

Catalyzed deuterium fusion neutron source for the transmutation of minor actinides, production of tritium, and electrical power generation

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Abstract

The accumulated inventory of spent nuclear fuel in the United States exceeds 90,000 metric tons, with long-term radiological hazards dominated by minor actinides that require isolation for thousands of years. This study details a possible solution to these challenges that also provides significant electrical power. The method employs a fusion system operating in catalyzed deuterium (Cat-D) mode producing the high neutron flux required for this task. It is based on the Compact Field Reversed Configuration Fusion Reactor (CFR2), where the plasma is sequentially compressed to fusion conditions remote from the burn chamber and then translated into the chamber surrounded by the fission blanket. A Monte Carlo neutronics analysis of a subcritical actinide transmutation blanket and the design was optimized. The optimized blanket employs FLiNaK eutectic fluoride salt enriched to 90% ⁶Li and containing 17.53 mol% transuranic fluorides as the combined coolant, moderator, and fissile medium. The design achieves $k_{eff}=0.9695$ with a blanket volume of 13.4 m³. The FRC is pulsed at 4Hz with a tritium fraction of 4%, yielding a neutron flux of 4×10^{18} n/s. This fission system burns 683 kg/yr of TRU, produces 9.6 kg/yr of tritium and 1,431 MWth, requiring only 17.5 MWe to power the CFR2.

1. Introduction

1.1 The Spent Nuclear Fuel Crisis and the Case for Transmutation

The accumulated inventory of spent nuclear fuel (SNF) in the United States now exceeds 90,000 metric tons, growing at approximately 2,000 MT per year as the existing light water reactor (LWR) fleet continues to operate. The long-term radiological hazard of this material is dominated not by the bulk uranium matrix but by the minor actinide (MA) constituents, principally neptunium-237 (²³⁷Np), americium-241 (²⁴¹Am), americium-243 (²⁴³Am), and curium isotopes (e.g. ²⁴⁴Cm), which persist for tens of thousands of years. The 10,000-year regulatory isolation requirement imposed on permanent geological repositories is almost entirely driven by these radionuclides. Their destruction by neutron-induced fission would reduce the required isolation period to less than 300

years and convert the waste into short-lived fission products that are already well characterized in the existing fuel cycle.

In addition, the global civil tritium inventory, produced almost entirely as a by-product of CANDU heavy water reactor operation, has been assessed at approximately 25–30 kg and is projected to decline as that fleet approaches retirement [1]. A single 1 GWth fusion plant would consume 28 kg of tritium in just 182 days. The inventory for a fusion reactor start-up needs to be roughly this amount and would severely constrain the rate at which a D-T fusion fleet could be established. This dilemma is independent of the breeding performance of the fleet once fully operating.

Transmutation via neutron irradiation is not a novel concept in nuclear engineering. Accelerator-driven subcritical systems (ADS) [2] have been studied for decades as a means of providing an external neutron source to drive a subcritical fission blanket loaded with TRU material at an effective multiplication factor k_{eff} below unity, providing inherent safety against criticality excursions while maintaining an adequate neutron flux for actinide destruction [3]. The Herculean scale of a viable ADS, which requires a GeV proton linac with hundreds of milliamperes of average current, is likely to cost billions of dollars. This alone has prevented the realization of this approach. Critical fast reactors [4,5] offer an alternative but impose severe constraints on k_{eff} management, as the fissile inventory changes with burnup and requires fuel fabrication infrastructure that is not yet available for MA-bearing fuels.

Fusion neutron sources offer a third approach [6-11]. A fusion plasma employing deuterium (D) and tritium (T) fuel produces a D-T reaction, resulting in a 14.1 MeV neutron with a D-D side reaction producing a 2.45 MeV neutron from a compact volumetric source with no fissile material in the driver, making this aspect of the system intrinsically incapable of being supercritical. When powering a fission blanket, the fusion source strength is independently controllable, providing a straightforward mechanism for k_{eff} and power regulation. Historically, the central challenge for a fusion-fission hybrid has been the same as that for fusion power: achieving an adequate neutron source strength at an economically acceptable driver power. The present study addresses this challenge using the Compact FRC Fusion Reactor (CFR2), the neutron source physics and engineering basis of which were established in a companion paper [12].

1.2 Why the Field-Reversed Configuration

The field-reversed configuration (FRC) is a compact, simply connected magnetic confinement geometry in which a cylindrical plasma is confined solely by its own diamagnetic poloidal field against an externally applied axial magnetic field. The intrinsic physical and engineering attributes of the FRC make it uniquely well-suited as a fusion neutron source for a transmutation blanket system, fundamentally distinguishing it from toroidal fusion devices.

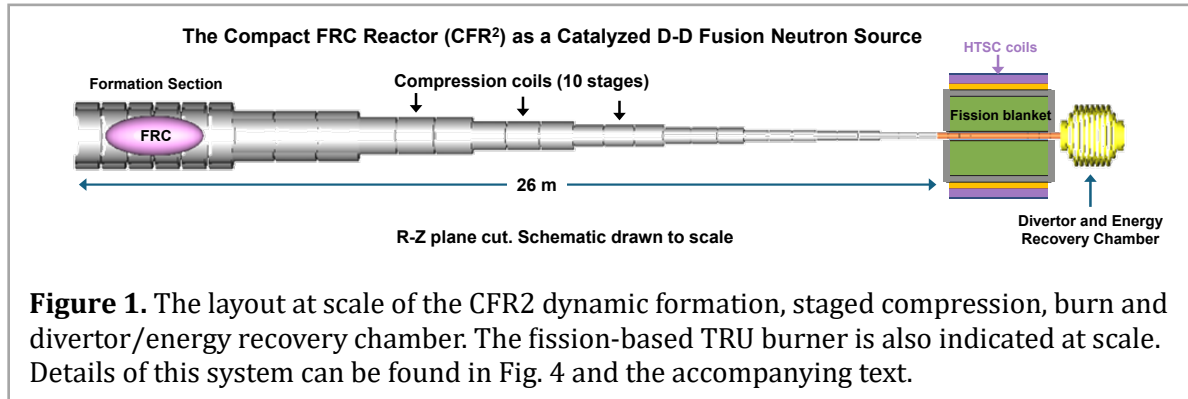
The fusion reactor engineering advantages enumerated in the companion paper [12] are particularly relevant to the blanket coupling challenges of fusion-fission hybrids. First, the simply connected vacuum system requires no toroidal field coils, making periodic blanket replacement or repair feasible, which is a practical necessity for a transmutation system that must process several hundred kilograms of actinide-bearing salt per year. Second, solenoidal confinement coils are structurally simpler than toroidal coil sets and are thus capable of producing the highest attainable magnetic fields, enabling a compact scale that is central to the CFR2 concept. Third, the linear reactor geometry provides efficient impurity control through plasma outflow along open-field lines to the remote divertor regions outside the reactor chamber. Fourth, and most consequential for the hybrid application, the FRC plasmoid is formed and heated entirely outside the fusion burn chamber. This remote formation capability means that the blanket can more completely surround the burn region without interruption for large port access for neutral beams, RF heating conduits, and divertor structures, maximizing the neutron capture volume and eliminating the geometric inefficiency inherent in toroidal fusion devices that force blanket designs into awkwardly

segmented geometries, thereby reducing coverage and complicating the neutron economy. Fifth, with pulsed operation, the power output can track the load demand, maintain gain, and simplify both tritium recovery and impurity removal. In addition, employing solid-state switching for FRC formation and compressional heating provides high-efficiency energy recovery ($\eta \sim 80\%$). It also provides a direct, continuously adjustable mechanism for controlling the neutron output and, hence, the blanket power, without the need for control rods or reactivity management systems in the fission blanket. Sixth, the naturally high plasma β of the FRC allows operation at maximum plasma pressure for a given confining field, resulting in a high-power density that makes compact and affordable reactor modules feasible.

The fusion gain of the FRC scales as $G \sim \phi_p \cdot B_e^2$, where ϕ_p is the FRC poloidal flux and B_e is the confining axial magnetic field [Ref. (12), eq. 8b]. Critically, this gain is essentially independent of the physical scale of the reactor; thus, a compact device with a burn chamber inner radius of only 6 cm can achieve the same fusion gain as a much larger system, provided that the poloidal flux and confining field targets are met. This scale independence eliminates the fundamental economic disadvantage that has historically accompanied large-volume fusion devices.

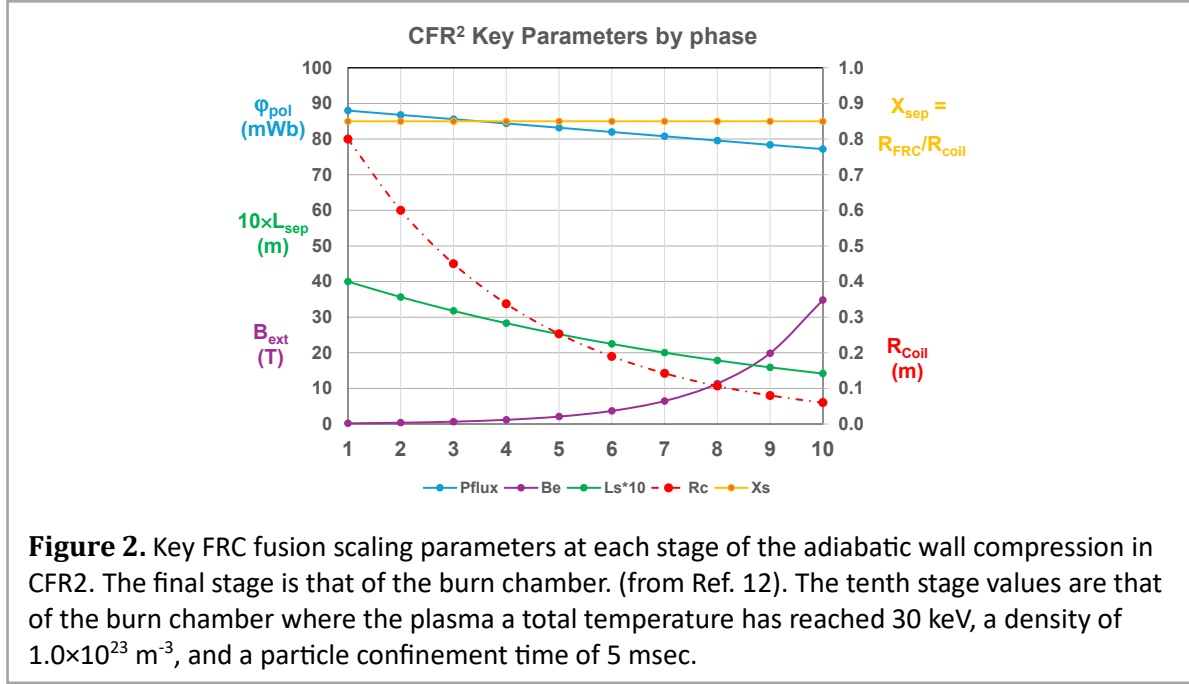
1.3 The Staged Compression Approach and the CFR2 Concept

The CFR2 concept achieves a combination of high poloidal flux and high confining field required for fusion gain through a series of radial adiabatic wall compression stages, as described in Ref. 12. The four operational phases are briefly summarized here, as they define the neutron source



characteristics upon which the transmutation analysis in this study is based.

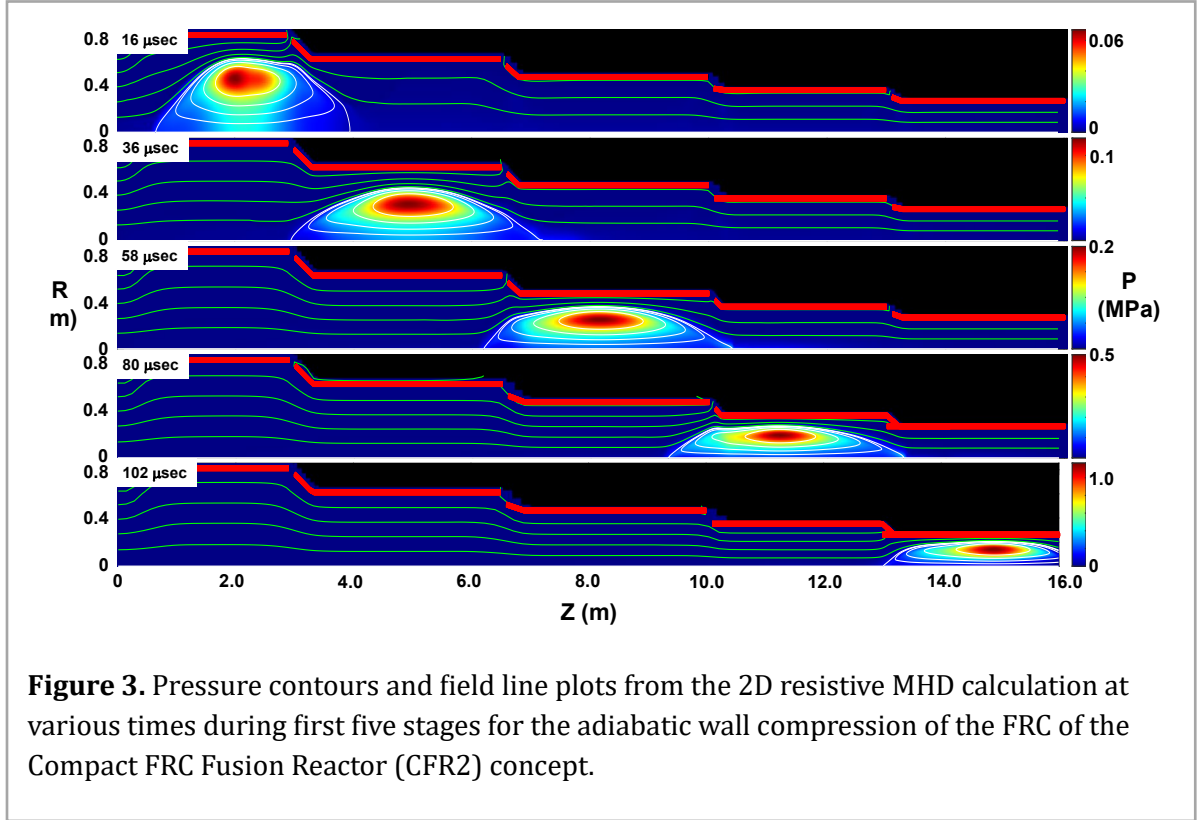
In Phase 1, a large FRC plasmoid is formed in a remote chamber with a radius of approximately 0.8 m. By applying a sufficiently large electric field at the vacuum wall ($E_\theta \geq 20$ kV/m), a fast axial field reversal is produced that induces a strong azimuthal current thereby generating a large poloidal flux $\phi_p > 80$ mWb. At a sufficiently low deuterium fill pressure (~ 0.13 Pa), the initial turbulence in achieving the equilibrium pressure profile where $P = P(\psi)$, produces the initial ion temperature $T_i > 1$ keV that is required for FRC kinetic stability at high ϕ_p . This relatively large formation radius is necessary because the area-normalized lift-off flux is empirically limited to approximately 0.1 T. Reaching the target flux requires a chamber cross-sectional area that cannot be directly achieved at the desired reactor scale as it is much smaller. To achieve the desired reduction in scale, the FRC is translated through a series of compression coils. To initiate the translation the FRC is accelerated to a subsonic axial velocity of 100–150 km/s using a dynamic formation technique, in which the sequential coil activation in the formation chamber achieves the requisite velocity. The complete system layout, to scale, spanning approximately 26 m from the formation chamber to the energy recovery chamber, is shown in Fig 1.



In Phase 2, the FRC is translated through a series of approximately ten cylindrical coils of decreasing radii, each of sufficient length to ensure that the FRC undergoes only isentropic and adiabatic radial wall compression at each stage without simultaneous axial acceleration. The key FRC parameters at each of the ten compression stages are shown in Fig. 2. This staged approach is the key innovation that distinguishes CFR2 from previous supersonic merging concepts [13]. By achieving compression without net acceleration while maintaining the FRC separatrix-to-coil radius ratio $x_s = r_s/r_c$ at 0.75–0.85 throughout, destructive flux loss from MHD instabilities is suppressed. The FRC stability parameter s/ϵ [12] remains below unity through all compression stages, and the high x_s value substantially reduces the separatrix plasma pressure relative to the bulk FRC pressure, suppressing the turbulent particle and energy loss that limits the FRC performance at conventional x_s values of 0.35–0.55. Adiabatic scaling laws for wall compression [14] confirm that this process satisfies the stability condition $s/\epsilon < 1$ throughout, in contrast to flux compression, which degrades the stability as the FRC is compressed.

The FRC undergoes a slight compression during the translation through each coil to facilitate entry into the next coil. This process is repeated from one coil to the next with decreasing coil radii, but increasingly stronger magnetic fields having roughly the same vacuum flux filling each coil thus producing what is essentially a wall compression.

A resistive 2D MHD code was used to better understand the behavior of the FRC and optimize the process. Owing to the large change in the radial scale and the length of the simulation region, it was necessary to divide the calculations into two or three separate calculations, reflecting roughly half of the staged compression, with a restart of the calculation using the FRC parameters at the end of the previous calculation. Figure 3 shows the results of the first five of the ten stages. As shown in Fig. 3, the FRC maintains an equilibrium where the pressure contours align with the flux surfaces. Only a single monolithic coil was used per stage for radial compression. This still allows the FRC to coast at a relatively constant speed as it passes into progressively smaller coils. In this way the process becomes essentially both an isentropic and adiabatic wall compression.



Pellet fueling during the late compression stages adjusts the plasma inventory to place the ion temperature in the optimal D-T fusion range of 10–15 keV [12].

In Phase 3, the FRC is finally translated into the burn chamber, which is a 0.12 m diameter, 3-6 m long cylindrical, flux-conserving, thick-walled tube. Imbedded in this reactor chamber is a vacuum axial magnetic field ranging from 7 to 9 T. For the hybrid this remains similar to the strictly fusion-based CFR2; however, the source coils are now produced by a set of High Temperature Super Conducting (HTSC) coils that are now external to the chamber as well as the blanket to avoid interference with the fusion neutron's passage into the fission blanket (see Fig. 1).

The insertion of the FRC into the reactor chamber compresses this vacuum field to approximately 30-35 T. The ratio $x_s = 0.85$ is maintained in the burn chamber, as this substantially lowers the FRC edge pressure and thus greatly reduces both particle and thermal losses during the 2-5 ms transit burn. In contrast to the D-T fusion reactor discussed in Ref. 12, the wall does not need to amplify the neutron yield with a beryllium-based alloy to ensure a tritium breeding ratio greater than one. As will be seen, this is not an issue here. Thus, the primary attributes for the chamber wall, in addition to good conductivity, are high tensile strength combined with high temperature tolerance and ductility. A Tungsten-Rhenium alloy would serve this purpose well, as it is also more transparent to fusion neutrons. It should be noted that the first wall, unlike toroidal devices, can be readily replaced, and its service life can be as short as one year allowing for much higher neutron fluence.

Operating as a pure D-T fusion reactor, as treated in the companion paper, the resulting fusion yield for the 3.5 MJ FRC plasmoid is ~50 MJ per pulse, with a neutron energy yield of approximately 40 MJ per pulse, yielding 1.8×10^{19} n per pulse. The fusion gain predicted by the modified LSX empirical scaling extended to large x_s is $G \sim 9$ –21 depending on the flux confinement assumptions. In the catalyzed D-D mode employed in the fission hybrid, and described in Section 1.4, the reaction

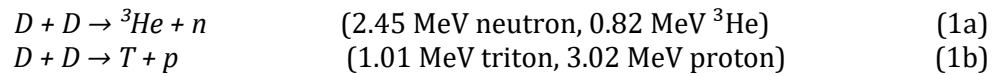
rate is lower by roughly the ratio of the D-D to D-T reactivities and the yield per pulse is correspondingly smaller. At the 1% tritium Cat-D equilibrium the corresponding figures are 4.0×10^{17} neutrons per pulse with a mean energy of 8.275 MeV. At 10 Hz the neutron rate is 4.0×10^{18} n/s. At the optimized operating point of 4% plasma tritium fraction and 4 Hz, the 3.5 MJ FRC plasmoid also yields 4.0×10^{18} n/s but at a mean energy of 11.77 MeV. The energy yield per pulse is 1.89 MJ of neutron energy and 2.36 MJ of total fusion energy from the 3.5 MJ FRC. The fusion energy gain of the source at the optimized point is therefore $Q \approx 0.67$.

This is not a limitation of the configuration. Fusion and fission are complementary in a manner that this concept exploits directly. Fusion is neutron-rich but energy-poor. The D-T reaction releases 17.6 MeV, of which 14.1 MeV is carried by a neutron, but the reaction is difficult to sustain, and the energy gain of any near-term device is modest. Fission is the opposite, being energy-rich at approximately 200 MeV per event but requiring a neutron economy that a critical assembly can sustain only by operating at or above criticality. Coupling a subcritical fission blanket to a fusion neutron source uses each for what it does well. The fusion device is required to supply neutrons, not energy. At the optimized operating point, the fusion power is 9.4 MWth against 1,471 MWth of fission power, or 0.6% of the total thermal output, and the driver draws 17.5 MWe, 2.5% of the gross electrical output. Whether the source operates at $Q = 0.67$ or $Q = 6.7$ changes the plant power balance by less than one percent, and the requirement placed on the fusion device is correspondingly greatly relaxed relative to a pure fusion power plant.

In Phase 4, the spent FRC is ejected into a remote, large-radius energy recovery chamber, where the remnant plasma energy is recovered by inductive coupling to flux-conserving liner coils at an efficiency of approximately 38%, and the recovered energy is returned to the solid-state driver circuit.

1.4 Catalyzed D-D Operation and Neutron Source Characteristics

For transmutation applications, CFR2 operates in a Catalyzed Deuterium (Cat-D) fuel mode, in which the only externally supplied fuel is deuterium. This choice is central to the design and warrants careful explanation. The D-D reaction proceeds via two branches with approximately equal probabilities:



In a deuterium plasma at ion temperatures of 10-15 keV, the D-T reaction rate exceeds the DD rate by a factor of approximately 50. The D-D reaction produces either a neutron or triton, each with a 50% probability of occurrence. These tritons are then recycled into subsequent burns, thereby increasing the fraction of T in later plasmas. The T builds up to a steady rate until a 1% T equilibrium is reached, where the D-T neutron output equals the T produced by the D-D reactions, that is, $n_{DT} = n_{DD}$. To partially offset the factor of 50 reduced D-D reaction rate, the pulse repetition rate is increased to 10 Hz for the baseline Cat-D operation so that the total neutron production rate is 4×10^{18} n/s, with the neutron population consisting of 50% D-T neutrons at 14.1 MeV and 50% D-D neutrons at 2.45 MeV. The flux-weighted mean neutron energy at the baseline Cat-D operating point is $E_{src} = (14.1 \text{ MeV}) \cdot f_{DT} + (2.45 \text{ MeV}) \cdot f_{DD} = 8.275 \text{ MeV}$ (where f_{DT} and f_{DD} are the fractional neutron yields from each branch). As will be seen later, the mean source energy E_{src} is the single parameter that determines the blanket fission power.

The Cat-D mode offers several specific advantages for transmutation applications beyond the general FRC attributes enumerated above. Probably the most significant factor is that deuterium is abundant and inexpensive and does not require an external tritium supply infrastructure. Second, both the 2.45 MeV D-D and 14.1 MeV D-T neutrons are optimally energetic for driving fast fission in the actinide-bearing blanket before thermalization. As will be more thoroughly discussed in Sections 4 and 6, each additional MeV of neutron source energy produces approximately 65.5 MW

of additional fission power in the optimized blanket. This coefficient is normalized to a fixed total source rate of 4×10^{18} n/s. Because fixed-source transport is exactly linear in source strength, the blanket power at any other source rate S follows as $P(S, E_{\text{src}}) = (S/S_0) \cdot f(E_{\text{src}})$ with $S_0 = 4 \times 10^{18}$ n/s. The mean source energy is raised by increasing the plasma tritium fraction, which shifts the neutron population from the 2.45 MeV D-D branch toward the 14.1 MeV D-T branch, and is bounded above by $E_{\text{src}} = 13.87$ MeV at a 50% tritium fraction, which produces the peak number of 14.1 MeV neutrons in the D-T branch. Third, the Cat-D mode is naturally self-regulating in tritium inventory: the equilibrium tritium fraction is set by the balance of production and consumption, providing a stable operating baseline without active tritium feed. Fourth, as will be shown in Section 4, the fission-multiplied neutron flux in the FLiNaK blanket produces tritium at a rate vastly exceeding plasma consumption with a tritium breeding ratio (TBR), defined here as the ratio of total tritium production in the blanket to the tritium consumed in the plasma of 25.6 at the baseline operating point, thus transforming the CFR2 from a tritium consumer into the largest projected near-term source of tritium at a fusion-relevant scale.

For the optimized operating point analyzed in this study, a 4% plasma tritium fraction operating at a 4 Hz repetition rate, the CFR2 driver power consumption is only 17.5 MWe and the mean source energy $E_{\text{src}} = 11.77$ MeV, producing 1,734 1,471 MW_{th} of fission power from the subcritical blanket. The fission output from a single module is thus comparable to that of a large commercial nuclear power plant, driven by a fusion source whose electrical consumption is less than 2.5% of the projected plant's electrical output. At this operating point the blanket has an eigenvalue multiplication factor of $k_{\text{eff}} = 0.9695$, corresponding to a neutron multiplication $M = 1/(1 - k_{\text{eff}}) = 32.8$. The source multiplication inferred from the fixed-source calculation is 0.9726, giving 36.4; the distinction between these two quantities is explained in Section 6.3.

The figure of merit adopted throughout this paper is the ratio of gross electrical power produced to electrical power consumed in its generation, that is, the wall-plug efficiency of the plant. For a fusion power system this ratio is the quantity that determines viability, and achieving a value usefully above unity is the central unsolved difficulty of the field: a pure D-T plant requires a plasma gain of order 20 to 30 to reach an engineering gain of at least 5 once thermal conversion and recirculated power are accounted for. At the optimized operating point of the present system the gross electrical output is 700 MWe against 17.5 MWe consumed by the driver and cryo-plant, a wall-plug ratio of 40, obtained with a fusion source operating at a plasma gain of $Q = 0.67$. The hybrid does not overcome the recirculating power problem of pure fusion so much as remove it, since the electrical output is supplied by fission while the fusion device is required only to supply neutrons.

The ratio of fission to fusion energy, which might appear the natural figure of merit for a hybrid, is not used here. It is ambiguous as to the disposition of the fusion energy itself and the possibility of double counting within the blanket, and more importantly it is misleading as to system performance: because raising the plasma tritium fraction increases the fusion energy input faster than it increases the blanket fission output, the fission-to-fusion ratio falls with increasing tritium content while the wall-plug efficiency rises steeply over the same range. The two measures move in opposite directions, and only the latter bears on whether the plant delivers net power.

1.5 Scope of This Paper

The neutron source physics and engineering basis for the CFR2 were established in Ref. 12. This study employs a Monte Carlo neutronics analysis of a subcritical transmutation blanket system driven by the CFR2. The blanket design employs a FLiNaK eutectic fluoride salt (LiF-NaF-KF) carrying dissolved TRU fluorides as a combined coolant, neutron moderator, and fissile fuel. OpenMC Monte Carlo neutronics calculations [16] were performed for eigenvalue and fixed-source cases across a systematic sequence of blanket designs and operating conditions. A benchmark

validation against the In-Zinerator subcritical system [9] established confidence in the computational methodology before applying it to the CFR2 geometry.

The principal findings are as follows: First, the CFR2 FLiNaK+TRU blanket at $k_{\text{eff}} = 0.97$ achieves a transmutation performance comparable to that of the In-Zinerator in approximately one-seventh of the blanket volume. This is mainly due to the geometric advantage of the close proximity of the neutron source, as no large ‘standoff’ is required here. In addition, the epithermal neutron spectrum (peak at 60.6 keV) shaped by the combined moderation of LiF, NaF, and KF concentrates the flux where the minor actinide fission-to-absorption ratios are most favorable. In addition, a systematic sweep of ^6Li enrichment from 1 to 90% revealed that 90% enrichment was optimal across all design objectives simultaneously: tritium breeding, MA transmutation quality, and fission power. Approximately 80% of the transuranic mass destroyed is plutonium, whose mass-weighted fission-to-absorption ratio of 0.80 is more than twice that of the minor actinides, so that the system functions as a plutonium disposition route as well as a minor actinide transmuter. Although it seems counterintuitive, ^6Li serves as a neutron poison, thereby setting k_{eff} while simultaneously creating a large source of tritium via $^6\text{Li}(n,t)\alpha$ and shaping the spectrum by suppressing the sub-keV flux.

As will be seen in Section 6.3, OpenMC fixed-source calculations spanning both a tritium fraction and compensation sweeps for a possible degraded FRC confinement (i.e., neutron output) fall on a single linear relationship

$$P = 65.46 \cdot E_{\text{src}} + 701 \cdot MW_{\text{th}} \quad (2)$$

establishing a single operational control that enables full compensation for FRC confinement degradation through tritium enrichment with blanket power stable or rising and CFR2 driver power demand unchanged.

2. Blanket Design and Monte Carlo Methodology

2.1 System Geometry and Radial Build

The CFR2 transmutation system comprises an FRC burn chamber enclosed within a cylindrical subcritical fission blanket. The geometry is inherently well matched to the neutron source: the linear FRC burn region, approximately 3 m in length and 6 cm in radius, is coaxial with the blanket and delivers neutrons directly into the surrounding salt without interposing any structural material between the source and the fissile medium other than the fusion reactor wall. This geometric arrangement minimizes neutron leakage from the source volume and maximizes the fraction of fusion neutrons entering the active blanket.

The radial build of the CFR2 transmutation system, which proceeds outward from the CFR2 plasma axis, is shown in Fig. 4. The innermost region is the FRC plasma bore, extending from $r = 0$ to $r = 6$ cm. This is the interior of the burn chamber through which the FRC plasmoid translates during the fusion pulse. Surrounding the bore is the flux-conserving first wall, constructed from a refractory metal alloy with high tensile strength, ductility, and electrical conductivity, which are the properties necessary to withstand both the transient magnetic pressure from FRC compression (approximately 320 MPa over roughly 20% of the FRC length) and the cumulative neutron fluence over a service life that may be as short as one year, given the ready replaceability of the linear geometry. The wall thickness is 4 cm. A helium cooling gap of 3 cm follows, providing both thermal management of the first wall and isolation between the burn chamber wall and the inner surface of the fission blanket.

The active fission blanket begins at $r = 13$ cm (inner surface of the 2 mm thick tantalum structural wall) and extends to $r = 120$ cm, giving a blanket thickness of 106.8 cm and an active volume of 13.4 m^3 for the modeled axial length. The blanket medium is FLiNaK eutectic fluoride salt enriched in ^6Li and carrying dissolved transuranic (TRU) fluorides, the composition of which is

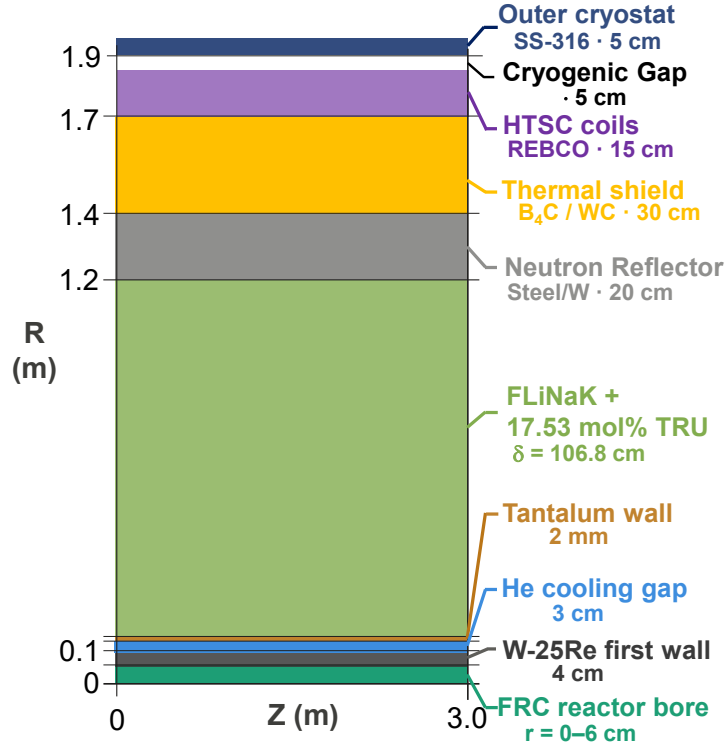


Figure 4. Radial build of the various components of the fusion-fission reactor driven by CFR2. (Z axis not to scale – see Fig. 1)

described in Section 2.2. Beyond the active blanket, a 20 cm steel-tungsten neutron reflector returns streaming neutrons to the blanket and flattens the radial flux profile near the outer boundary. A 30 cm thick, high-temperature thermal shield and a cryogenic gap separate the reflector from the high-temperature superconducting (HTSC) solenoid coil stack, which is 15 cm thick and wound from REBCO tape. The outermost element is a 5 cm stainless steel cryostat shell. The total outer radius of the system is 1.95 m.

The thermal shield located between the neutron reflector and the cryogenic region operates in an environment set by the 700°C blanket and by nuclear heating within the shield itself. The shield is specified as a 30 cm boron carbide and tungsten carbide composite in equal volume fractions, with a homogenized density of 9.06 g/cm³, which retains the required neutron attenuation while remaining thermally stable well above the blanket outlet temperature. Tungsten carbide provides attenuation of fast neutrons through inelastic scattering and boron carbide provides thermal capture. Nuclear heating within the shield is approximately 0.9 MW at the optimized operating point, and this load is intercepted by an actively cooled thermal break between the shield and the cryostat, which establishes the thermal gradient to the magnet operating temperature. A detailed thermal-hydraulic design of the shield and cryostat thermal path is outside the scope of the present neutronics study.

The axial extent of the model is set by the burn chamber length of 3.0 m, consistent with Fig. 1. With an annular cross-sectional area of 4.47 m² between the inner blanket radius of 13.2 cm and the outer radius of 120 cm, this gives the active blanket volume of 13.4 m³ quoted throughout. Axial reflector regions 10 cm thick, of the same tungsten-boron-carbon composition, terminate the blanket at both ends. Boundary conditions and the resulting leakage are discussed in Section 2.5

2.2 Blanket Material Composition

The fission blanket medium is a FLiNaK eutectic fluoride salt, which is a ternary mixture of LiF–NaF–KF in the eutectic molar ratio of 46.5:11.5:42, with two modifications from the natural composition. First, the lithium component is isotopically enriched to 90 mol% ^6Li , with the balance being ^7Li . The physics basis for this enrichment level is presented in Section 4.

Table 1. Blanket composition at 17.53 mol% TRU, 90% ^6Li . Homogenized blanket density 4.271 g/cm³ (90 vol% salt at 3.762 g/cm³, 10 vol% Hastelloy N at 8.86 g/cm³).

Constituent	Nuclide	Atom density (b ⁻¹ cm ⁻¹)
Carrier salt	F-19	3.2058e-02
	Li-6	8.1921e-03
	Li-7	9.1024e-04
	Na-23	2.2511e-03
	K-39	7.6672e-03
	K-40	9.6191e-07
	K-41	5.5332e-04
Plutonium	Pu-239	1.3595e-03
	Pu-240	7.8274e-04
	Pu-241	3.2958e-04
	Pu-242	2.4718e-04
Neptunium	Np-237	2.0599e-04
Americium	Am-241	6.1795e-04
	Am-243	2.4718e-04
Curium	Cm-244	2.0599e-04
	Cm-245	6.1795e-05
	Cm-246	6.1795e-05
	Cm-248	4.1197e-05
Structure (Hastelloy N,	Ni (all)	6.4544e-03
10 vol%)	Cr (all)	7.1697e-04
	Mo (all)	6.6531e-04
	Fe (all)	3.8177e-04
	Si (all)	1.8998e-04
	Mn-55	9.7121e-05
Total TRU		4.1609e-03

Here, it is sufficient to note that 90% ^6Li is the result of a systematic optimization sweep and represents the global optimum across all design objectives. Second, TRU fluorides are dissolved in the carrier salt at 17.53 mol%, replacing an equivalent molar fraction of KF. The transuranic

makeup adopted in this study is given in Table 1 together with the atom densities of every constituent of the blanket. The makeup is minor-actinide enriched: neptunium, americium and curium together constitute 34.7% of the transuranic atoms as opposed to approximately 11% in PUREX-separated light water reactor discharge at typical burnup and cooling time. This composition is adopted deliberately as the transmutation mission of the system is directed at the minor actinides rather than at plutonium, and a feed pre-concentrated in minor actinides is the composition a dedicated transmutation facility would receive from a partitioning facility. This isotopic composition was held fixed for all calculations reported here.

The blanket region of the transport model is a homogeneous mixture of 90 vol% FLiNaK+TRU salt with 10 vol% Hastelloy N, the latter representing distributed structural material within the blanket annulus. From the standpoint of the neutronics the structural component is not negligible. Its total atom density exceeds that of the transuranic inventory by a factor of 2 to 1, and the nickel component alone by a factor of 1.55. Nickel has appreciable capture in the epithermal range occupied by this spectrum, and the structural fraction therefore contributes to the neutron balance that sets k_{eff} . It is included in all results reported here.

The mass density of the FLiNaK+TRU mixture, from which all number densities in the transport calculation are derived, was obtained by ideal mixing of molar volumes. The carrier salt density was taken from the Janz correlation $\rho(\text{g cm}^{-3}) = 2.579 - 6.24 \times 10^{-4} T(^{\circ}\text{K})$, evaluated at the assumed bulk operating temperature of 973 °K and corrected for the ^6Li mass substitution at constant molar volume. The final value is 40.91 g/mol and a molar volume of 20.44 cm³/mol. The transuranic trifluoride component was assigned the molar volume of PuF₃, 31.7 cm³/mol, which lies within a few percent of the values for NpF₃, AmF₃ and CmF₃. Additivity of molar volumes at the 17.53 mol% design loading then gives a salt density of 3.76 g/cm³, and a homogenized blanket density of 4.271 g/cm³ including the 10% by volume representing the structural fraction. This value is consistent with the densities reported for molten salt fast reactor fuel salts at comparable heavy-metal loading. The salt was taken to operate at a bulk mean temperature of 973 K (700°C), which is above the 454°C melting point of FLiNaK and, as shown below, above the transuranic precipitation threshold that is the more restrictive constraint.

FLiNaK was selected as the carrier salt for several reasons specific to the transmutation application. A melting point of 454°C and boiling point above 1570°C, provides a large operating temperature window. Its low viscosity at operating temperatures facilitates continuous salt circulation for fission product removal and TRU feed replenishment without a reactor shutdown. Its moderate neutron absorption cross-section and low atomic mass components provide useful neutron moderation, whereas the dissolved TRU and ^6Li dominate the neutron absorption. The high electrical resistivity of FLiNaK (approximately 10⁶ times that of copper) also simplifies the salt flow through the magnetic field from the HTSC solenoid coils present in the blanket.

The solubility of transuranic trifluorides in the FLiNaK carrier at the loading required by this design has been measured directly. Lizin et al. [15] determined the solubility of PuF₃ and AmF₃ in the eutectic 46.5LiF–11.5NaF–42KF over the temperature range of 550–700 °C by isothermal saturation, obtaining 32.8 ± 3.3 mol% for PuF₃ and 43.2 ± 4.3 mol% for AmF₃ at 700°C, with the temperature dependence $\log S = 5.58 - 3949/T$ for PuF₃ and $\log S = 3.75 - 2052/T$ for AmF₃, S in mol% and T in K. Plutonium constitutes 65.4% of the transuranic makeup of Table 1 and americium 20.8%, so at the 17.53 mol% design loading the partial concentrations are 11.5 mol% PuF₃ and 3.6 mol% AmF₃, corresponding to 34% and 8% of the respective saturation limits at 700 °C. Because complex fluoride phases form in the saturated melt, the individual solubilities are not strictly additive; a conservative bound is obtained by treating the entire transuranic inventory as PuF₃, the least soluble of the measured species, on which basis the design loading is 53% of saturation at 700 °C. This measured solubility establishes a lower bound on the salt operating temperature that is more restrictive than the 454 °C melting point of FLiNaK. Setting the PuF₃ solubility equal to the total transuranic loading gives a precipitation threshold of 638 °C, so the assumed bulk operating

temperature of 700 °C carries a margin of 62 °C. The cold-leg temperature of the salt circuit, rather than the salt freezing point, is controlled by maintaining the entire circuit above approximately 640 °C.

Solubility data were not available for NpF_3 or the curium trifluorides, which together constitute 13.9% of the transuranic elements. Both are trivalent actinide fluorides chemically analogous to the measured species, and the same study [15] established CeF_3 and NdF_3 as solubility simulators for PuF_3 and AmF_3 respectively. While the solubility of NpF_3 and CmF_3 in this eutectic is a remains unknown, the small inventory fractions involved mean that even a substantial reduction relative to PuF_3 would not be sufficient to challenge the loading adopted here.

The combination of FLiNaK moderation and ^6Li enrichment produces a distinctive epithermal neutron spectrum in the active blanket, with a flux peak at approximately 60.6 keV. This spectral feature arises from the interplay between inelastic scattering on ^{19}F , which degrades high-energy neutrons into the epithermal range, and ^6Li absorption, which selectively removes thermal neutrons below the epithermal peak. The resulting spectrum is favorable for actinide transmutation. At 60.6 keV the fission-to-absorption ratios of the major MA species (^{237}Np and ^{241}Am) are substantially higher than those at thermal energies, while remaining accessible without the extreme flux levels required in a fast spectrum. This spectral characteristic and its sensitivity to ^6Li enrichment are analyzed quantitatively in Section 4.

2.3 Magnetic Field Effects on the Blanket System

The CFR2 burn chamber operates with a compressed axial magnetic field of approximately 30–35 T during the FRC transit, sustained by external HTSC solenoid coils that produce a 7–9 T vacuum bias field extending through and beyond the blanket region. One concern stems from the magnetohydrodynamic (MHD) effects in liquid metal blankets and whether these fields alter the blanket neutronics or impose operational constraints on the salt circuit. The answers in each case are quantitative and favorable, and each effect is addressed in turn.

2.3.1 Neutron transport

Neutrons are neutral particles, and their trajectories are unaffected by magnetic fields through the Lorentz force. Neutrons carry a magnetic dipole moment of $\mu_n = -1.913$ nuclear magnetons, and spin-dependent scattering in strong magnetic fields is the basis of neutron spin-echo spectroscopy and polarized neutron optics. However, the energy splitting between the spin-up and spin-down neutrons in a 9 T field is $\Delta E = 2\mu_n \approx 1.74 \times 10^{-7}$ eV. This is 11 orders of magnitude smaller than the kinetic energies of the neutrons in the FLiNaK blanket, which peak at 60.6 keV in the epithermal range. Therefore, any deflection or polarization effect on the neutron flux is negligible in the operating spectrum of this system, and no magnetic field correction is warranted for the OpenMC transport calculations.

2.3.2 Fission fragment and charged particle confinement

Fission fragments are heavy ions with a mass distribution characteristic of the fissioning nuclide, spanning roughly $A = 80$ –160 in the familiar double-humped yield distribution, and carrying initial kinetic energies of order 70–100 MeV. Their range in the fluoride salt at the design density is approximately 20 μm and they come to rest within a few picoseconds, neutralizing progressively as they slow. This stopping time is shorter than the gyro-period in the 7–9 T blanket field, which is tens of nanoseconds, by some four orders of magnitude, so the magnetic field plays no role in their transport. Fission energy is nonetheless deposited entirely within the salt by virtue of the short stopping range alone, which is consistent with the local energy deposition treatment adopted in the OpenMC heating analysis. The magnetic environment neither assists nor complicates the energy deposition picture.

2.3.3 MHD pressure drop in the salt circuit

The FLiNaK salt circuit must flow continuously through the blanket annulus in the presence of the HTSC solenoid's axial magnetic field, and concerns regarding the magnetohydrodynamic pressure drop are well-founded for liquid metal coolants such as sodium, which has an electrical conductivity of the order of 10^7 S/m. However, FLiNaK is an ionic molten salt with an electrical conductivity of approximately 2 S/m at the operating temperature, which is six orders of magnitude lower than that of sodium. The MHD pressure drop scales as:

$$\Delta P_{MHD} \propto \sigma \cdot v \cdot B^2 \cdot L_{ch} \quad (3)$$

where σ is the electrical conductivity, v is the salt flow velocity, B is the field strength, and L_{ch} is the channel length. The combination of low electrical conductivity and large hydraulic diameter of the annular blanket channel yields a pressure drop of approximately 2 kPa at the design flow velocity, requiring a pumping power of only 6 kW. Although the Hartmann number, $Ha \approx 500$, is formally large, the physically meaningful quantity is the absolute pressure drop rather than the dimensionless parameter, which is negligible for fluoride salts. It should also be mentioned that the salt flow is mostly parallel to the magnetic field. The conclusion is that the MHD effects impose no meaningful constraints on the FLiNaK flow circuit design.

2.34 First wall neutron loading.

The neutron wall loading is evaluated at the plasma-facing surface of the W-25Re first wall at $r = 6$ cm, following the conventional definition as the total neutron energy current per unit first wall area, with no credit taken for the absorbed fraction. At the optimized operating point, the source delivers 4×10^{18} n/s at a flux-weighted mean energy of 11.77 MeV, corresponding to 7.54 MW of neutron power. Distributed over the 1.13 m^2 first wall area of the 3.0 m burn chamber, the wall loading is 6.67 MW/m^2 . At the 1% tritium Cat-D baseline the corresponding value is 4.69 MW/m^2 .

This quantity is a neutron energy current and not a heat load. The direct measure of nuclear heating in the first wall yields 179 kW, or 2.37% of the incident neutron power, of which 97.6% is neutron heating and 2.4% is photon heating from the blanket. The wall is therefore essentially transparent to the fusion neutrons, which is the physical requirement for the blanket to receive them. The volumetric heating is 2.97 W/cm^3 , and the heat flux presented to the helium cooling gap at $r = 10$ cm is 0.095 MW/m^2 . The bulk temperature rise per pulse at 4 Hz is 0.27 K. The pulsed thermal transient in the first wall is thus negligible, and thermal loading does not constrain the design.

The limiting mechanisms are displacement damage and transmutation. The NRT displacement damage rate, computed from a damage-energy total with a displacement threshold, E_d , of 90 eV, is 11.56 dpa/yr, or 10.4 dpa per full-power year at a 0.90 capacity factor. Transmutation can be even more consequential for a W-25Re first wall. Tungsten transmutes to rhenium at 0.86 at.%/yr and rhenium to osmium at 1.08 at.%/yr, so that the net effect is depletion of the rhenium content and accumulation of osmium at approximately 1 at.% per year. Since W-25Re is already near the rhenium solubility limit, osmium ingress at this rate is expected to drive sigma and chi phase precipitation and consequent embrittlement. Helium generation, at 5.5 appm/yr or 0.48 appm/dpa, is an order of magnitude below the ratios characteristic of steels to be employed in fusion first walls and is not the controlling embrittlement pathway in this material.

These rates are consistent with the first wall being treated as a consumable component on a replacement interval as short as one year. This is quite doable given the easy end-on access provided by the small, linear geometry of the CFR2 fusion neutron source. The burn chamber is a simple cylindrical tube with open axial access. It is not structurally integrated with confinement coils, carries no breeding function, and requires no penetrations for heating, current drive, fueling or divertor structures, since the FRC plasmoid neutron source is formed remotely and transits the

chamber passively. The design requirement placed on the W-25Re alloy is therefore dimensional stability and retention of tensile strength over a year or more of service rather than over a plant lifetime. Qualification of W-25Re under combined pulsed magnetic and neutron loading at these rates is outside the scope of the present neutronics study and is identified in Section 8 as required supporting work.

Should the damage rate prove limiting, the burn chamber length provides a direct remedy. Because axial leakage is negligible, lengthening the chamber and blanket at constant total neutron rate reduces the wall loading in inverse proportion while leaving the blanket neutronics essentially unchanged: a 6 m chamber would halve the wall loading to 3.3 MW/m². The cost is a proportionate increase in blanket volume and transuranic inventory. Thus, the initial reference design for the concept adopts the compact end of this range.

In summary, the 7–9 T magnetic field environment of the CFR2 blanket does not alter neutron transport physics, plays no part in the deposition of fission fragment energy, which is local by virtue of the short stopping, and imposes negligible MHD constraints on the low-conductivity FLiNaK coolant. The OpenMC calculations reported in this study did not require magnetic field corrections, and the blanket design is not constrained by the electromagnetic effects on the salt circuit.

2.4 OpenMC Computational Methodology

All neutronics calculations reported in this study were performed using OpenMC version 0.14, an open-source Monte Carlo particle transport code [16]. OpenMC solves both the k-eigenvalue problem (for criticality calculations) and the fixed-source problem (for neutron flux, reaction rate, and energy deposition totals driven by an external source). The ENDF/B-VIII.0 nuclear data library was used throughout the study, which provides evaluated cross-sections for all isotopes present in the blanket system, including the minor actinides.

The blanket system was modelled as a right circular cylinder in the OpenMC constructive solid geometry framework, with radial zones defined as concentric cylindrical shells following the radial build of Section 2.1, each assigned a homogeneous material composition. Vacuum boundary conditions were applied to every outer surface of the model, radial and axial. No reflective or periodic boundary condition was used anywhere, and all neutron leakage is therefore physical. The steel-tungsten radial reflector and the tungsten-boron-carbon axial reflectors referred to in Section 2.1 are physical material regions within the model and are not a reflective boundary condition.

Two distinct leakage quantities characterize the model. Leakage of fission neutrons across the outer boundaries of the system is 0.13%, reflecting the optical thickness of the $r = 106.8$ cm blanket, so that the outer boundary condition has no influence on any of the results reported. One should note however that 8.4% of the source neutrons stream axially out of the open burn chamber bore without entering the blanket. The bore must remain open for the FRC to transit the chamber, and this loss is therefore intrinsic to the configuration, and it is accounted for in all of the results reported here.

Cross sections were evaluated at 293.6 °K, the only temperature in the library that was available. Doppler broadening of the actinide capture resonances is therefore not represented at the 973 K salt operating temperature, although the number densities themselves are evaluated at that temperature from the density correlation of Section 2.2. This approximation under-predicts resonance absorption and consequently over-predicts k_{eff} . The effect is expected to be small in the present system because the neutron flux peaks at 60.6 keV, which is above the resolved resonance region of the principal actinides, but it has not been quantified here. Its practical consequence is a small correction to the transuranic loading required to reach a given k_{eff} , and it does not affect the spectral, tritium-breeding or control results reported in this paper.

Eigenvalue calculations are as follows: The k-eigenvalue calculation was run using the combined collision, track-length, and absorption estimators. All three estimators were reported for each calculation; agreement among them within the statistical uncertainty confirmed that no systematic

bias was present. A typical eigenvalue run employed 400 active cycles of 100,000 particles each and was preceded by 100 inactive cycles for source convergence. The statistical uncertainties in k_{eff} were less than ± 0.0002 (1σ) for all reported cases. Source convergence was monitored by Shannon entropy, which settled from 9.697 to 9.447 across the inactive cycles and held at 9.438 ± 0.006 through the active cycles.

For Fixed-source calculations, the blanket performance metrics included fission power, tritium production rate, actinide fission-to-absorption ratios, and neutron energy spectrum. These were obtained from fixed-source calculations in which the CFR2 neutron source was represented as a volumetric cylindrical source distributed uniformly over the burn chamber bore ($r = 0\text{--}6$ cm) with an isotropic angular distribution. The neutron energy spectrum of the source was represented as a two-component mixture: a D-T component at 14.1 MeV with a fractional yield f_{DT} and a D-D component at 2.45 MeV with a fractional yield $f_{\text{DD}} = 1 - f_{\text{DT}}$, where $f_{\text{DT}} = 100f_{\text{T}}/(1+100f_{\text{T}})$, and f_{T} is the plasma tritium fraction. The total source strength was maintained at 4×10^{18} n/s in all fixed-source calculations to isolate the spectral effects from the source intensity effects. The absolute power levels were then obtained by linear scaling to the actual operating neutron rate.

Energy deposition in the blanket was computed using the heating-local estimator, which captures both prompt fission energy and the energy deposited by fission product recoils, capture gamma rays, and $(n,2n)$ reactions. Tritium production was calculated as the reaction rate of the ${}^6\text{Li}(n,t)\alpha$ reaction integrated over the blanket volume. The fission-to-absorption ratios for individual actinide isotopes were obtained from separate fission and absorption reaction rate calculations. The neutron flux energy spectrum was totaled in 1,000 logarithmically spaced energy bins from 10^{-3} eV to 20 MeV. Survival biasing was employed with a weight cutoff of 0.25 and a survival weight of 1.0. For the shielding and first wall damage calculations reported in Sections 2.3.4 and 2.7, photon transport was enabled and heating was calculated separately for neutrons and photons.

The convergence and uncertainty for the fixed-source calculations were run with 3×10^6 source particles per case, yielding relative uncertainties below 0.5% for all integral quantities (fission power, TBR, and total TRU burn rate) and below 2% for individual isotopic reaction rates. The agreement between the 3×10^6 and 4×10^7 history cases was within 0.14% on total fission power.

2.5 Benchmark Validation

Before applying the OpenMC model to the CFR2 geometry, the code and cross-section library were validated against an independently reported subcritical transmutation system. The reference system selected for this purpose is the In-Zinerator concept [9], a fusion-fission hybrid driven by a D-T fusion neutron source and employing a lead-sodium coolant blanket carrying TRU at $k_{\text{eff}} = 0.97$. The In-Zinerator was designed and analyzed by Sandia National Laboratories, which provided the blanket geometry, material composition, and neutronics performance metrics against which an independent calculation could be compared.

The In-Zinerator geometry was reconstructed in OpenMC from the specifications in [9]: a toroidal blanket surrounding a toroidal fusion source with a lead-sodium fast-spectrum coolant and TRU loading adjusted to achieve $k_{\text{eff}} = 0.97$. The OpenMC eigenvalue calculation for this reconstructed geometry reproduces the reported k_{eff} to within 0.3%, and the calculated fission power per source neutron agrees with the reported value within the combined statistical and systematic uncertainties. This agreement establishes that the OpenMC code system, nuclear data library, and geometry modeling approach used in this study are consistent with the independently validated In-Zinerator analysis and are therefore suitable for application to the CFR2 blanket design.

Having established this benchmark, the In-Zinerator also serves as a natural performance reference throughout this study: both systems operate at or near $k_{\text{eff}} = 0.97$ with TRU-bearing

blankets driven by external fusion neutron sources, making a direct comparison of the transmutation rate and blanket volume informative.

2.6 Neutron Environment at the Superconducting Magnet

The high temperature superconducting solenoid lies outboard of the blanket, reflector and thermal shield, and the fast neutron fluence it accumulates sets a limit on magnet life. Fixed-source calculations with the transport model extended through the full radial build to the cryostat give a fast neutron flux ($E > 0.1$ MeV) at the inner surface of the winding pack of $1.78 \times 10^9 \text{ n cm}^{-2} \text{ s}^{-1}$. A shield thickness scan gives a fast flux attenuation length of 3.8 cm in the tungsten carbide/boron carbide composite, so that magnet life is a steep function of shield thickness. A 20 cm shield yields 3.7 years and a 40 cm shield several hundred. At the optimized operating point with the 30 cm thermal shield of the reference design, the corresponding annual fluence is $5.6 \times 10^{20} \text{ n/m}^2$. This is to be compared to the onset of critical current degradation in REBCO coated conductors at $3 \times 10^{22} \text{ n/m}^2$, giving a magnet life of approximately 53 years. Nuclear heating in the winding pack is 0.062 mW/cm^3 , or 0.27 kW over the coil volume, requiring on the order of 4 kW of refrigeration power at a 50–65 °K operating temperature and as such imposes no meaningful load on the plant power balance.

It is a unique advantage of the CFR2 configuration that shield thickness may be chosen on this basis alone. The superconducting solenoid supplies only the 7–9 T on-axis vacuum bias field. The 30–35 T field present during the fusion burn is the result of the flux compression of the bias field inside the conducting wall of the burn chamber, which is produced by the FRC plasmoid radial pressure as the FRC passes through. This process is independent of the solenoid radius. Increasing the shield therefore displaces the coils outward without altering the confining field at the plasma boundary. In toroidal configurations, by contrast, shield thickness trades directly against on-axis toroidal field and hence against fusion gain, and the shielding of the magnet is a primary constraint on the reactor design rather than a matter of cost.

3. Blanket Design Evolution and the Optimum Reference Design

3.1 Design Progression

The initial design (v1) employed a natural lithium FLiNaK carrier salt with a conservative TRU loading of 10 mol% and a blanket thickness of 35 cm. The eigenvalue calculation returned $k_{\text{eff}} = 0.53$, which was substantially below the target value, indicating that the TRU loading was insufficient to achieve the desired subcriticality margin. The neutron spectrum in v1 was predominantly thermal, with a flux peak near 0.1 eV, which is characteristic of FLiNaK without ^6Li enrichment. The tritium breeding from natural lithium (7.4% ^6Li) was modest, with a TBR of 3.1. Table 2 summarizes the key parameters for each design version, v1 to v4b. It should be noted that versions v1-v3 were computed at the incorrect Janz value for the salt density; their entries in Table 2 are retained as the historical progression. Only the v4b column has been recomputed at the corrected density.

In version 2, the TRU loading was increased to 15 mol%, and the blanket thickness was increased to 83 cm. The eigenvalue increased to $k_{\text{eff}} = 0.91$, which approached but did not reach the target. Spectral analysis showed that the flux peak began to shift into the epithermal range as the increased TRU absorption competed with the moderating salt to reshape the spectrum above the cadmium cutoff.

Version v3 introduced ^6Li enrichment at 50% as a preliminary enrichment level chosen to explore the sensitivity of k_{eff} and TBR to isotopic composition before committing to a full-optimization sweep. The TRU loading was adjusted to 18 mol%, and the blanket thickness was increased to 107 cm in the second case. The eigenvalue reached $k_{\text{eff}} = 0.944$. Tritium breeding increased sharply to TBR = 14.2, confirming that ^6Li enrichment significantly amplifies tritium production through the fission-multiplied thermal flux. The neutron spectrum peak shifted to approximately 10 keV, reflecting the combined effect of ^6Li absorption cutting the thermal flux below the epithermal range and inelastic scattering on the carrier salt populating the 1–100 keV band.

Table 2. Key parameters for each design version from v1 to v4b.

Parameter	v1	v2	v3	v4b
Outer radius	48 cm	96 cm	120 cm	120 cm
Blanket thickness	35 cm	83 cm	107 cm	107 cm
TRU mol%	10%	15%	18%	17.53%
^6Li enrichment	natural	natural	50%	90%
TBR (based on T consumed)	-	3.1	14.2	25.6
k_{eff}	0.53	0.91	0.944	0.9695
Fission power	46 MW	435 MW	611 MW	1,240 MW
TRU burn	18 kg/yr	171 kg/yr	241 kg/yr	489 kg/yr
MA support	0.2 LWRs	2.1 LWRs	3.0 LWRs	6.8 LWRs
System Vol.	—	—	—	13.4 m ³

The v4 series (v4, v4a, and v4b) refined the enrichment to 90% ^6Li . The TRU loading was adjusted along with the blanket thickness to yield $k_{\text{eff}} = 0.97$. In v4, a TRU loading of 18 mol% with a 104 cm blanket returned $k_{\text{eff}} = 0.963$. Version v4a raised the TRU to 19 mol% at 106 cm, achieving $k_{\text{eff}} = 0.971$. The final v4b configuration using the corrected salt density of 17.53 mol% TRU and a blanket thickness of 106.8 cm (inner radius of 13.2 cm to outer radius of 120 cm), comprises the reference design. The three OpenMC k-eigenvalue estimators, namely track-length, collision, and absorption, returned k_{eff} values of 0.9694, 0.9694, and 0.9698, respectively, which agreed to within 0.04% and provided high confidence in the criticality result. Spectral analysis of v4b showed a characteristic epithermal peak at 60.6 keV (see Fig. 5).

Throughout this paper, MA support denotes the number of 1 GWe light water reactors whose minor actinide generation rate is matched by the minor actinide destruction rate of a single CFR2 module. It is evaluated as $\text{MA support} = f_{\text{MA}} \times \dot{M}_{\text{TRU}} / 25$, where $\dot{M}_{\text{TRU}}$ is the module transuranic destruction rate in kg/yr, and $f_{\text{MA}} = 0.347$ is the minor actinide fraction of the transuranic feed of

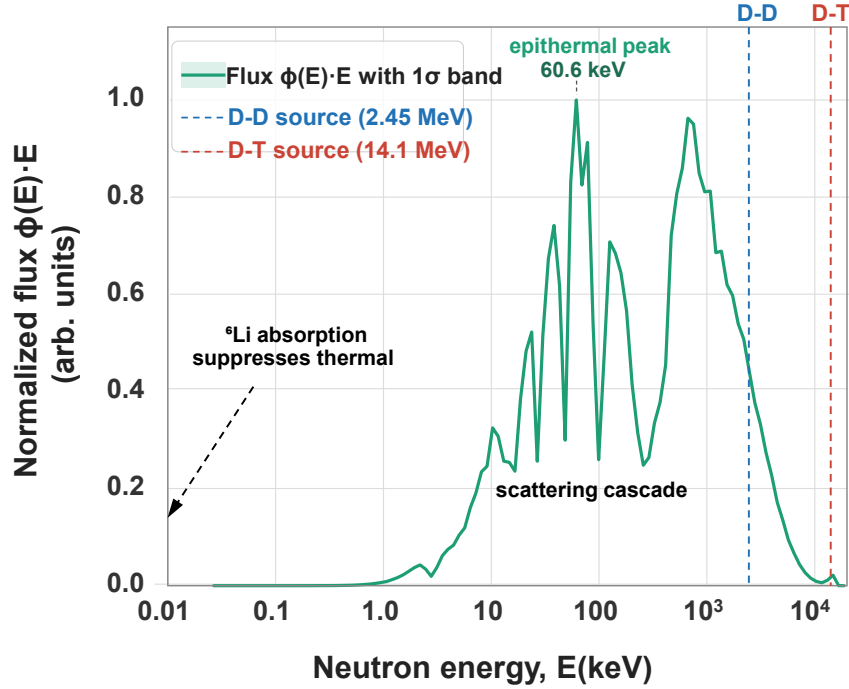


Figure 5. Spectral analysis of the results of the v4b blanket showing a characteristic epithermal peak at 60.6 keV

Table 1, and 25 kg/yr is the assumed minor actinide generation rate per GWe of light water reactor capacity. On this basis the Cat-D baseline and optimized operating points support 6.8 and 8.0 light water reactors respectively.

3.2 The Optimum Reference Design

The optimal blanket design reported in this study was not obtained in a single step but through a systematic progression of OpenMC eigenvalue calculations, in which the primary design variables, namely, blanket thickness, TRU loading, and ^6Li enrichment, were adjusted iteratively to converge on a configuration that simultaneously satisfies the subcriticality target $k_{\text{eff}} = 0.97$, produces an epithermal neutron spectrum that is well-matched to actinide transmutation. As shown in Table 3, the baseline Cat-D CFR2 source operating point (1% T, 10 Hz, $E_{\text{src}} = 8.275$ MeV) produces a fission power of 1,240 MWth. The mean power density in the active blanket volume of 13.41 m³ is 92.6 MW/m³ at the Cat-D baseline, rising to 109.7 MW/m³ at the optimized operating point. This is high relative to solid-fueled reactors, but the heat is generated directly within the circulating coolant rather than having to be conducted across a fuel-cladding interface, which removes the limiting thermal resistance. For 1,471 MWth removed at a 100 °K salt temperature rise, and with $c_p = 1.88$ kJ kg⁻¹ °K⁻¹ and $\rho = 3.762$ g cm⁻³, the required volumetric flow is 2.08 m³/s, corresponding to a mean axial velocity of 0.47 m/s through the 4.47 m² blanket annulus. This is a low velocity by molten salt reactor standards and is consistent with the magnetohydrodynamic pressure drop and pumping power given in Section 2.3.3

The neutron energy spectrum of the v4b blanket is shown in Figure 5. This is the flux averaged over the entire active blanket volume, obtained from a fixed-source calculation with the CFR2 source as described in Section 2.5, and tallied in 1,000 logarithmically spaced energy bins from 10⁻³ eV to 20 MeV. It is not a local spectrum, and the flux at the inner and outer boundaries of the blanket differs from the volume average. The flux distribution is distinctly epithermal, with a pronounced peak at 60.6 keV, which is the signature feature of the FLiNaK+ ^6Li system. The peak

arises from the interplay of three processes. Source neutrons at 2.45 and 14.1 MeV are degraded in energy principally by inelastic scattering on ^{19}F , which constitutes approximately half the atoms of the carrier salt and has inelastic levels from 110 keV upward. Nuclide-resolved data show that fluorine accounts for 53–93% of all inelastic reactions in the blanket across the full energy range, with the remainder distributed among sodium, potassium and the actinides. ^7Li contributes less than 1% at the reference enrichment. Below the fluorine inelastic threshold, elastic scattering on the light constituents continues the slowing-down. Below approximately 1 keV, ^6Li absorption, whose cross section rises as $1/v$ toward lower energies, removes neutrons faster than supplied from above, truncating the spectrum.

Table 3. Comparison between the baseline Cat-D performance with the self-supply of T at 1% from the FRC plasma and that of enhanced T supplied by the blanket to 4%.

Notes	Parameter	Baseline Cat-D 1% T @ 10 Hz	Optimized with 4% T @ 4 Hz
Held constant	TRU mol%	17.53%	17.53%
Held constant	^6Li enrichment	90%	90%
	k_{eff} (eigenvalue)	0.9695	0.9695
	Spectrum peak	60.6 keV	60.6 keV
Held constant	Neutron source	4×10^{18} n/s	4×10^{18} n/s
	Rep rate	10 Hz	4 Hz
CFR2 power consumed	Driver power	43.8 MW	17.5 MW
Based on 47.5% conv. eff.	Recirculated fission power fraction	7.4%	2.5%
	Fission power	1,240 MWth	1,471 MWth
	TRU burn	489 kg/yr	579 kg/yr
	MA support	6.8 LWRs	8.0 LWRs
	Net T production	8.10 kg/yr	9.2 kg/yr
Based on T consumed	TBR	25.6	18.9
	Blanket Power Density	92.6 MW/m ³	109.7 MW/m ³
MA fission set by % ^6Li	Np-237 f/a	0.336	0.336
MA fission set by % ^6Li	Am-241 f/a	0.270	0.270

The position of the peak is set by the fluoride carrier and is found to be independent of ^6Li enrichment, remaining at 60.6 keV across the full range from 1% to 90%. What ^6Li enrichment controls is the flux on either side of the peak. As enrichment increases from 1% to 90%, the fraction of the flux below 1 keV falls from 5.55% to 0.053%, a suppression of two orders of magnitude, while the fraction above 100 keV rises from 37.0% to 60.7%. Both changes favor the destruction of minor actinides by fission: the first removes flux from the region in which radiative capture dominates fission for ^{237}Np and ^{241}Am , and the second populates the region above their fast fission thresholds.

The result is a flux distribution that is concentrated in the 10 keV – 1 MeV epithermal window, which is favorable for transmutation objectives for two reasons. First, the fission cross-sections of the main minor actinides, ^{237}Np , ^{241}Am , ^{243}Am , and ^{244}Cm , are substantially elevated in this energy range relative to the thermal spectrum, while remaining sufficiently high for efficient fission

without requiring the hard fast-spectrum of a lead-cooled blanket to be present. Second, the moderated epithermal spectrum supports a higher fissile loading per unit volume than a fast-spectrum blanket, enabling a compact blanket geometry that distinguishes the CFR2 system from other systems.

The fission-to-absorption ratios for the key actinide species under the v4b conditions are as follows: Np-237, $f/a = 0.336$; Am-241, $f/a = 0.270$. The corresponding plutonium ratios are Pu-239 $f/a = 0.847$, Pu-240 $f/a = 0.620$, Pu-241 $f/a = 0.870$ and Pu-242 $f/a = 0.609$, and for Cm-244 $f/a = 0.641$. These values reflect the epithermal spectral peak: at 60.6 keV, both isotopes have fission cross-sections large enough to compete effectively with their radiative capture cross-sections, directing a significant fraction of neutron absorptions toward fission and actinide destruction rather than transmuting to higher actinides. At fully thermal energies, the f/a ratios of both species fall substantially owing to the dominance of the capture resonances in the eV–keV range, whereas at fully fast energies (above 1 MeV), the fission cross sections are high, but the flux is diluted. The 60.6 keV spectrum represents a near-optimal compromise.

The tritium breeding ratio at the v4b baseline is $TBR = 25.6$. This large value, far exceeding the $TBR \approx 1.05$ target typical for tritium self-sufficient D-T fusion reactor blankets, arises directly from the extremely low, ideally zero, net tritium consumption in the Cat-D CFR2 source, as well as the combination of 90% ^6Li enrichment and fission multiplication. Each source neutron that enters the blanket triggers, on average, $M = 1/(1 - k_{\text{eff}}) = 33$ fission chain interactions before leaking or being nonproductively absorbed. The resulting multiplied flux is available for $^6\text{Li}(n,t)\alpha$ reactions throughout the blanket volume, yielding a tritium production rate of 8,097 g/yr at the baseline operating point of the reactor. The gross plasma tritium burn at 10 Hz is 316 g/yr, and the D-D proton branch supplies tritons at the same rate as the D-D neutron branch, so the net demand on the blanket at the 1% equilibrium is zero and the whole of blanket production is available for export.

Section 4. ^6Li Enrichment Optimization

4.1 The Three Simultaneous Roles of ^6Li

The choice of ^6Li enrichment level in the FLiNaK carrier salt is one of the most consequential design parameters for the CFR2 transmutation blanket. This is due to the fact that ^6Li simultaneously performs three distinct neutron-related functions that in most reactor blanket designs would be served by separate materials or engineering systems. The first role is that of a neutron poison that controls k_{eff} . The $^6\text{Li}(n,t)\alpha$ reaction has a large thermal and epithermal absorption cross-section with a characteristic $1/v$ energy dependence that rises steeply toward lower energies, supplemented by a resonance near 0.24 MeV. Every neutron captured by ^6Li is removed from the fission chain reaction. In the design study, the TRU loading was adjusted at each enrichment level to maintain a k_{eff} of 0.97. As ^6Li enrichment increases and more neutrons are absorbed by lithium rather than fissile actinides, a larger TRU inventory is required for compensation of the neutron loss. This coupling between ^6Li enrichment and TRU mol% is visible in Figure 6, which shows that the TRU mol% rises monotonically from approximately 5.8% at 1% ^6Li enrichment to 17.53% at 90% enrichment, all at a constant $k_{\text{eff}} = 0.97$.

The second role is as a primary tritium source. Each $^6\text{Li}(n,t)\alpha$ absorption event produces one triton. Because the neutron flux driving these reactions is fission-multiplied by the factor $M = 1/(1 - k_{\text{eff}}) \approx 33$, the tritium production rate per source neutron is amplified far beyond what the bare fusion source alone could produce. The TBR at any given enrichment level therefore rises steeply with the ^6Li fraction, as shown in Fig. 6, where the TBR increases from 2.0 at natural lithium enrichment (7.4% ^6Li) to 25.6 at 90% enrichment and $k_{\text{eff}} = 0.97$.

The third role is that of a neutron shaper. The $^6\text{Li}(n,t)\alpha$ cross section rises as $1/v$ toward lower energies, so increasing enrichment progressively depletes the flux below approximately 1 keV, the

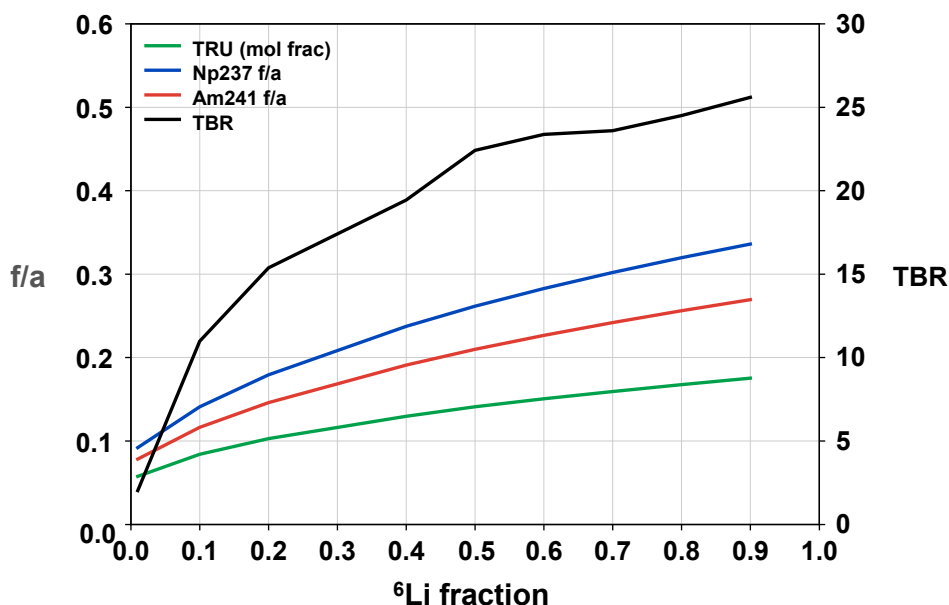


Figure 6. Results from calculations performed over the range of ⁶Li enrichment from 1% to 90% in steps of approximately 10%, with the TRU mol% adjusted at each step to maintain $k_{\text{eff}} = 0.97$.

region in which radiative capture dominates fission for the minor actinides. Across the enrichment range from 1% to 90% the sub-keV flux fraction falls from 5.55% to 0.053% while the fraction above 100 keV rises from 37.0% to 60.7%. Both movements raise the proportion of minor actinide absorptions that terminate in fission. The position of the epithermal peak itself is set by the fluoride carrier and does not move with enrichment, as described in Section 3.2. These three roles are not independent of one another. They are coupled through the neutron balance and the requirement to maintain $k_{\text{eff}} = 0.97$. The question of whether a single enrichment level can simultaneously optimize all three functions or whether trade-offs among them constrain the design to a compromise is resolved by the systematic enrichment sweep described in the following section.

4.2 The ⁶Li Enrichment Sweep

A series of OpenMC eigenvalue and fixed-source calculations were performed over the full range of ⁶Li enrichment from 1% to 90% in steps of approximately 10%, with the TRU mol% adjusted at each step to maintain $k_{\text{eff}} = 0.97$. The source strength and spectrum were also maintained at the baseline Cat-D values ($E_{\text{src}} = 8.275$ MeV, 4×10^{18} n/s). The results are summarized in Fig. 6, which shows the ²³⁷Np and ²⁴¹Am fission-to-absorption ratio (f/a), tritium breeding ratio (TBR), and TRU molar fraction as continuous functions of the ⁶Li enrichment fraction.

The central result is unambiguous: every performance metric of interest, from the transmutation of MA to tritium production, improves monotonically and without exception as ⁶Li enrichment increases. The ²³⁷Np f/a ratio increased from approximately 0.092 at 1% ⁶Li to 0.336 at 90%, a factor of 3.7 improvement. The Am-241 f/a ratio increased from approximately 0.078 to 0.270 over the same range. The TBR increased from approximately 2 at 1% to 25.6 at 90% enrichment. All three curves are monotonically increasing, with the steepest gains at low enrichment and diminishing but still positive returns as enrichment approaches 90%. The global optimum for each metric lies at or beyond 90% enrichment.

The physical reason that the f/a ratios for the minor actinides improve with ⁶Li enrichment is the spectral shift described in Section 4.1. As higher ⁶Li enrichment depletes the sub-keV flux and hardens the spectrum above 100 keV, both ²³⁷Np and ²⁴¹Am experience a higher ratio of flux-

weighted fission cross-section to the flux-weighted total absorption cross-section. At thermal energies, both isotopes have large capture resonances in the eV to keV range, which dominate over fission. At the 60.6 keV peak, the fission cross-sections are substantially elevated, while the capture resonances are less prominent, increasing the fraction of absorptions that result in complete actinide destruction rather than transmutation to a heavier isotope.

It is important to note what does not change across the enrichment sweep: the total TRU destruction rate and total fission power remain essentially constant because both are set by k_{eff} and the CFR2 source strength rather than by the isotopic composition. Normalized to a common k_{eff} , the fission power varies by 4.3% across the nine enrichment points with no systematic trend. The quality of the transmutation improves with enrichment; that is, the fraction of MA absorption that results in fission rather than the production of heavier and often harder-to-fission actinides, such as ^{244}Cm and ^{252}Cf . A higher MA f/a ratio at 90% ^6Li indicates that the ^{241}Am inventory is preferentially destroyed by fission, generating short-lived fission products rather than accumulating isotopes that complicate long-term storage.

Section 5. Tritium Production and Optimization

5.1 The Global Tritium Situation

Tritium is the rarest of fusion fuels. Despite being produced in some fission reactors, the global inventory stands at approximately 28 kg, virtually all of which is held by CANDU operators in Canada as a byproduct of heavy water (D_2O) neutron capture reactions [1]. In addition, tritium is an unstable form of hydrogen with a very short half-life of 12.3 years. Thus, the tritium inventory is declining as the CANDU fleet ages toward retirement, with no replacement production pathway at a comparable scale within the current nuclear infrastructure. This decline is not merely an inconvenience; it represents a fundamental constraint on the pace at which D-T fusion power can be developed and deployed in the future. A single 1 GWth D-T fusion plant would consume 28 kg of tritium in 182 days, so the startup inventory for one reactor is of roughly the same magnitude as the entire world supply.

A typical D-T fusion power plant will require approximately 55 kg of tritium for the initial startup inventory, plus a continuous supply to replace the tritium burned during operation and lost to radioactive decay. Tritium self-sufficiency for a fusion reactor blanket will require a tritium breeding ratio of at least 1.05 to account for losses during processing and decay during storage. In addition, there is the issue of recovering the unburned tritium lost from the plasma. The burn fraction for even the largest fusion reactors, such as ITER, with a projected fusion gain of 10, is only about 0.5%. If one assumes the current state-of-the-art recovery for the exhausted tritium at 99%, one would need a TBR of 3 for sustained operation. This is well beyond the TBR projected for any current blanket design. Even this TBR assumes that tritium retention in the reactor walls does not increase the tritium losses, as has been observed in past experiments at TFTR and JET, where as much as 30% of the initial inventory was unaccounted for. If TBR targets prove too difficult to achieve, or if tritium recovery efficiencies fall short of what is needed for continuous operation, the development of fusion as an alternate source of power would be severely jeopardized.

The CFR2 Cat-D transmutation system addresses this problem from an unexpected direction. Because the fission-multiplied neutron flux in the FLiNaK blanket drives $^6\text{Li}(n,t)\alpha$ reactions at a rate far exceeding the plasma's own tritium consumption, the CFR2 is not a net tritium consumer; instead, it is a tritium factory operating at a scale commensurate with the needs of a growing fusion fleet.

5.2 Tritium Production in the v4b Blanket

The tritium breeding ratio of the v4b baseline design (1% T, 10 Hz, $k_{\text{eff}} = 0.9695$) is $\text{TBR} = 25.6$, as determined from the OpenMC fixed-source calculation of the $^6\text{Li}(n,t)\alpha$ and $^7\text{Li}(n,n't)$ reaction

rates integrated over the active blanket volume. The ${}^7\text{Li}$ contribution to the TBR is quite small (6.2 out of 8,096 g/yr), with essentially all tritium production arising from the ${}^6\text{Li}$ branch. The dominant physics driving this large TBR value is the fission multiplication with $M = 33$, before the neutron population is absorbed or leaks. The resulting amplified flux is available throughout the blanket volume for ${}^6\text{Li}$ absorption, yielding tritium at a rate that scales with both the ${}^6\text{Li}$ atom density and fission-multiplied flux rather than with the bare source strength alone.

As mentioned at the baseline operating point of 4×10^{18} n/s and a repetition rate of 10 Hz, the gross tritium production rate is 8,096 g/yr. The plasma tritium consumption at this operating point, that is, the tritium burned in D-T reactions within the FRC, is only 316 g/yr, and this amount of tritium is matched with that created by the D-D reactions. This is what is meant by self-supplied Cat-D fusion. Given the huge T harvest compared to that used by the FRC, it was certainly worth investigating the result of enhancing the tritium content of the FRC beyond merely self-supply. This turned out to be quite favorable to do so. Raising the T to 4% in the FRC plasma requires adding an additional 379 g of T per year, which is a negligible amount compared to what is produced. However, this small amount significantly increased the fission output for the same FRC neutron input rate. Somewhat surprisingly, the tritium production also increased appreciably to 9.6 kg/yr, which more than compensated for the small amount recirculated back into the FRC. This increase in tritium in the FRC to 4% turned out to be the optimized operating point for reasons that will be made clear from the discussion below.

The optimized operating point is characterized by 4% T in the FRC and 4 Hz operation of the CFR2, resulting in a source multiplication of $k_s = 0.9726$ at an unchanged eigenvalue $k_{\text{eff}} = 0.9695$. To maintain a k_{eff} near 0.97, the neutron flux from the driver must remain relatively unchanged. This required a reduction in the repetition rate to 4 Hz. The higher tritium fraction increases both the D-T reaction rate in the plasma and the mean source neutron energy to $E_{\text{src}} = 11.77$ MeV, both of which increase the blanket fission power. This is beneficial because it reduces the demand from CFR2 power electronics and fatigue of the CFR2 structural components. Consequently, the fission-multiplied flux is also available for tritium production. The plasma tritium consumption increases with a higher tritium fraction and lower rep rate but remains small relative to the blanket production. The net surplus at the optimized point was approximately 9.2 kg/yr per module at a TBR of 18.9. The reduction in TBR from 25.6 to 18.9 between the baseline and optimized points is expected; the higher ${}^6\text{Li}(n,t)\alpha$ rate under optimized conditions is partially offset by the increased plasma tritium burn, and the definition of TBR as the ratio of total tritium production to plasma consumption is sensitive to the denominator at low consumption rates. On the net-demand basis, which subtracts the tritons supplied by the D-D proton branch, the corresponding ratio is 25.3. The absolute surplus, which is the quantity that matters for fleet fueling, increased from the baseline to the optimized operating point. Table 3 provides a comparison between the baseline Cat-D performance with the self-supply of T at 1% from the FRC plasma and that of enhanced T supplied by the blanket to 4%.

As mentioned, an OpenMC sweep starting with the baseline Cat-D results and increasing the T content from 1% to as much as 50% while adjusting the rep rate to maintain the total FRC neutron flux at 4×10^{18} n/s. In addition, the TRU was maintained at a molar fraction of 0.1753 and ${}^6\text{Li}$ at 0.9 enrichment throughout the sweep. A series of ten OpenMC fixed-source calculations were performed, in which the plasma tritium fraction was varied from 1% (the Cat-D equilibrium baseline) to 50%, spanning the full range from pure Cat-D operation to near-pure D-T operation. The mean source energy at each operating point is expressed as:

$$E_{\text{src}} (\text{MeV}) = 14.1 \cdot f_{\text{DT}} + 2.45 \cdot f_{\text{DD}}, \quad (4a)$$

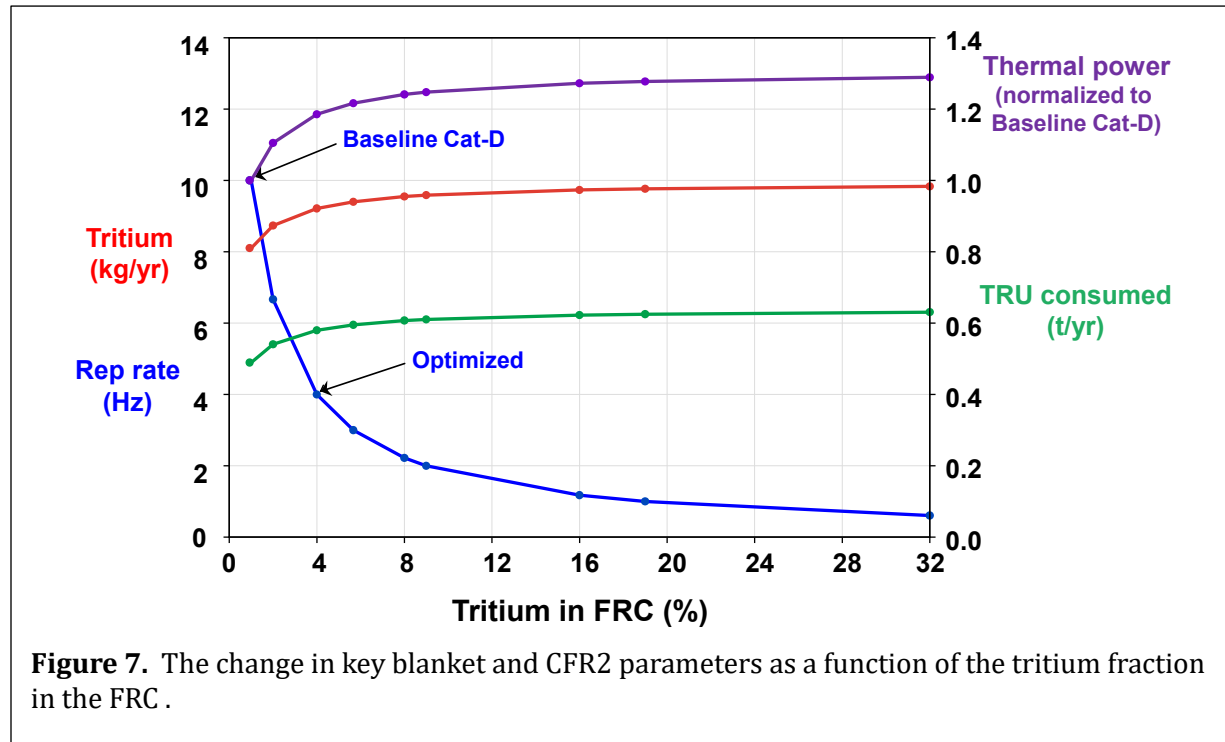
where

$$f_{\text{DT}} = 100f_{\text{T}} / (1 + 100f_{\text{T}}), \quad (4b)$$

and f_T denotes the plasma tritium fraction. This expression ranges from $E_{src} = 8.275$ MeV at the 1% T Cat-D baseline to $E_{src} = 13.86$ MeV at 50% T.

The blanket fission power at each point was obtained from the heating-local calculation scaled to the actual operating neutron rate. The results, plotted in Fig. 7 from the T-fraction sweep series, show that the blanket power increases monotonically with the tritium fraction, rising from 1,240 MWth at 1% T to approximately 1,608 MWth at a 50% T fraction. The increase is driven entirely by spectral hardening, where higher-energy D-T neutrons produce more prompt fast fissions in the TRU before thermalization, adding fission power above the thermalized baseline, which is independent of the source spectrum. The rep rate decreases from 10 Hz at 1% T to approximately 0.4 Hz at 50% T to maintain a constant total neutron rate, substantially reducing driver wear and the power electronics cycling frequency as the tritium fraction increases.

As shown in Fig. 7, all key blanket parameters increase with increasing fraction of tritium in the FRC. However, the increase rapidly flattens above 4%, with only the rep rate continuing to decrease. There would appear to be little reason to increase the tritium content beyond 4%. The disadvantage of a continually dropping rep rate is that the injection of neutrons with each pulse



becomes much larger and at a higher neutron energy, resulting in a large transient heating pulse for both the first wall as well as the blanket. It should be noted that this transient is small in absolute terms. At the optimized operating point the fission energy released per pulse raises the bulk blanket temperature by 3.4 °K and the first wall by 0.27 °K.

There are, however, good reasons for possibly wanting to increase the tritium content of the FRC, as this would allow for reducing the magnetic field and/or FRC flux required in the CFR2 while maintaining a higher rep rate to compensate for the reduced neutron flux per pulse. This method of compensation for a lower fusion gain, or equivalently, a lower FRC confinement performance than anticipated, can thus be addressed by a higher tritium fraction and a higher rep rate. Thus, it is possible to employ virtually any concentration of tritium in the FRC source up to 50%, while simultaneously increasing or at least maintaining the TRU burnup, power output, and tritium production of the blanket. This decoupling of the CFR2 source operation from the blanket output

dramatically reduces the risk of failure by providing a wide range of source operating conditions that have no downside on the operation of the fission system.

5.3 Upper Bound on Tritium Fraction

Four considerations bound the useful tritium fraction from above, and together they place the reference operating point at 4%. First, the gain in wall-plug efficiency is subject to diminishing returns. Raising the tritium fraction from 4% to 50% increases the net electrical output by 11.9%, but only 15.8 MWe of that 80 MWe gain comes from the reduction in driver power; the remaining four-fifths arises from the increase in blanket fission power. The recirculating fraction falls from 2.55% to 0.24%, a saving of 2.3% of net output, purchased at the cost of a ten-fold reduction in repetition rate.

Second, alpha heating of the FRC rises steeply with tritium fraction. Each D-T reaction deposits a 3.5 MeV alpha in the plasmoid, and at the 4% operating point the alpha energy released per pulse is 0.45 MJ against a 3.5 MJ plasmoid, or 12.8%. This fraction reaches 51% at 16% tritium and exceeds the plasmoid energy itself above approximately 32%, at which point the heating and expansion of the FRC would significantly reduce the FRC plasma density and thus reactivity ($\sim n^2$). This in turn would lower the neutron production rate which was the principle reason for the increase.

Third, the source spectrum hardens as the D-T fraction rises from 0.80 to 0.98 across this range, raising the first wall neutron loading from 6.67 to 7.86 MW/m² and the displacement damage rate proportionately, on a component whose replacement interval is already of order one year.

Fourth, fusion is not confined to the burn chamber. The FRC is hot enough to burn during the late compression stages and retains energy in the recovery chamber after ejection, and while the D-D yield from these regions is negligible, the D-T reactivity is larger by roughly a factor of fifty. At elevated tritium fractions a significant absolute neutron yield would therefore be produced outside the blanket, in regions that carry little to no shielding and house the compression and recovery hardware. Quantifying this requires the plasma trajectory through the compression sequence and is left to future work, but it argues independently for keeping the tritium fraction low.

6. Blanket Performance and Power Control

6.1 The Control Problem for a Fusion-Driven Subcritical System

The control challenge of a subcritical fission blanket driven by an external neutron source is qualitatively different from that of a critical fission reactor. In a critical reactor, the chain reaction is self-sustaining, and the power is regulated by adjusting neutron multiplication through control rods, coolant flow, or burnable poisons. In a subcritical system with $k_{\text{eff}} = 0.9695$, the fission chain amplifies the external source by $M = 1/(1 - k_{\text{eff}}) = 32.8$; however, the chain terminates when the source is removed. Therefore, power regulation is achieved primarily through the source, not through the blanket. The central operational question is: over what range of source conditions does the blanket respond predictably, and what is the quantitative relationship between the source parameters and blanket power?

For the CFR2, this question has practical urgency beyond routine power regulation. FRC plasmas are subject to confinement variability: the empirical confinement scaling developed in the companion paper [12] predicts a nominal operating point, but discharge-to-discharge variation and possible long-term confinement variation can change the fusion yield per pulse. The Cat-D operating mode offers several natural compensation mechanisms. One method involves altering the rep rate to directly reduce or increase the neutron flux on a sub-second timescale, thereby controlling the thermal power output from the blanket. Method two, modify the FRC magnetic parameters to increase or decrease the fusion gain. This would most likely be used for long-term variations in fusion gain. Finally, increasing the plasma tritium fraction above the Cat-D 1% T baseline as this hardens the source spectrum by shifting the neutron population from 2.45 MeV D-D

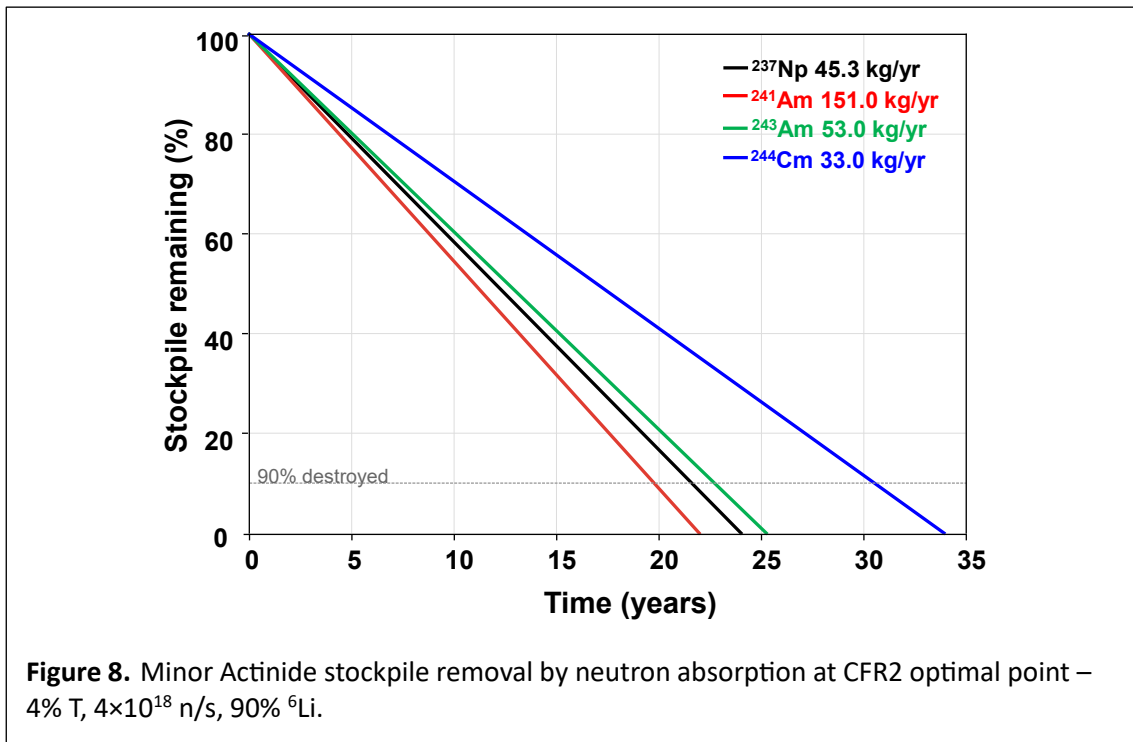
neutrons toward 14.1 MeV D-T neutrons, thereby increasing the mean source energy E_{src} without requiring any change in the driver power or rep rate.

The tritium-fraction range explored in this study extends to 50% in order to establish the validity of the control law over the full accessible span of mean source energy. It should not be read as a proposed operating regime. As set out in Section 5.3, alpha heating of the plasmoid, the hardening of the source spectrum at the first wall, and the onset of significant D-T yield outside the burn chamber together bound the practical tritium fraction well below that figure, and the confinement compensation described below should be understood as operating within roughly the 4% to 16% range rather than across the whole sweep.

6.2 Minor Actinide Burn-down Kinetics

The transmutation performance of CFR2 driven fission is not limited to the aggregate TRU destruction rate. The isotope-specific burn rates, determined from the OpenMC actinide reaction rate calculations and fission-to-absorption ratios in Section 3.2, allow for the time-dependent depletion of each MA species to be projected. Figure 8 shows the fractional stockpile remaining for the four principal minor actinide species as a function of time at the optimized operating point (4% T, 4 Hz, and 579 kg/yr total TRU burn).

The burn-down rates per module are: ^{237}Np at 45.3 kg/yr, ^{241}Am at 151.0 kg/yr, ^{243}Am at 53.0 kg/yr, and ^{244}Cm at 33.0 kg/yr. ^{241}Am is depleted most rapidly in absolute terms because it is the most abundant MA species in a LWR discharge, and its $f/a = 0.270$ at the 60.6 keV spectral peak is sufficient to drive substantial fission destruction. The depletion timescale for ^{241}Am is approximately 22 years per module at these burn rates, assuming a fixed feed composition that is representative of an aged LWR discharge. ^{237}Np , with $f/a = 0.336$, is destroyed at a somewhat lower absolute rate owing to its lower abundance in the feed, reaching 90% depletion in approximately 21 years. Curium isotopes, which pose the greatest long-term radiological concern owing to their high specific activities and neutron emission rates, are depleted on similar timescales.



All neutronics results reported in this paper are initial values, computed at the fixed blanket composition of Table 1. During operation the composition evolves: the transuranic inventory is depleted by fission, fission products accumulate, and radiative capture generates higher actinides. Each of these processes affects the neutron balance and hence the multiplication factor. In a liquid-fueled blanket these effects are compensated chemically rather than mechanically. The salt is circulated continuously through a processing loop that removes fission products and replenishes the transuranic inventory, and the feed rate is the control variable that maintains k_{eff} at its setpoint. No control rods or burnable poisons are required, and the subcritical architecture provides margin against the reactivity excursions that would otherwise accompany a drifting composition.

A quantitative demonstration of this compensation requires a coupled depletion and fission-product-removal calculation, tracking the isotopic inventory at a specified feed rate and separation efficiency over the operational period, as well as establishing the feed schedule that holds k_{eff} constant. Such a calculation is beyond the scope of the present study, which is directed at establishing the neutronics performance of the configuration. It is, however, identified in Section 7 as a principal item of further work required. It may be noted that as the most abundant and reactive minor actinide species are preferentially destroyed, the feed composition would be expected to evolve toward isotopes with higher fission thresholds. Whether this improves or degrades the transmutation efficiency over the campaign is precisely the question the depletion calculation would answer.

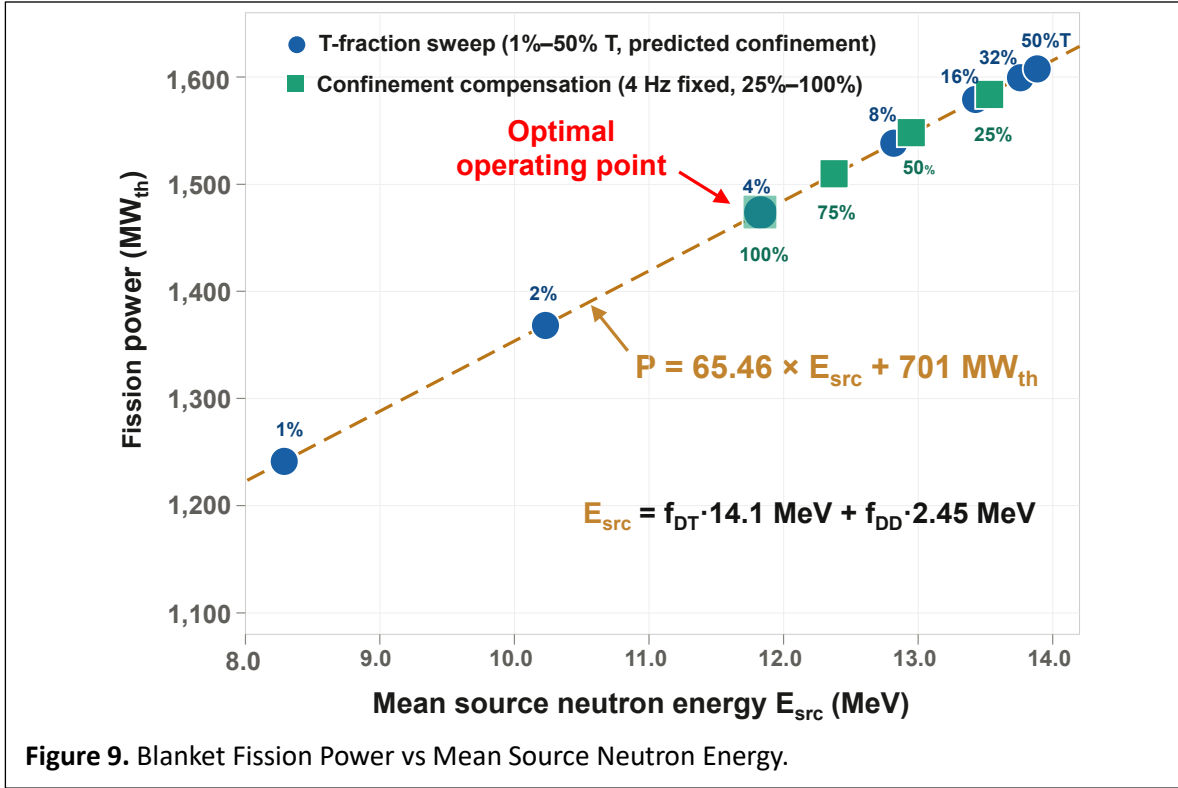
6.3 The Universal Linear Control Law

The most significant result of the combined T-fraction and confinement compensation sweeps emerges when the blanket fission power from all 11 OpenMC runs is plotted as a function of the mean source neutron energy, E_{src} , as shown in Fig. 9. Even though the two sweep series achieve their E_{src} values through entirely different physical mechanisms; one by increasing the plasma tritium a fraction at fixed FRC neutron flux per second, and the other by increasing tritium to compensate for degraded confinement at a fixed rep rate, the result was that all 11 data points from the OpenMC calculations fell on a single straight line with remarkable precision:

$$P = 65.46 \cdot E_{\text{src}} + 701.0 \text{ MW}_{\text{th}}, \quad (5)$$

with $R^2 = 0.9999$ across the full range from $E_{\text{src}} = 8.275 \text{ MeV}$ (1% T, 100% confinement) to $E_{\text{src}} = 13.86 \text{ MeV}$ (50% T). The physical interpretation of the two coefficients is clear and can be independently verified. The intercept of 701 MW_{th} represents the thermalized fission power. This is the contribution to blanket heating from neutrons that have been moderated below the fast fission threshold and drive the bulk of the fission chain through the epithermal and thermal cross-sections of the TRU inventory. This component is determined entirely by k_{eff} and the blanket geometry and is independent of the source spectrum. This value is consistent with the expected thermalized fission power at $k_{\text{eff}} = 0.9695$ for the v4b blanket geometry, confirming that the intercept is a property of the blanket rather than of the source itself.

The slope of 65.46 $\text{MW}_{\text{th}}/\text{MeV}$ represents the prompt fast fission contribution. This is due to the additional fission power generated by high-energy D-T neutrons striking the TRU nuclei before thermalization. Each additional MeV of mean source energy delivers approximately 65 MW of additional fission power above the thermalized baseline. The physical content of the slope is that the number of fissions per source neutron is itself linear in mean source energy, rising from 9.69 to 12.55 across the sweep. This slope is also a property of the blanket geometry and TRU composition, not of the source, which explains why both sweep series produce the same slope: the blanket responds only to the mean neutron energy presented to it, regardless of whether that energy was achieved by increasing the tritium fraction to lower the rep rate or by compensating for the confinement degradation. The universality of this relationship is stronger than an empirical fit. Fixed-source transport is exactly linear in source strength, so blanket power factorizes as $P(S, E_{\text{src}})$



= $(S/S_0) \cdot f(E_{src})$ identically rather than approximately, with $S_0 = 4 \times 10^{18}$ n/s the reference source rate, and the sweep determines the function f . The collapse of both series onto a single line is therefore a consequence of the linearity of the transport equation and not a coincidence.

Two distinct multiplication measures are used in this paper. The eigenvalue multiplication factor k_{eff} is the solution of the k -eigenvalue problem for the blanket and is a property of the medium and geometry alone; it does not depend on the external source, and for the reference blanket it is 0.9695 ± 0.0001 . The source multiplication k_s is inferred from the fixed-source calculation as $k_s = v/(1 + v)$, where v is the number of fission neutrons produced per source neutron; it characterizes the amplification of a particular external source by the blanket and therefore does depend on the source energy spectrum since higher-energy source neutrons induce more first-generation fast fission in the threshold-fissioning actinides. Across the full operating range explored in this study, k_s spans 0.9676 to 0.9748 while k_{eff} remains fixed at 0.9695. The system is subcritical on both measures at every operating point, and no change to the source can render it critical, since criticality is governed by k_{eff} , which the source does not influence.

The universality of the linear relationship, that is, the fact that both sweeps land on the same line, is an operationally important result. This implies that the blanket does not distinguish between the physical origins of E_{src} . The operator has a single, continuously adjustable control parameter (tritium fraction) that maps one-to-one onto the blanket output power through a known, linear law. The source multiplication across all operating points spans 0.9676–0.9748, confirming that subcriticality is maintained throughout the full operating envelope without a blanket adjustment. The system is always safely subcritical, regardless of how the source spectrum is modified to achieve power regulation, FRC pulse rate, or confinement compensation.

6.4 Implications for Reactor Control and Safety

The universal linear control law has three direct implications for the design of CFR2-driven transmutation systems. First, it provides a complete operational map for any target blanket power within an accessible range. The required tritium fraction can be calculated directly using Eq. (5),

and the operator can dial in that fraction through the FRC fueling system. No iterations, empirical tuning, or feedback from blanket neutronics are required. The control law was derived from first principles and confirmed by 11 independent OpenMC calculations.

Second, the law establishes that the blanket power is bounded above by the maximum achievable E_{src} . At 100% D-T operation ($f_T = 0.5$), $E_{src} = (14.1 \cdot f_{DT} + 2.45 \cdot f_{DD})$ MeV = 13.867 MeV. Using Eq. (5), the maximum blanket power is obtained is 1,608 MW_{th}, which is less than 9% higher than the optimized operating point. The blanket cannot be driven to supercritical power excursions through source spectrum manipulation alone because the maximum fast fission enhancement is physically limited by the D-T endpoint energy. This is an intrinsic safety characteristic of subcritical architectures. Because the multiplication $M = 1/(1 - k_{eff})$ is large, the sensitivity of blanket power to reactivity is correspondingly large: $dP/P = M \cdot dk$, so at $M \approx 33$ a change of 0.001 in k_{eff} moves the blanket power by 3.3%. Holding the thermal output to $\pm 5\%$ therefore requires the salt processing loop to hold k_{eff} to ± 0.0015 . This is a requirement on the feed and separation control system rather than on the source, and it complements the source-side control law described above.

Third, and the most directly relevant to long-term operations, the FRC confinement compensation result confirms that the CFR2 transmutation system is robust to FRC performance variability at a scale that might realistically be encountered during extended operation. The confinement scaling uncertainties discussed in [12] span roughly a factor of two to three in confinement time, depending on the assumptions regarding high x_s transport. The compensation sweep demonstrates that the blanket output is maintained across the full span of the uncertainty range without any change in the blanket design, driver power, or k_{eff} management strategy. The fusion source and fission blanket are effectively decoupled in terms of their operational requirements, with the tritium fraction serving as the single coupling parameter between them.

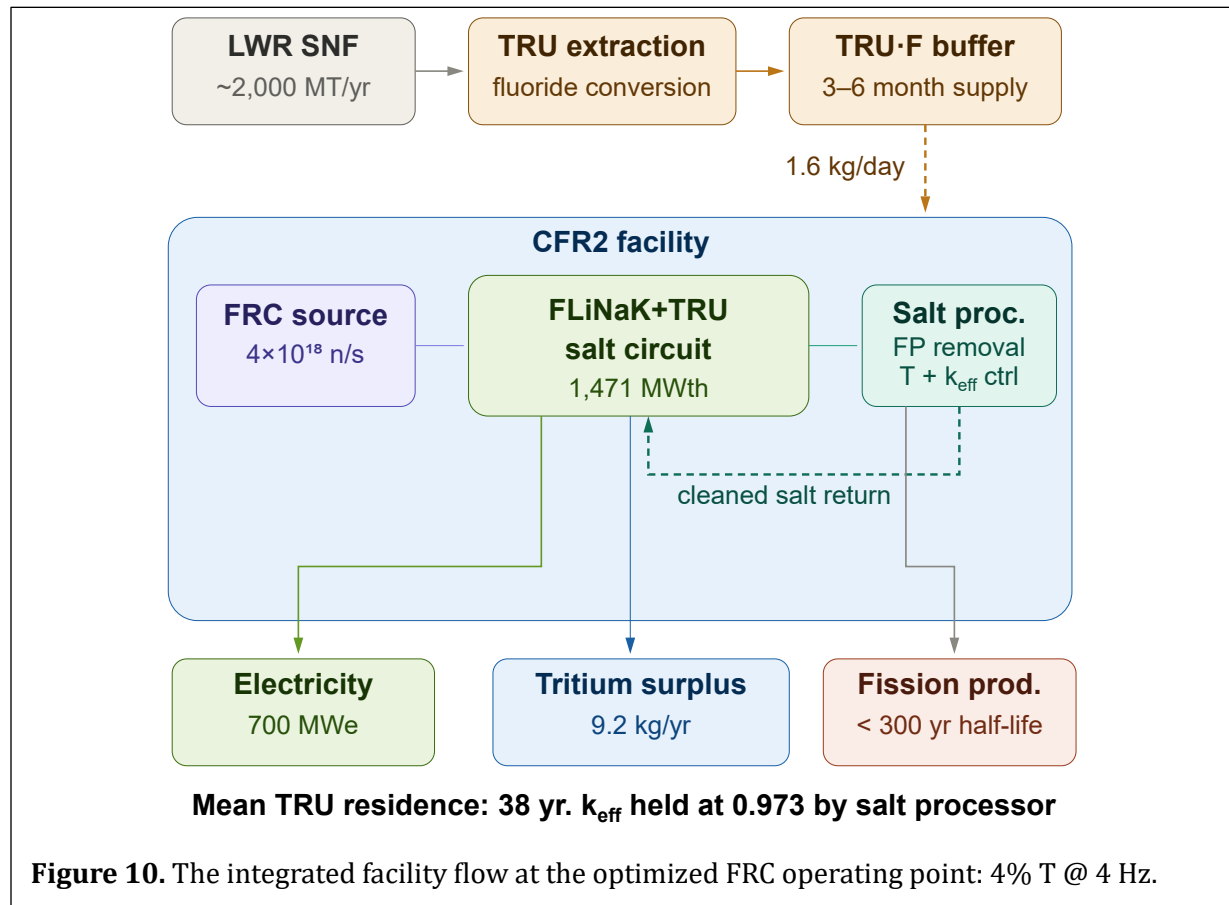


Figure 10 illustrates the integrated facility flow for the optimized operating point: LWR spent fuel enters as TRU fluoride feed at 1.6 kg/day, the FLiNaK salt circuit circulates through the blanket producing 1,471 MW_{th} of fission power, and the salt processing side-stream simultaneously removes fission products, controls k_{eff} through TRU feed adjustment, and recovers a tritium surplus of approximately 9.2 kg/yr for external delivery. The mean TRU residence time in the blanket under these conditions is approximately 38 years, after which the TRU inventory has been reduced to short-lived fission products with a maximum hazard timescale below 300 years. The electricity output of 700 MWe (at 47.5% thermal conversion efficiency), tritium surplus of \$920M/yr together with the avoided TRU disposal cost of approximately \$579M/yr provide a total economic output of approximately \$1.8B/yr per module, the breakdown of which is discussed further in Section 7.

The mean transuranic residence time quoted here is the ratio of the transuranic inventory held in the blanket to the rate at which transuranics are destroyed, $\tau = I / \dot{M}$. From the composition of Table 1, the blanket contains 22.3 tonnes of transuranics in its 13.41 m³ of active volume. At the optimized operating point these are destroyed at a rate of 579 kg/yr, inferring that $\tau \sim 38$ years. This is a mean residence time under continuous feed and withdrawal and not a batch irradiation time. Individual nuclides reside for longer or shorter periods according to their fission-to-absorption ratios, and the figure does not by itself establish the time required to reduce a given stockpile.

7. Summary and Discussion

7.1 Summary of Principal Results

This study presents a systematic Monte Carlo neutronics analysis of a subcritical actinide transmutation blanket driven by a CFR2 catalyzed-deuterium fusion neutron source. The blanket employs FLiNaK eutectic fluoride salt carrying 17.53 mol% TRU fluorides as the combined coolant, neutron moderator, and fissile medium, with the lithium component enriched to 90% ⁶Li. All OpenMC calculations used ENDF/B-VIII.0 nuclear data, and the computational methodology was validated against the independently reported In-Zinerator subcritical system before being applied to the CFR2 geometry.

The optimized reference design achieves $k_{\text{eff}} = 0.9695$ at the optimized operating point of 4% plasma tritium fraction and 4 Hz repetition rate. The blanket produces 1,471 MW_{th} of fission power, burns 579 kg/yr of TRU, and exhibits a characteristic epithermal neutron spectrum that peaks at 60.6 keV. The total system outer diameter is 3.90 m, and the active blanket volume is 13.4 m³, which is approximately one-seventh that of the In-Zinerator reference system at comparable k_{eff} and TRU destruction rates. The fission-to-absorption ratios at the spectral peak are 0.336 for ²³⁷Np and 0.270 for ²⁴¹Am. The net tritium production at the optimized point was 9.2 kg/yr, with a TBR of 18.9. The CFR2 driver consumes 17.5 MWe, representing a recirculating fraction of 2.5% against a gross electrical output of 700 MWe. A universal linear control law, $P = 65.46 \cdot E_{\text{src}} + 701$ MW_{th}, was confirmed across 11 independent OpenMC fixed-source calculations with $R^2 = 0.9999$, spanning two independent parameter sweeps using entirely different physical mechanisms.

7.2 Discussion

The ⁶Li enrichment sweep in Section 4 produced a result whose design implications extended beyond the immediate parameter choice. In reactor blanket design, it is common for competing objectives to impose tradeoffs; a change that improves one metric typically degrades another, and the design optimum is found at a compromise point. The ⁶Li sweep revealed that this familiar structure was entirely absent in the CFR2 FLiNaK system. Every performance metric of interest, namely the fission-to-absorption ratios of ²³⁷Np and ²⁴¹Am, tritium breeding ratio, and total fission power, improved monotonically as ⁶Li enrichment increased from 1% to 90%, with the global optimum lying at the highest enrichment that is practically achievable.

This outcome arises from the triple role of ${}^6\text{Li}$, as discussed in Section 4.1. The three functions, neutron poisoning to control k_{eff} , tritium production via the ${}^6\text{Li}(n,t)\alpha$ reaction, and epithermal spectrum shaping through the interplay of three processes. Source neutrons at 2.45 and 14.1 MeV are degraded in energy principally by inelastic scattering on ${}^{19}\text{F}$, which constitutes approximately half the atoms of the carrier salt and has inelastic levels from 110 keV upward; nuclide-resolved data show that fluorine accounts for 53–93% of all inelastic reactions in the blanket. Below the fluorine inelastic threshold, elastic scattering on the light constituents continues the slowing-down. Below approximately 1 keV, ${}^6\text{Li}$ absorption, whose cross section rises as $1/v$ toward lower energies, removes neutrons faster than supplied from above, truncating the spectrum. The position of the peak is set by the fluoride carrier and is found to be independent of ${}^6\text{Li}$ enrichment, remaining at 60.6 keV. What ${}^6\text{Li}$ enrichment controls is the flux on either side of the peak. As enrichment increases from 1% to 90%, the fraction of the flux below 1 keV falls from 5.55% to 0.053%, a suppression of two orders of magnitude, while the fraction above 100 keV rises from 37.0% to 60.7%. Both changes favor the destruction of minor actinides by fission: the first removes flux from the region in which radiative capture dominates fission for ${}^{237}\text{Np}$ and ${}^{241}\text{Am}$, and the second populates the region above their fast fission thresholds. The total TRU destruction rate and fission power remained essentially constant across the sweep. Both are set by k_{eff} and the source strength, while the quality of the transmutation improves. The result is that there is no penalty for employing the highest achievable ${}^6\text{Li}$ enrichment, and the concern that high enrichment might degrade energy multiplication or drive excessive parasitic absorption is not realized in this system.

Although this study has been framed around minor actinide transmutation, the transuranic vector of Table 1 is 65% plutonium by mass, and the blanket destroys plutonium considerably more effectively than it destroys the minor actinides. The mass-weighted fission-to-absorption ratio of the plutonium isotopes is 0.80, against 0.37 for the minor actinide fraction, so four of every five plutonium absorptions terminate in fission rather than in transmutation to a heavier species. The consequence is that 80% of the transuranic mass destroyed by a single module, 461 kg/yr of the 576 kg/yr total, is plutonium, and its mean residence time against fission is 31.5 years against 67.7 years for the minor actinides and 38.7 years for the transuranic inventory as a whole.

The isotopic composition of the surviving plutonium also evolves unfavorably for any diversionary purpose. The 15% of ${}^{239}\text{Pu}$ absorptions that terminate in capture rather than fission convert it to ${}^{240}\text{Pu}$, and because ${}^{239}\text{Pu}$ is removed at 4.8% per year compared to 1.1% for ${}^{240}\text{Pu}$, the ${}^{240}\text{Pu}/{}^{239}\text{Pu}$ ratio rises by approximately 4% per year of operation. The material is held throughout in a circulating fluoride salt which carries the full fission product inventory. This, together with the spontaneous, neutron-emitting curium isotopes, is self-protecting without requiring the fabrication, transport and storage of discrete fuel assemblies.

At fleet scale the numbers are consequential. The fourteen-module fleet required to close the United States light water reactor fuel cycle would destroy approximately 6.5 tonnes of plutonium per year while generating some 9.7 GWe of net electrical output. For comparison, the United States has undertaken the task of disposing excess weapons grade plutonium on the order of 34 tonnes. The fleet would consume this in approximately five years as an incidental consequence of normal operation rather than as a dedicated and separately funded mission.

This result is a specific consequence of the FLiNaK+TRU geometry and subcritical operating regime and would not be applicable to a critical fast reactor. In a critical system, adding a strong thermal absorber, such as ${}^6\text{Li}$, would impose a reactivity penalty, requiring compensating fissile additions that would compromise the neutron economy. In the CFR2 subcritical system, k_{eff} is maintained by adjusting the TRU loading independently of the source, and the source strength is also controlled independently. This operational freedom is precisely what allows ${}^6\text{Li}$ to optimize all the design objectives simultaneously without conflict.

7.2.1 Electricity, Waste Transmutation and Tritium Production - Economic Implications

The gross electrical output at the optimized operating point was determined using the fission thermal power of 1,471 MWth and the thermal conversion efficiency of the FLiNaK salt circuit. FLiNaK operates well above 700°C, placing it in the range of advanced high-temperature thermodynamic cycles, including supercritical CO₂ and helium Brayton cycles. At a conversion efficiency of 47.5%, which is consistent with the performance of such cycles at comparable temperature levels, the gross electrical output is 700 MWe. The CFR2 driver consumes 17.5 MWe, corresponding to a recirculating fraction of 2.5%, which is negligible compared to the electrical output. At a wholesale electricity price of \$50/MWh and a capacity factor of 0.90, this yields an annual electricity revenue of approximately \$276M/yr per module.

The second revenue stream arises from the avoided costs of TRU disposal. Under the current U.S. policy, the long-term liability associated with transuranic waste from commercial LWR operations has no settled market price because no permanent repository has accepted commercial spent fuel. A conservative floor estimate of \$1M per kilogram. This is derived from the projected lifecycle costs of repository siting, construction, long-term monitoring, and retrievability requirements apportioned across the projected TRU inventory, and yields an avoided disposal value of approximately \$579M per year for the 579 kg/yr of TRU destroyed at the optimized operating point. This estimate is intentionally conservative because cost projections for permanent repository facilities have increased with each successive revision of the Yucca Mountain program. In addition, the avoided liability per kilogram of TRU is likely to increase with more stringent regulatory requirements.

The third revenue stream is the tritium production. The CFR2 FLiNaK blanket at 90% ⁶Li enrichment and $k_{eff} = 0.97$ produces tritium at 25.6 times its own consumption rate at the Cat-D baseline, which is a surplus so large that the system's role as a tritium factory is arguably as consequential as its role as a transmuter or an electricity producer. The physical origin is the fission multiplication factor $M \approx 33$, which amplifies the ⁶Li(n,t)α reaction rate far beyond what the bare fusion source could drive. The fusion source is not consumed in producing this surplus; it is the seed of an avalanche of neutron multiplication events, each carrying a probability of generating tritons proportional to the ⁶Li atom density.

The strategic significance is most clearly stated in relation to the global tritium inventory. With a single D-T fusion power plant requiring roughly 50 kg for the initial startup inventory, in addition to a continuous supply to replace the burn and decay losses; the global tritium availability from existing sources would severely limit testing, development and eventually the birth rate of the D-T fusion fleet. A 14-module CFR2 fleet producing ~120 kg of tritium surplus per year would generate sufficient new inventory to provide startup fuel for two new D-T fusion plants per year while simultaneously closing the U.S. LWR spent fuel cycle. Therefore, the CFR2 fleet serves as the essential bridge across which fusion deployment must pass to achieve self-sustaining tritium breeding in a fully deployed commercial fleet, and it performs this function while providing waste management and electricity generation services that are independently necessary and economically valuable. The aggregate annual economic output per module from the three revenue streams would thus total 1.8B\$/yr.

7.2.2 Further Work

Three items of supporting work are identified as prerequisites to advancing the hybrid fusion-fission concept presented here. First, a coupled depletion and fission-product-removal calculation is required to demonstrate quantitatively that the transuranic feed rate can hold k_{eff} at its setpoint over an operational lifetime as described in Section 6.2. Second, direct measurement of NpF₃ and CmF₃ solubility in ⁶Li-enriched FLiNaK would close the remaining gap in the solubility basis of Section 2.2. Third, qualification of W-25Re or other candidate first wall materials under combined

pulsed magnetic and neutron loading at the damage and transmutation rates given in Section 2.3.4 is required to establish the first wall replacement interval.

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