

Accelerator-Driven Thorium Cycle Power Reactor: Design and Performance Calculations

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Abstract – A systematic design is being explored for an accelerator-driven thorium-cycle power reactor. The core for a GW_{th} reactor is driven by a pattern of seven 800 MeV proton beams in a hexagonal pattern. A flux-coupled stack of isochronous cyclotrons capable of producing all 7 beams has been described in an earlier paper. The isotope inventory, power distribution, and neutron gain have been simulated through an eight year period of reactor operation. Primary features of efficiency, gain stability, and reliability are reported.

I. INTRODUCTION

Accelerator-driven thorium cycle (ADTC) reactors have been proposed many times for the future of nuclear fission power [1]. A high-energy proton beam is used to produce fast neutrons by spallation, the fast neutrons transmute ^{232}Th to produce ^{233}U , and the ^{233}U undergoes stimulated fission. The benefits are obvious: thorium is an abundant fuel; the reactor operates as a subcritical amplifier; the fast neutrons used for transmutation of ^{232}Th into ^{233}U also consume long-lived waste isotopes; the reactor produces almost no ^{235}U or ^{238}Pu and so largely decouples from proliferation concerns.

Two things have been missing to make a practical ADTC system design. First, no accelerator technology was available that could deliver ~15 MW of 800 MeV proton beam – the power needed to drive a GW_{th} core. Second, a central drive beam cannot efficiently drive a cylindrical core, because the fission products formed in the inner regions of the core would absorb neutrons sufficiently to turn off the outer regions early in the fuel life-time, so that only a fraction of the fuel could be consumed.

We overcame both difficulties by using a flux-coupled stack of seven isochronous cyclotrons [2], sharing a single flux return, which produce 7 independent beams of 800 MeV protons. Each cyclotron operates with parameters similar to those of the PSI cyclotron in Switzerland, which has operated with high reliability for the past 30 years. Together they provide the necessary beam power to drive a GW_{th} core with criticality $k \sim 0.975$.

The seven beams are delivered to the core in hexagonal (6 on 1) pattern, as shown in Figure 1.

Each proton drive beam is surrounded by a spallation region containing Pb moderator. The fuel is a mixture of ^{232}Th and ^{233}U , encased in pins of radius 3.5 mm with steel cladding of thickness 0.35 mm. Molten lead permeates the pin bundles as well as the spallation zone and outer shielding region, and provides convective heat transfer to heat exchangers located well above the core.

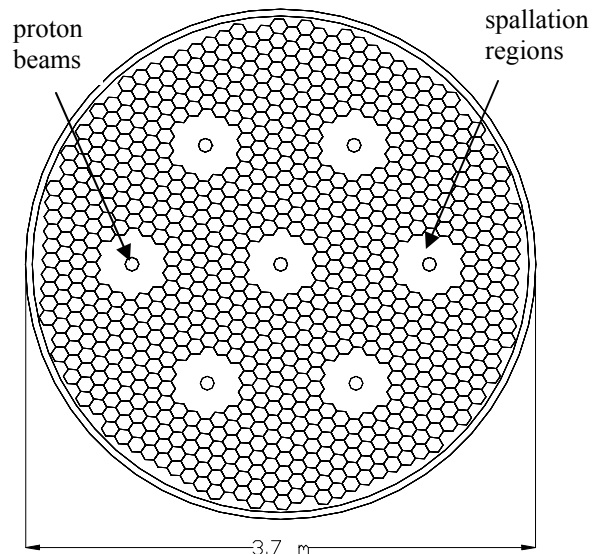


Figure 1. Top view of ADTC core with 7 drive beams.

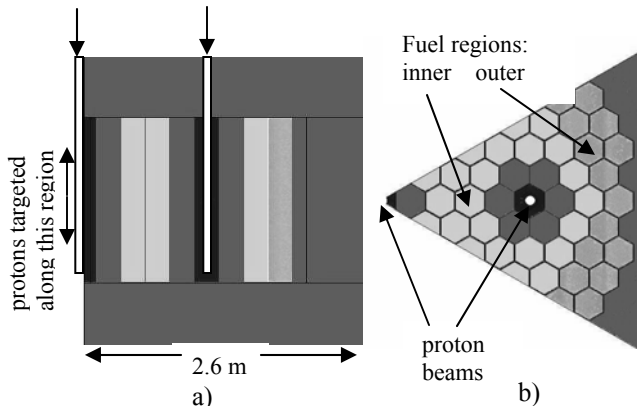


Figure 2. Cross section of core from center out: a) side view; b) top view.

II. CORE DESIGN

The reactor core design must be optimized in order to maximize benefits from the distribution of drive beams and also to minimize the required proton beam power. The arrangement of proton beams, spallation zones, and fuel bundles is shown in Figure 2 in a cross-section from the center out along a plane intersecting an outer beam tube.

Table 1 summarizes the parameters of the core geometry. The fuel region is subdivided into an inner region and an outer region. The inner and outer fuel regions have the same size fuel assemblies (and pin size), but with different pitch between fuel pins (different total number of fuel pins). An oxide fuel composition of 90% ^{232}Th /10% ^{233}U is assumed. The initial ^{233}U component would be recovered from recycling the spent fuel from a previous ADTC core. It speeds the start-up of the core, since otherwise an initial inventory of ^{233}U would have to be developed by transmutation.

Each proton beam traverses a beam tube into the core and is transversely modulated so that protons strike the side walls of the beam tube uniformly along a 50 cm length, providing a line source for spallation. The proton beam energy was chosen to be 800 MeV, providing a fast neutron yield of ~ 20 n/p.

Molten lead is used as moderator, heat transfer medium, and shielding/reflection of neutrons at the core boundaries. The outer radius of the lead is chosen large enough so that it does not affect neutronics in the core.

The neutronics in the core was simulated using MCNPX [3] and SCALE4.3 [4]. In order to optimize the fuel bundle size, we calculated the power output and fuel mass as a function of bundle size, keeping the pin size and total number of bundles constant. The results are shown in Figure 3. We chose a bundle size of 18 cm to be optimum.

Table 1. Main Parameters of the Core

Overall lead volume		
radius	2.6	m
height	8	m
mass	2000	ton
Overall core dimensions		
Radius	1.5	m
Height	1.5	m
Fuel bundles		
Fuel pin		
composition	90% Th^{232} , 10% U^{233}	
radius	0.355	cm
cladding thickness	0.055	cm
Bundle size (flat to flat)	18	cm
Inner fuel region		
number of bundles	6x20	
pins per bundle	271	
Outer fuel region		
number of bundles	6x14	
pins per bundle	331	
Starting fuel inventory:		
^{232}Th	21,137	kg
^{233}U	2,221	kg

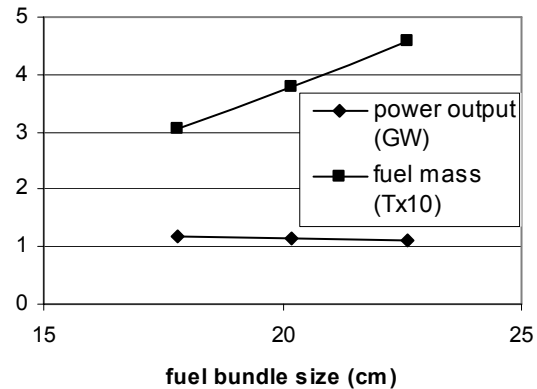


Figure 3. Optimization of fuel bundle for power output and fuel mass.

III. CRITICALITY AND OUTPUT POWER

We designed the core for a criticality $k = 0.975$. Dependence of k on core temperature and on moderator density is a very important issue for stability. We calculated k at $T = 700$ C and at room temperature, taking into account the temperature dependence of the density of lead moderator and also the Doppler broadening of neutron capture resonances.

The density ρ [g/cm^3] of molten lead as a function of T [$^{\circ}\text{C}$] is (from Ref. 5):

$$\rho(T) = 10.71 - .00139(T - 213)$$

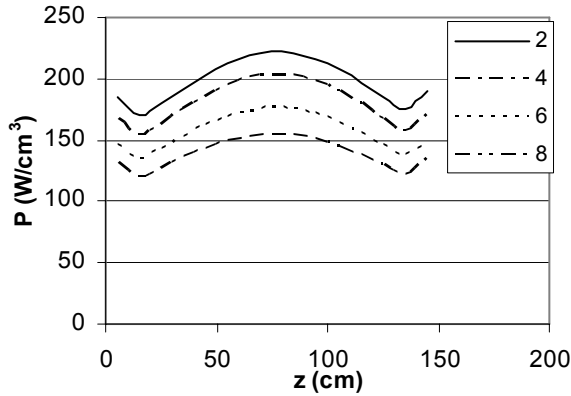


Figure 4. Power distribution in the core at startup.

We calculated the effect of Doppler broadening using the cross section libraries provided with MCNPX. We found no net effect on k with temperature. These libraries are given for room temperature, however, and we do not have high-temperature data for ^{232}Th . In order to check the validity of the extrapolation techniques in MCNPX, we also calculated Δk over the same temperature range for ^{235}U using two temperature-specific cross section libraries, ENDF60C ($T=294^\circ\text{K}$) and ENDF62MT ($T=900^\circ\text{K}$). The calculated values were $k(294^\circ\text{K}) = 0.8147(4)$ and $k(900^\circ\text{K}) = 0.8144(5)$, a change $\Delta k = -.0003$.

The most appropriate way to calculate the change in criticality with temperature is to evaluate the source-dependent criticality k_{src} , including the contribution from the spallation of drive protons.

We calculated criticality in this manner for 5 cases. Case 1 uses a lead density of 10.33 gr/cm^3 ($T = 700^\circ\text{C}$). Case 2 uses a lead density of 10.56 gr/cm^3 ($T = 400^\circ\text{C}$). Next three cases were calculated changing density of the lead only in some regions keeping temperature equal 400°C . Results show that deposited energy decreases if only lead density in the fuel region was changed. The corresponding relative absorption and escape rate of neutrons were 0.6089 and 0.6085 for the two cases.

In order to calculate k_{src} we follow Ref. 4⁶:

$$k_{\text{src}} = \frac{\nu \cdot N_{(n,f)} + 2 \cdot N_{(n,2n)} + 3 \cdot N_{(n,3n)}}{N_{\text{total}}},$$

where $N_{(n,f)}$, $N_{(n,2n)}$, $N_{(n,3n)}$ and N_{total} are the numbers of neutrons that gave fission, $(n,2n)$, $(n,3n)$ reactions and total number of neutrons respectively. The first term in the nominator is

$$\nu \cdot N_{(n,f)} = \int dE \cdot \phi(E) \cdot (\sigma_{(n,f)} + 2 \cdot \sigma_{(n,2nf)} + 3 \cdot \sigma_{(n,3nf)}).$$

The second and third terms are

$$N_{(n,2n)} = \int dE \cdot \phi(E) \cdot \sigma_{(n,2n)},$$

$$N_{(n,3n)} = \int dE \cdot \phi(E) \cdot \sigma_{(n,3n)}.$$

We obtain $k_{\text{src}}(700^\circ\text{C}) = 0.9760$, $k_{\text{src}}(400^\circ\text{C}) = 0.9771$, yielding $\Delta k = -0.0011$ for $\Delta T = 300^\circ\text{C}$.

Table 2. Temperature and pressure distribution in the core

	Inner region	Outer region	
Pin pitch	1.09	0.99	cm
Pin heat/length	203	135	W/cm
ΔT top-bottom	200	200	C
Pb velocity	1.02	0.96	cm/s
Pb viscosity	.0043	.0048	Poise
Pressure drop	27	38	kPa

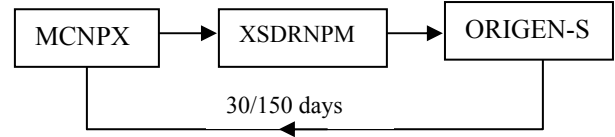


Figure 5. Calculation sequence for life cycle simulation.

By contrast, calculations of an earlier design of an ADTC core [7], using a single proton drive beam on the core axis, yielded a much larger $\Delta k = -0.01$ over the same temperature range. The reduction of Δk is a consequence of the more uniform proton source geometry in our design: any given location in the core fuel is supplied with neutrons from several neighboring proton drive beams, and the percolation distance is greatly reduced.

IV. POWER DISTRIBUTION AND TEMPERATURE

Figure 4 shows the power distribution in the core at the beginning of the fuel cycle. The curves show the power density vs. z in succeeding rounds of fuel assemblies: round #2 are the innermost assemblies in Figure 2, round #4 are the 4th from center, round #8 are the next-to-outermost assemblies, etc.

Figure 4 illustrates an important benefit of distributed proton drive: it produces a very flat power distribution in both z and r within the core.

At time $t=0$ we will be producing in average 126 W/cm^3 or $\sim 45 \text{ W/g}$. The maximum temperature gradient within in the fuel and the cladding of a pin (assuming conductivity 45 W/mK) is $\Delta T = 318^\circ\text{C}$.

Table 2 shows the calculated temperature distribution and hydraulic pressure drop in the inner and outer regions of the core. The molten lead cooling provides realistic convective heat transfer to a heat exchanger that can be located $\sim 4 \text{ m}$ above the top of the core. This has the benefits that the tubing and fittings of the heat exchanger are not subject to extreme neutron flux, and tritium production in the water of a steam exchanger is eliminated.

V. POWER AND CRITICALITY THROUGH CORE LIFETIME

The isotope inventory, power distribution, and criticality were modeled through the life cycle of the core, following the simulation sequence shown in Figure 5. Each cycle of calculations represents a period of operation. A constant drive proton current of 2 mA/beam is

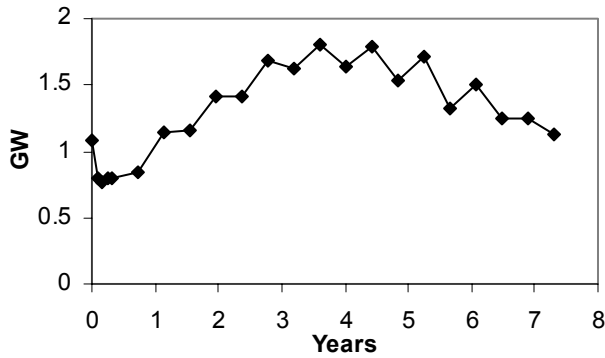


Figure 6. Power vs. time through core life cycle.

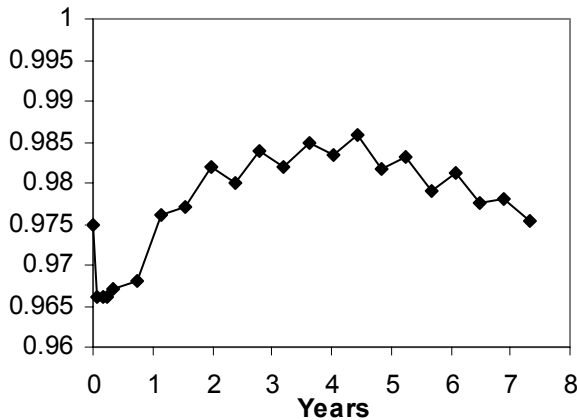


Figure 7. Criticality vs. time through core life cycle.

assumed. First the neutron spectrum is generated in MCNPX, then updated cross-sections are calculated using XSDRNPM, and finally the isotope inventories are calculated using ORIGEN-S. Then the cycle is repeated. For initial steps (while the isotopic balance is equilibrating) the sequence is calculated on a 30-day interval. Thereafter a 150-day interval is used.

Figure 6 shows the power vs. time through the life cycle of the core. The slight oscillation is an artifact of the 27-day half-life of ^{233}Pa which paces the transmutation sequence $^{232}\text{Th}+n\rightarrow^{233}\text{Th}\rightarrow^{233}\text{Pa}+\beta$, $^{233}\text{Pa}\rightarrow^{233}\text{U}+\beta$, $^{233}\text{U}+n\rightarrow\text{fission}$. Each 150-day time step produces an overshoot of the change in 'true' ^{233}U inventory, with consequent error in power output.

The core begins the fuel cycle with $\sim 800 \text{ MW}_{\text{th}}$ output, then increases power to a peak of $\sim 1.6 \text{ GW}_{\text{th}}$ at 4 years, then decreases gradually for a total fuel cycle life of ~ 10 years. The power could of course be leveled by adjusting the proton drive current through the life cycle.

Figure 7 shows the criticality k as a function of time through the core life cycle. k is calculated as the eigenvalue solution to the neutron matrix.

Figure 8. Depletion of a) ^{232}Th ; b) ^{233}U .

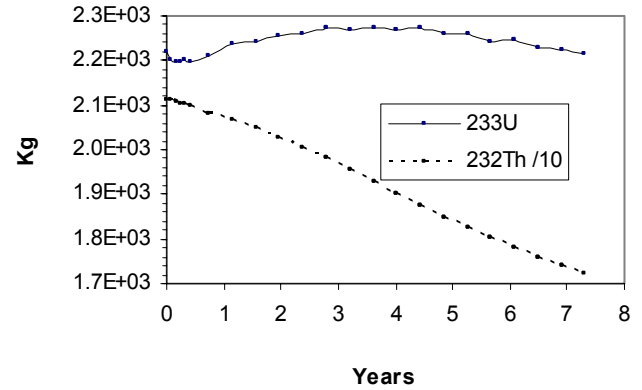
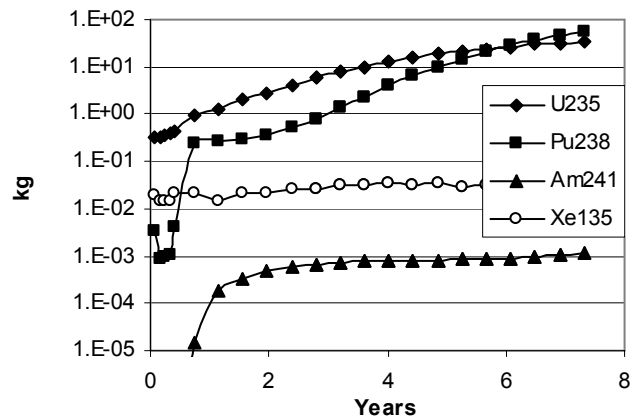


Figure 9. Inventory of ^{235}U , ^{238}Pu , ^{241}Am , and ^{135}Xe .



In calculating the above simulations, we ignored the effect of the structural elements (pin cladding, beam tubes, bundle supports). Adding the effect of the structural elements reduces k by $\Delta k = -0.013$, which can be corrected readily by adjustment of the pin geometry.

VI. DEPLETION CALCULATIONS

Figure 8 shows the inventories of ^{232}Th and ^{233}U in the core as a function of time through the core life. Figure 9 shows the inventories of bomb-capable isotopes ^{235}U and ^{238}Pu . Note that the total inventory never exceeds $\sim 50 \text{ kg}$, a factor of >100 less than that in critical reactors using uranium or plutonium fuel.

Also shown in Figure 9 are the inventories of ^{241}Am , a principal contributor to the long-lived waste problem from thermal reactors. Its inventory is $\sim 10^4$ less than that produced from thermal reactors. Finally the inventory of ^{135}Xe is shown. It has a giant capture resonance for fast neutrons and acts as a parasite for the fast neutrons needed to drive transmutation. This inventory is less problematic for the 7-drive ADTC reactor than for earlier single-beam designs.

VII. EFFECT OF LOSING A DRIVE BEAM

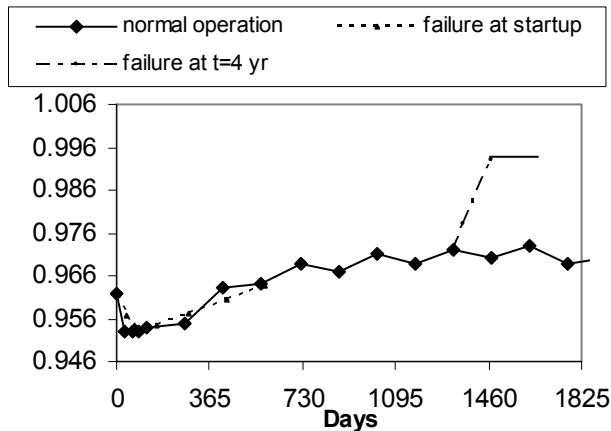


Figure 10. Criticality response following failure of one drive beam.

Reliability is a key concern in developing a realistic ADTC system. In our accelerator design, the 7 beams from the flux-coupled isochronous cyclotron stack operate independently. Many unscheduled interruptions of beam from a cyclotron require no access to the accelerator, and beam can be restored after ~5-60 minutes (*e.g.* inflector arc). Other interruptions can require access to the accelerator for periods of ~1-6 hours. Longer access is very rarely required.

One of the advantages of using the cyclotron stack is the possibility to operate the power plant even when one cyclotron fails. The best scenario for reactor management in the case of such failure of one of the proton feeds is to rearrange beams in the switchyard such that only the central target region lacks proton drive beam.

We examined the effect on reactor operation from two such scenarios. The first case considers a beam failure right at the beginning of the fuel cycle; the second case considers a failure after 4 years operation. In both cases it was assumed that reactor continued to operate until the next scheduled monthly maintenance; then it was shut down for 10 days for major repairs. The reactor is then restarted with all 7 beams delivering full current.

Figure 10 shows the criticality k as a function of time following each of these cases. Normal operation is shown by the solid line (correction has been made for the effect of structural elements). The case of failure at the start doesn't lead to any problem. The output power drops by ~15%, but that could be compensated by increasing the currents in the other six drive beams.

Failure in the middle of the fuel cycle creates a more serious effect. The inventory of ^{233}Pa in the portion of the core that is not being driven decays into ^{233}U with a half-life of 27 days, but in the fuel region close to the beam that is off no ^{233}U is consumed by fission. This leads to a local increase in fuel, and hence criticality. In the asymptotic limit that all Pa decays, k would increase by ~0.02 as shown. Since one must provide stability even in this

worst-case scenario, this consideration presents the most stringent limit on the normal-operation value of k and therefore the efficiency of the overall ADTC system. Assuming power output of $1.5 \text{ GW}_{\text{th}}$, beam power of 12 MW (7 beams, each 2 mA and 800 MeV), accelerator system efficiency ~50%, and electric conversion efficiency ~60%, the overall power cost for driving the fission reaction with an accelerator is ~3% of reactor output.

III. CONCLUSIONS

We have carried an optimization of primary core design and a simulation of life-cycle operation for a ~ GW_{th} ADTC power reactor. By taking advantage of the 7 drive beams from a flux-coupled isochronous cyclotron stack, we are able to obtain uniform power distribution, stable sub-critical operation, efficient consumption of fuel loading in a ~10 year lifetime, and extremely little production of bomb-capable isotopes and long-lived waste isotopes. In addition we are able to demonstrate the potential for the ADTC reactor to continue operating if any one drive beam fails, providing significant reliability for routine operation. The ADTC reactor appears to be an excellent candidate for future power reactor technology.

ACKNOWLEDGMENTS

This work was funded in part by the U.S. Dept. of Energy, grant # DE-FG03-95ER40924.

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