SUB-CRITICAL TRANSMUTATION REACTORS WITH TOKAMAK FUSION NEUTRON SOURCES BASED ON ITER PHYSICS AND TECHNOLOGY

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A series of design scoping and fuel cycle studies for sub-critical fast transmutation reactors driven by tokamak fusion neutron sources has been carried out to determine if the requirements on the tokamak neutron sources are compatible with the fusion physics and technology design database that will exist after the operation of ITER and to determine if there is a significant advantage in fuel cycle flexibility due to sub-critical operation that would justify the additional cost and complexity of a fusion neutron source. The fast reactor technologies are based on reactor concepts being developed in the DoE Generation-IV and Advanced Fuel Cycle initiatives.

I. INTRODUCTION

For many years there has been a substantial international R&D activity devoted to closing the nuclear fuel cycle. During the 1990s this activity emphasized the technical evaluation of transmutation reactors that would fission the transuranic (TRU) content of the accumulating spent nuclear fuel (SNF) discharged from conventional nuclear power reactors[1-4], thus reducing the requirements for long-term geological high-level waste repositories (HLWRs) for the storage of SNF. With the recently increasing recognition that nuclear power is the only environmentally sustainable way to meet the world's expanding energy requirements in the near term, the emphasis in the new century has broadened to also include extracting more of the potential energy content in uranium by first transmuting the "fertile" ^{238}U into fissionable ^{239}Pu. This growing realization of the need for an expanded global role for nuclear power has led to a number of government policy inititatives aimed at closing the nuclear fuel cycle—the Advanced Fuel Cycle Initiative (AFCI), the Generation-IV Initiative (GEN-IV) and most recently the Global Nuclear Energy Partnership (GNEP).

There would be advantages in being able to operate the transmutation reactors sub-critical, with a neutron source to provide the neutrons needed to maintain the fission chain reaction, e.g. the achievement of higher levels of burnup for a given batch of TRU fuel. Almost all of the studies in the 1990s of sub-critical transmutation reactors were based on use a proton accelerator with a spallation target as a neutron source, although there were a few studies of the use of D-T fusion neutron sources.

The concept of using a D-T tokamak fusion neutron source based on ITER physics and technology[5] to drive a sub-critical fast transmutation reactor based on nuclear and separations technologies being evaluated in the AFCI[6] and GEN-IV[7] initiatives has been developed in a series of studies[8-19] at Georgia Tech over the past several years. The general design objective was a 3000 MWth, passively safe, sub-critical fast reactor driven by a fusion neutron source that could fission the TRU in the SNF discharged annually by three 1000 MWe LWRs. The general fuel cycle objective was > 90% burnup of this TRU (in order to reduce the HLWR requirements by an order of magnitude relative to the present once-through LWR fuel cycle) while minimizing the nuclear fuel reprocessing steps. The designs were constrained to use ITER physics and technology for the fusion neutron source, to use nuclear and reprocessing technology being evaluated in the GEN-IV and AFCI studies, to use extensions of existing nuclear fuel technology but with TRU, and to achieve tritium self-sufficiency for the fusion neutron source.

II. THE FTWR AND GCFTR STUDIES

Sub-critical transmutation reactors based on two of the nuclear technologies being developed in the GEN-IV studies have been examined in the Georgia Tech studies. The Fusion Transmutation of Waste Reactor (FTWR) series of studies was based on a variant of the Lead Cooled Fast Reactor and the Sodium Cooled Fast Reactor in the GEN-IV initiative--a fast-spectrum reactor using a metal fuel consisting of TRU alloyed with zirconium in a zirconium matrix and cooled by a liquid metal (Li17Pb83 eutectic), which also served as the tritium breeder. The Gas Cooled Fast Transmutation Reactor (GCFTR) series of studies was based on a variant of the Gas Cooled Fast Reactor in the GEN-IV initiative--a fast-spectrum reactor using TRU-oxide fuel in coated TRISO particle form in a SiC matrix cooled by He. Both the FTWR and GCFTR

cores are annular and located outboard of the toroidal plasma chamber. The core plus plasma chamber were surrounded first by a reflector (tritium breeding blanket for GCFTR) and then by a shield to protect the magnets from radiation damage and heating, as indicated in Fig. 1 for the GCFTR design.

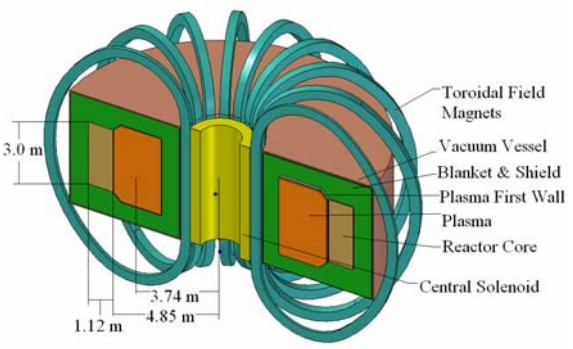

Fig. 1 Gas Cooled Fast Transmutation Reactor

The ANL metal fuel, liquid metal cooled reactor design[20] was adapted to accommodate Pb-Li eutectic coolant and TRU-Zr fuel for the FTWR designs. The fast, He-cooled reactor designs being developed under the GEN-IV Program guided the choice of the GCFTR core design, and the coated fuel particle technology being developed in the NGNP program[21] was adapted for TRU-oxide fuel for the GCFTR. Tritium breeding was accomplished in the Pb-Li coolant in the FTWR designs and in a Li_2O blanket surrounding the plasma and reactor core in the GCFTR designs.

The fusion technology was based on the ITER design[5,22]. The superconducting magnet design was based directly on the ITER cable-in-conduit design scaled down to maintain the same stress level. The first-wall and divertor designs were adapted from the ITER design to accommodate the different coolants. A LHR heating and current drive system was adapted from ITER.

The radial build dimensions of the FTWR and GCFTR concepts were determined from the engineering and physics constraints[22] and are given in Table I.

III. TOKAMAK NEUTRON SOURCE

The principal tokamak neutron source parameters for the FTWR and GCFTR series of transmutation reactors are given in Table II. The requirements on β_N and confinement are within the range routinely achieved in present experiments, and the requirements on β_N, confinement, energy amplification Q_p, and fusion power level are at or below the ITER level, except for the FTWR-AT and GCFTR-3 design concepts. The requirement on the current-drive efficiency, after calculation of bootstrap current fraction using ITER scaling, is only somewhat beyond what has been achieved to date (γ_{CD} = 0.45 in JET and 0.35 in JT60-U). The ongoing worldwide tokamak program is addressing the current-drive/bootstrap current/steady-state physics issue. The current-drive efficiency/bootstrap fraction needed for FTWR/GCFTR is certainly within the range envisioned for Advanced Tokamak operation and may be achieved in ITER. Although single numbers are given for the parameters in Table II, there is a range of operating parameters within which a given neutron source intensity (fusion power level) can be achieved, as shown in Fig. 2 for a 7.2MA design that can achieve P_{fus} = 200 MW.

TABLE I Dimensions (m) of FTWR and GCFTR Designs

Parameter	FTWR	FTWR-SC	FTWR-A	GCFTR	GCFTR-2	GCFTR-3
Major Radius, R_0	3.10	4.50	3.86	4.15	3.74	3.76
Fluxcore, R_{fc}	1.24	1.10	0.65	0.66	0.66	0.88
CS+TF, Δ_{mag}	0.57	1.68	1.20	1.50	1.13	0.91
Refl+Shld, Δ_{rs}	0.40	0.65	0.90	0.86	0.87	0.89
Plasma, a_{plasma}	0.89	0.90	1.10	1.04	1.08	1.08
Core						
Inner Radius, R_{in}	4.00	5.40	5.00	5.25	4.84	4.85
Radial Width, W	0.40	0.40	0.40	1.12	1.12	1.12
Height, H	2.28	2.28	2.28	3.00	3.00	3.00

TABLE II Tokamak Neutron Source Parameters for FTWR and GCFTR Transmutation Reactors[a]

Parameter	FTWR[9]	FTWR -SC[10]	FTWR -AT[11]	GCFTR[12]	GCFTR -2[13]	GCFTR -3[14]	ITER[5]
Max. Fusion power, P_{fus} (MW)	150	225	500	180	180	500	410
Max.Neutron source, $S_{fus}(10^{19}$ #/s)	5.3	8.0	17.6	7.1	7.1	17.6	14.4
Major radius, R (m)	3.1	4.5	3.9	4.2	3.7	3.7	6.2
Aspect ratio, A	3.5	5.0	3.5	4.0	3.4	3.4	3.1
Elongation, κ	1.7	1.8	1.7	1.7	1.7	1.7	1.8
Current, I (MA)	7.0	6.0	8.0	7.2	8.3	10.0	15.0
Magnetic field, B (T)	6.1	7.5	5.7	6.3	5.7	5.9	5.3
Safety factor, q_{95}	3.0	3.1	3.0	3.0	3.0	4.0	
Confinement, $H_{IPB98}(y,2)$	1.1	1.0	1.5	1.0	1.0	1.06	1.0
Max. Normalized beta, β_N	2.5	2.5	4.0	2.0	2.0	2.85	1.8
Max. Plasma Power Mult., Q_p	2.0	2.0	4.0	2.9	3.1	5.1	10
CD efficiency,γ_{cd} $(10^{-20}$ A/Wm$^2)$	0.37	0.23	0.04	0.5	0.61	0.58	
Bootstrap current fraction, f_{bs}	0.40	0.50	≥0.90	0.35	0.31	0.26	
Max. Neut. Flux, Γ_n (MW/m^2)	0.8	0.8	1.7	0.9	0.6	1.8	0.5
Max. FW Heat flux, q_{fw} MW/m^2)	0.34	0.29	0.5	0.23	0.23	0.65	0.15
Availability (%)	≥ 50	≥ 50	≥ 50	≥ 50	≥ 50	≥50	

[a] Calculated on the basis of the physics and engineering constraints described in Ref. 22. All superconducting except FTWR. All based on ITER physics except AT.

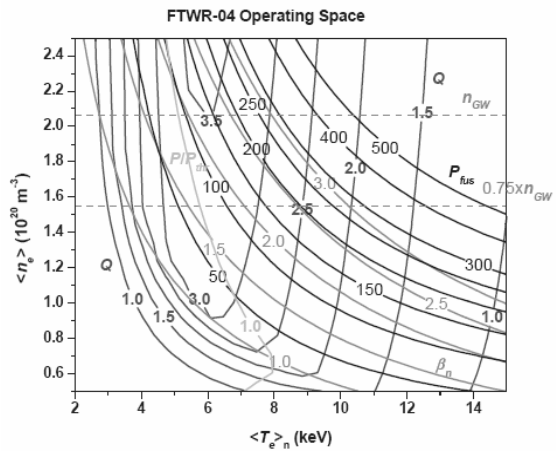

Fig. 2 Operating Space for 7.2 MA Neutron Source that Achieves up to P_{fus} = 200 MW

IV. TRANSMUTATION REACTOR CORES

IV.A. FTWR

The fuel is a transuranic zirconium alloy (TRU-10Zr) dispersed in a zirconium matrix in pin form and clad with a ferritic steel similar to HT-9. The relative amounts of transuranics and zirconium in the fuel region are adjusted to achieve the desired neutron multiplication (k_{eff} = 0.95) at the beginning of each burn cycle. At equilibrium, the transuranics will constitute approximately 45% of the fuel volume. The annular transmutation reactor core is outboard of the plasma, and both are surrounded by reflector and shield (Fig. 1). The design of the FTWR transmutation reactor is based on the ANL ATW reactor design studies[20]. The same pin and assembly geometry was used, with the exception that the length of the assembly was increased to 228 cm. Table III gives the basic data for the fuel assembly design. The reactor core is 40 cm thick and consists of 470 assemblies, 1/5 of which will be 'half assemblies' placed in the gaps along the interior and exterior surfaces of the reactor region to produce a more uniform annular distribution.

For 3000 MW_{th} total reactor power uniformly distributed in the fuel pins, the volumetric heat source is q''' = 42.2 MW/m^3. The main coolant parameters are given in Table III. The required pumping power is 130 MW, the majority of which is needed to overcome MHD losses.

IV.B. GCFTR

Design concepts were developed for several TRISO (tri-material isotropic) and BISO (bi-material isotropic) particles. The reference TRISO particle (Fig. 3) has a TRU-oxide kernel (330 μm diameter) surrounded by a 50% porous buffer layer (73 μm) of ZrC to allow for fission product recoil and gas buildup. Next is an inner WC layer (10 μm), then a SiC structural layer (67 μm), and finally an outer WC layer (15 μm). These particles are embedded in a SiC matrix, then formed into a fuel pin clad with ODS steel, as indicated in Table III. Nominal thermal parameters are given in Table III. A He coolant v/o ≥

25% would be adequate for heat removal under normal operating conditions.

A cross-section of the fuel assembly for the GCFTR is shown in Fig. 4, and the materials composition for the two reactor types are given in Table IV.

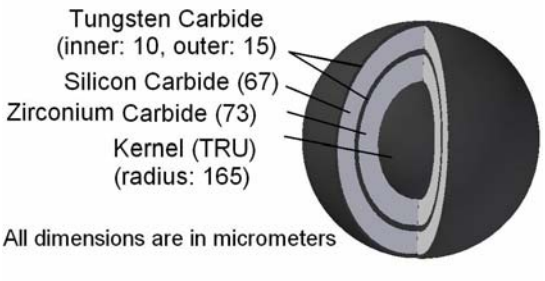

Fig. 3 TRISO Fuel Particle

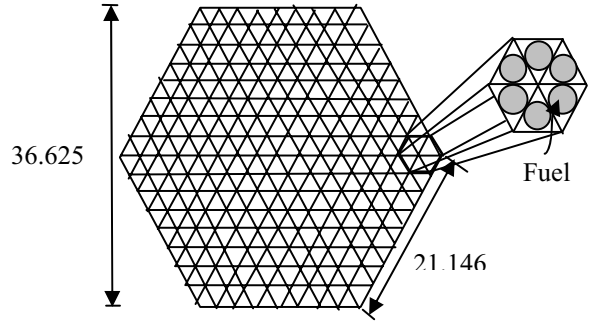

Fig. 4 Fuel Assembly for GCFTR

TABLE III Core and Fuel Assembly Parameters

		FTWR	GCFTR
Core in radius, cm		500	485
Core width, cm		40	112
Core height, cm		228	300
Pin Diameter, cm		0.635	1.526
Pins per assembly		217	384
Assy. Flat to Flat,cm		16.1	36.6
Assy. Length, cm		228	300
Assemblies in Core		470	245
Core Cool Flow,kg/s		51630	3280
Coolant T_{in}/T_{out} , ^{o}K		548/848	553/767
Materials (v/o)	Fuel	20	60
	Structure	10	10
	Coolant	70	30

TABLE IV Materials Compositon of FTWR and GCFTR

Component	FTWR	GCFTR
Reactor		
Fuel	TRU-Zr metal in Zr matrix	TRU-oxide TRISO,SiC matrix
Clad/structure	FeS/FeS	ODS/ODS
Coolant	LiPb	He
Trit. Breeder	LiPb	Li_2O
Reflector	FeS, LiPb	ODS, He, Li_2O
Shield	FeS, LiPb, B_4C, ZrD_2, W	ODS, HfC, Ir, Cd, WC, B_4C, He
Magnets	NbSn,NbTi/He (OFHC/LN_2)	NbSn/He
First-Wall	Be-coated FeS, LiPb	Be-coated ODS, He
Divertor	W-tiles on Cu-CuCrZr, LiPb	W-tiles on Cu – CuCrZr, He

A direct Brayton cycle would be used to convert the 3000 MWt to 1020 MWe. Taking into account power requirements to run the GCFTR, the net electrical power produced would be ≈ 700 MWe.

VI. FUEL CYCLE ANALYSIS

The great advantage of sub-critical operation is the variety of transmutation fuel cycles that it makes available, some of which are examined in this section. The composition changes in the fuel cycle were calculated with the REBUS fuel cycle code[23] and the ORIGEN-S burnup code[24].

V.A. FTWR

In the FTWR reference fuel cycle[15] the fuel will remain in the reactor for 5 burn cycles of 564 days each and then be reprocessed, blended with 'fresh' SNF and fabricated into new fuel elements for re-insertion into a FTWR. The fuel will be “shuffled” to a new location in the reactor after each burn cycle and removed for reprocessing after the fifth burn cycle.

A first generation FTWR operating at 3000 MWt will process approximately 74 MT of transuranics from LWR SNF, of which approximately 56% will be fissioned, 0.2% will be lost to the waste streams, and 44% will be recycled in a second generation FTWR. The second and subsequent generations of FTWRs will use the fuel from the previous generation FTWRs and therefore operate in the equilibrium mode shown in Table V

over their entire life. BOC and EOC refer to beginning and end of cycle.

TABLE V: An Equilibrium 5-Batch Reprocessing Fuel Cycle for 3000 MWt TRU Fueled FTWR (23 MT Initial TRU Load)[15]

Burn cycle, d	564
5-batch residence, y	7.7
TRU burn/residence, %	29
SNF disposed, MT/yr	101
Fast fluence/residence, 10^{23} n/cm^2	3.4
BOC k_{eff}	0.925
EOC k_{eff}	0.836
BOC P_{fus}, MW	61
EOC P_{fus}, MW	150

Repeated recycling of the discharged transuranics from FTWRs in successive generations of FTWRs will ultimately result in the destruction of up to 99.4% of the transuranics discharged from LWRs. At equilibrium, each 3000 MWt FTWR would be able to process the TRU discharged from three 3000 MWt conventional LWRs, so that it is possible to envision a fleet of conventional and transmutation reactors in the thermal power ratio 3/1.

V.B. GCFTR

A similar reprocessing fuel cycle was developed for the GCFTR, as indicated in Table VI. However, the emphasis in the GCFTR studies was achieving > 90% burnup of the TRU in the coated fuel particles *without reprocessing* and then removing the > 90% depleted fuel from the reactor and directly depositing it in a HLW repository. Leaving the highly depleted fuel, together with the accumulated fission products in the reactor long enough to achieve such deep burnup would lead to a much less reactive core (e.g. lower multiplication factor, k). The results in Table VI are indicative of the burnup (about 15%) that can be achieved without reprocessing and with $P_{fus} \leq 200$ MW.

TABLE VI: An Equilibrium 5-Batch Re-processing Fuel Cycle for 3000 MWt TRU Fueled GCFTR (37 MT Initial TRU Load)[17]

Burn cycle, d	376
5-batch residence, y	5.2
TRU burn/residence, %	15.3
SNF disposed, MT/yr	98
Fast fluence/residence, 10^{22} n/cm^2	3.9
BOC k_{eff}	0.936
EOC k_{eff}	0.900
BOC P_{fus}, MW	122
EOC P_{fus}, MW	199

By increasing the limit on the fusion neutron source from 200-500 MW [$P_{fis} - k_{eff} P_{fus} / (1 - k_{eff})$][25], it is possible to extend the allowable reactivity decrement due to burnup and accumulation of fission products, hence to increase the length of the burn cycle. Several 5-batch, "once-through", non-reprocessing fuel cycles in which the reactivity decrement associated with fuel burnup was compensated by an increase in neutron source strength to obtain a longer burn cycle length are summarized in Table VII. A 400 MW fusion neutron source enables achievement of a 5-batch, 2400 day burn cycle, fuel cycle in a 3000 MWt GCFTR, which is sufficient to obtain > 90 % TRU burnup without reprocessing.

TABLE VII: Once-Through Steady-State 5-Batch Non-Reprocessing Fuel Cycles for 3000 MWt TRU Fueled GCFTR (37 MT Initial TRU Load)[14]

Burn cycle, d	600	1200	1800	2400
5-batch residence, y	8.2	16.4	24.7	32.9
TRU burnup, %	24.9	49.7	72.4	93.7
SNF disposed, MT/yr	101	101	98	95
Fast Fluence, 10^{23} n/cm^2	0.7	1.3	3.0	4.3
BOC k_{eff}	0.987	0.917	0.856	0.671
EOC k_{eff}	0.927	0.815	0.714	0.611
BOC P_{fus}, MW	13	83	144	329
EOC P_{fus}, MW	73	185	286	389

More efficient utilization of the energy content of uranium not only requires that the TRU in SNF discharged from conventional LWR reactors be recovered and fissioned, but that some significant fraction of the > 99% of natural uranium that is ^{238}U be transmuted to ^{239}Pu and subsequently fissioned. Two possible steady state fuel cycles for a GCFTR fueled with a mixture of 70% ^{238}U and 30% TRU in oxide form are shown in Table VIII. A 3000 MWt GCFTR with a 500 MWt fusion neutron source could achieve > 75% utilization of the energy content of uranium (as compared to the present < 1%).

TABLE VIII: Once-Through Steady-State 5-Batch Non-Reprocessing Fuel Cycles for 3000 MWt 30%TRU-70% ^{238}U Fueled GCFTR (37 MT Initial TRU + ^{238}U Load)[14]

Parameter		
Burn cycle, d	600	1800
5-batch residence, y	8.2	24.7
TRU+ ^{238}U burnup, %	24.9	72.4
Fast fluence, 10^{23} n/cm^2	1.2	4.4
BOC k_{eff}	0.590	0.577
EOC k_{eff}	0.576	0.534
BOC P_{fus}, MW	410	423
EOC P_{fus}, MW	424	466

VI. TRITIUM SELF-SUFFICIENCY

Tritium accumulation calculations were performed in order to insure that the amount of tritium produced during operation is enough for the plasma to be self-sufficient. In the FTWR the Li-PB coolant was also the tritium breeder. For GCFTR, a tritium breeding blanket about 15 cm thick surrounded the plasma chamber and reactor core. Lithium oxide (Li_2O) was chosen as a representative form for the lithium, although hydroxide formation problems may require another form (e.g. lithium silicate or titinate). On-line extraction of tritium from Li_2O requires operation between 400 ºC and 800 ºC. Below 400 ºC the rate of tritium diffusion out of the individual grains of Li_2O is too slow, and above 800 ºC the particles swell and seal off porous channels through which the tritium must percolate to reach the helium purging channels.

The total mass of lithium in the GCFTR blanket is 2.24×10^5 kg, and the total volume of the lithium blanket is 7.07×10^7 cm^3. Calculations used the flux distributions from a multigroup r-z model in the ORIGEN-S burnup code to calculate the producion and decay of the tritium in the blanket.

The amount of tritium that must be produced over a burn cycle for self-sufficiency is the amount required to replace the tritium burned over that burn cycle and to provide for one week of operation after restart, allowing for a 60-day decay between shutdown and restart of the next cycle. For the 600 day burn cycle, this requirement is for the production of 63.8 kg over the cycle. The calculations described above predict the production of 64.1 kg over the cycle, from which it may be concluded that the GCFTR-3 is tritium self-sufficient.

VII. COMPONENT LIFETIMES

The design lifetime of the FTWR and GCFTR neutron source is 40 years at 75% availability, or 30 EFPY. The superconducting magnets are shielded to reduce the fast neutron fluence to the superconductor and the rad dose to the insulators below their respective limits—10^{19} n/cm^2 fast neutron fluence for Nb_3Sn and 10^9 rads for organic insulators (10^{12} rads for ceramic insulators). The first-wall of the plasma chamber and the plasma-facing part of the divertor will accumulate fast neutron fluences of 7.5 and 5.8×10^{23} n/cm^2, respectively, over the 30 EFPY lifetime. The radiation damage limit of the ferritic or ODS steel first-wall structure is estimated to be 1.5-3.0×10^{23} n/cm^2, which implies that it will be necessary to replace the first- wall 2-4 times over the 30 EFPY lifetime. Erosion of the divertor by the incident plasma ion flux will necessitate several replacements over the 30 EFPY lifetime.

The achievement of the fuel cycles discussed above of course is contingent on the reactor fuel and structure radiation damage lifetimes. The FTWR fuel cycle would accumulate a fast neutron fluence of 3.4×10^{23} n/cm^2 over a 5-batch residence time, which is at the upper limit of the estimated lifetime fluence for the ferritic steel cladding and assembly structure. The fuel would then be reprocessed, reclad , recycled and placed into a new structural assembly. The similar reprocessing fuel cycle for the GCFTR would accumulate a fast neutron fluence of 3.9×10^{22} n/cm^2 over a 5.2 year residence time, while the non-reprocessing GCFTR fuel cycles of Table IX would accumulate up to 4.3×10^{23} n/cm^2. Unfortunately, there is little data for TRISO particles in fast spectra.

Component radiation damage lifetime estimates are summarized in Table IX.

TABLE IX. Component Radiation Damage Lifetimes

Component	**GCFTR-3 fast neutron fluence n/cm^2>0.1MeV)**	**LIMIT fast neutron fluence (n/cm^2>0.1MeV)**
Reactor		
Clad		
8.2 yr, 25% TRU burnup	6.9×10^{22}	1.5-3.0x10^{23} [a] ?
32.9yr, 94% TRUburnup	4.3×10^{23}	1.5-3.0x10^{23} [a] ?
TRISO Fuel particle		
8.2 yr, 25% TRU burnup	6.9×10^{22}	?
32.9yr, 94% TRUburnup	4.3×10^{23}	?
Neutron Source		
TFC Nb_3Sn 30 EFPY	1.6×10^{18}	1x10^{19} [b]
TFC insul 30EFPY	3.1×10^7 rad	10^9-10^{12} rad [b]
First-wall 30EFPY	7.5×10^{23}	1.5-3.0x10^{23} [a] ?
Divertor		Plasma erosion

[a] estimated 100-200 dpa [b] M. Sawan U. Wisc.

VIII. PASSIVE SAFETY

A thermal analysis of the core was performed under severe LOCA (loss-of-coolant) conditions--complete loss of normal core cooling. It was assumed

that the neutron source was immediately shut down and that all heat addition came from decay heat, which was calculated using the ORIGEN-S code at a burnup of 3000 days with a fuel composition of 30% U-238/ 70% TRU. The initial decay heat represented 8.8% of total thermal output, falling to 2.4% after one hour.

Once coolant is lost, the only significant process for removing heat is through thermal radiation transfer. The results of a sophisticated computation[26] of a LOCA in the annular core of a helium cooled Prismatic Fueled Reactor (PFR) were scaled according to surface area and temperature to obtain values for the amount of heat rejected from the core by radiation in the GCFTR-3. Using the mass, specific heat, and scaled rejection heat of the reactor core, the temperature change of the clad was calculated, as shown in Fig.5. The clad inner temperature peaks at 2736K well above the melting point of the clad material (1600K) and the fuel (2300K).

An accumulator system was designed to provide emergency core cooling. The accumulator design was a ring header in the shape of a torus, which was located beneath the core. Attached to the torus were 24 55m^3 standby helium tanks. The torus is connected to the reactor via four 6 inch inner diameter injection headers, each containing a flow restrictor and check valve in series, and the entire system was pressurized to 6 MPa. The effectiveness of the accumulator is shown in Fig. 5.

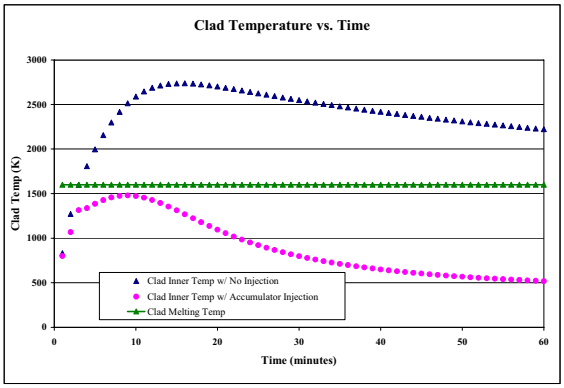

Fig. 5: Accumulator effect on cladding temperatures following LOCA

IX. TECHNICAL REQUIREMENTS FOR NEUTRON SOURCE VIS-A-VIS ELECTRIC POWER PRODUCTION

The technical requirements for a tokamak fusion neutron source that would fulfill the transmutation mission are significantly less demanding than for an economically competitive tokamak electrical power reactor and somewhat less demanding than for a DEMO, as indicated in Table X.

TABLE X: Tokamak Neutron Source, Electric Power and DEMO Requirements

Parameter	ITER	Transmutation	ElectricPower[26]	DEMO[27]
Confinement $H_{IPB98}(y,2)$	1.0	1.0-1.1	1.5-2.0	1.5-2.0
Beta β_N	1.8	2.0-2.9	> 5.0	> 4.0
Power Amplification Q_p	5-10	3-5	> 25	> 10
Bootstrap Current Fraction f_{bs}		0.2-0.5	0.9	0.7
Neutron wall load (MW/m^2)	0.5	0.5-1.8	> 4.0	> 2.0
Fusion Power (MW)	410	200-500	3000	1000
Pulse length/duty factor	modest	long/steady-state	long/steady-state	long/steady-state
Availability (%)	<10	> 50	90	50

X. CONCLUSIONS

Sub-critical operation, with a neutron source, provides nuclear reactors with additional flexibility in achieving fuel cycles that better utilize fissionable material and that reduce long-lived transuranic isotopes in the material ultimately deposited in high-level-waste repositories, thus for realizing the ultimate objective of closing the nuclear fuel cycle. A tokamak D-T fusion neutron source based on ITER physics and technology, and for which ITER operation would serve as a prototype, would meet the needs of such transmutation reactors, thus enabling fusion to contribute to solving the world's energy and environmental problems at a much earlier stage than would be possible with pure fusion electricity production.

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REFERENCES

1. "First Phase P&T Systems Study: Status and Assessment Report on Actinide and Fission Product Partitioning and Transmutation", OECD/NEA, Paris (1999).
2. "Proc. 1st-5th NEA International Exchange Meetings", OECD/NEA, Paris (1990,92,94,96,98).
3. "Nuclear Wastes--Technologies for Separations and Transmutations", National Research Council, National Academy Press, Washington (1996).
4. "A Roadmap for Developing Accelerator Transmutation of Waste (ATW) Technology", US Dept. Energy report DOE/RW-0519 (1999).
5. AFCI websites http://www.nuclear.gov/afci and http://apt.lanl.gov.
6. GEN-IV roadmap website http://gif.inel.gov/roadmap/.
7. W. M. STACEY, "Capabilities of a DT Tokamak Fusion Neutron Source for Driving a Spent Nuclear Fuel Transmutation Reactor", *Nucl. Fusion,* ***41***, 135 (2001).
8. W. M. STACEY, J. MANDREKAS, E. A. HOFFMAN, et al., "A Fusion Transmutation of Waste Reactor", *Fusion Sci. Technol.,* ***41***, 116 (2002).
9. A. N. MAUER, W. M. STACEY, J. MANDREKAS and E. A. HOFFMAN, "A Superconducting Fusion Transmutation of Waste Reactor", *Fusion Sci. Technol.,* **45**, 55 (2004).
10. J. MANDREKAS, L. A. COTTRILL, G. C. HAHN and W. M. STACEY, "An Advanced Tokamak Neutron Source for a Fusion Transmutation of Waste Reactor", Georgia Tech report GTFR-167 (2003).
11. W. M. STACEY, et al., "A Sub-Critical, Gas-Cooled Fast Transmutation Reactor (GCFTR) with a Fusion Neutron Source", *Nucl. Technol.*, **150**, 162 (2005).
12. W. M. STACEY, et al., "A Sub-Critical, He-Cooled, Fast Reactor for the Transmutation of Spent Nuclear Fuel", *Nucl. Technol.*, **156**, 99 (2006).
13. W. M. STACEY, et al., "Advances in the Sub-Critical, Gas-Cooled, Fast Transmutation Reactor Concept", *Nucl. Technol.*, (July, 2007).
14. E. A. HOFFMAN and W. M. STACEY, "Comparative Fuel Cycle Analysis of Critical and Subcritical Fast Reactor Transmutation Systems", *Nuclear Technol.*, **144**, 83 (2003).
15. E. A. HOFFMAN and W. M. STACEY, "Nuclear Design and Analysis of the Fusion Transmutation of Waste Reactor", *Fusion Sci. Technol.*, **45**, 51 (2004).
16. J. W. MADDOX and W. M. STACEY, "Fuel Cycle Anaysis of a Sub-Critical , Fast, He-Cooled Transmutation Reactor with a Fusion Neutron Source", *Nucl. Technol.*, 159, 94 (2007).
17. W.M. STACEY, J. MANDREKAS and E.A. HOFFMAN, "Sub-Critical Transmutation Reactors with Tokamak Fusion Neutron Sources", *Fusion Sci. Technol.*, **47**, 1210 (2005).
18. W. M. Stacey, "Transmutation Missions for Tokamak Fusion Neutron Sources", *Fusion Engr. Des.*, **82**, 11 (2006).
19. R. N. HILL and H. S. KHAHIL, "Physics Studies for Sodium Cooled ATW Blanket", Argonne National Lab report ANL/RAE/CP-105355 (2001).
20. F. H. SOUTHWORTH, et al., "The Next Generation Nuclear Plant (NGNP) Project", Proc. Global-3 Conf. (2003).
21. W. M. STACEY, *Fusion Plasma Physics*, Wiley-VCH, Berlin (2005), Ch. 19.
22. B. J. TOPPEL, "A User's Guide to the REBUS-3 Fuel Cycle Analysis Capability", ANL-83-2, Argonne National Laboratory (1983).
23. "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," NUREG/CR-200, Rev. & (ORNL/NUREG/CSD-2/V2/R7) Oak Ridge National Laboratory (2004).
24. W. M. STACEY, *Nuclear Reactor Physics*, Wiley-Interscience, New York (2001) Ch 2.
25. ARIES web site http://www-ferp.ucsd.edu/ARIES/.
26. W. M. STACEY "Tokamak Demonstration Reactors", *Nucl. Fusion*, **35**, 1369 (1995).