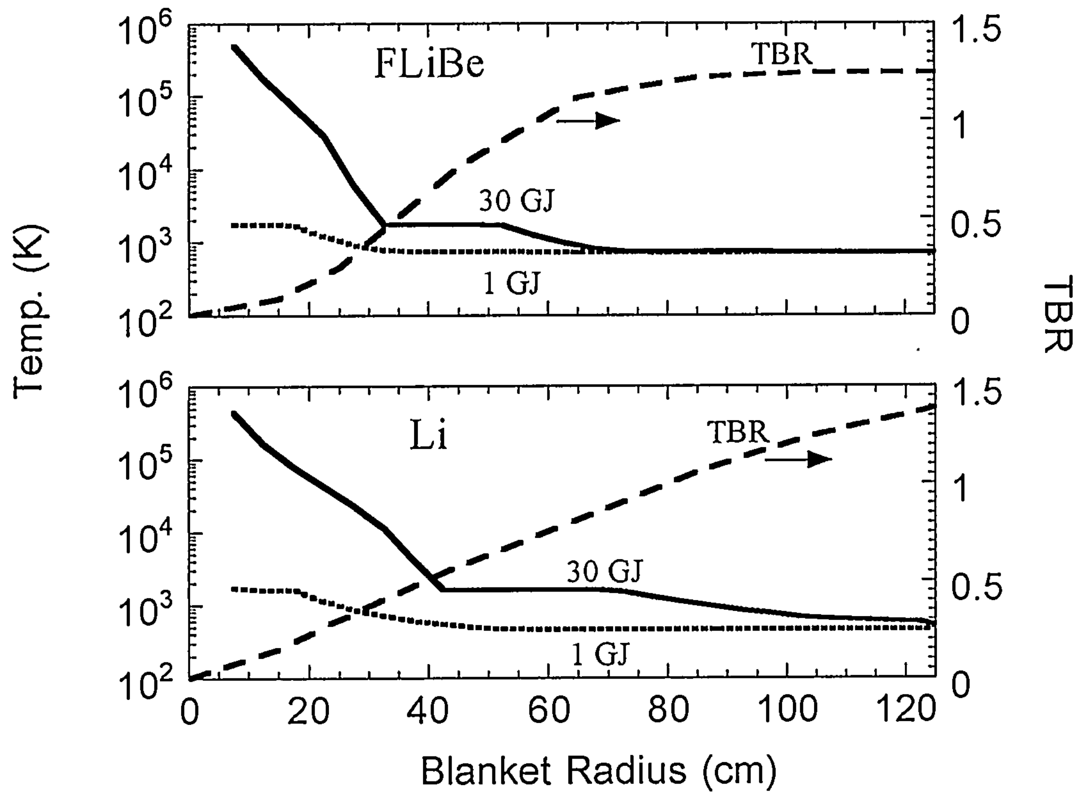


Fig. 15: Temperature as a function of radius from machine center for both natural lithium and FLiBe breeder blankets for a 30 GJ target yield (solid) and a 1 GJ target yield (dotted). Both blankets were assumed to have an initial temperature just 50 K below their respective melting points (350 K for Li and 742 K for FLiBe). The tritium breeding ratios are also plotted as a function of radius (dashed) for reference.