II. NEXT-STEP OPTION PHYSICS (Grant ER54350)

A. SUMMARY

For more than a decade we have been involved in physics and design analysis of possible next-step tokamak options, including first ITER, later FIRE and most recently a tokamak neutron source for a near-term transmutation reactor for burning the transuranics in spent nuclear fuel. We have also recently supported the National Transport Code Coordination activity under this grant. In recent years, much of the effort has been devoted to defining the physics and performance characteristics required of a tokamak fusion neutron source that could drive a sub-critical reactor for the transmutation of the transuranics in spent nuclear fuel. This document provides a final report for the activity in each of these areas for the last grant period.

B. SUB-CRITICAL TRANSMUTATION REACTORS WITH A TOKAMAK FUSION NEUTRON SOURCE

1. Background

At the present rate of nuclear power production in the USA the accumulation of spent nuclear fuel (SNF) discharged from the reactors in the "once-through" fuel cycle will require the opening of a new high-level waste (HLW) repository on the Yucca Mountain scale about every 30 years. The repository requirements can be greatly (an order of magnitude or more) reduced (and the nuclear fuel utilization can be increased) if the plutonium and higher transuranics in the SNF is recycled and used as fissionable fuel in other reactors designed for that purpose, since the decay heat of these transuranics is the principal constraint on the volume of spent fuel that can be stored in a repository. Studies over the past decade or so[1-3] confirmed the technological feasibility of spent fuel transmutation and have identified the potential advantage of sub-critical reactors driven by neutron sources for this purpose. Accelerator-spallation neutron sources have been extensively studied for this application, but relatively little effort has been devoted to investigating the application of fusion neutron sources.

2. Previous Work on Transmutation Reactor Fusion Neutron Sources under Grant 54350.

For the past several years we have investigated the required characteristics of a tokamak D-T neutron source to drive a sub-critical reactor for the purpose of the transmutation of spent nuclear fuel, vis-à-vis the existing tokamak plasma physics and fusion technology database, with the objective of developing a tokamak neutron source that will be prototyped by ITER. The investigation has included the conceptual design of fast-spectrum, sub-critical transmutation reactors that are compatible with the tokamak neutron source geometry and that are based on the nuclear, fuel, materials and separation

technologies that are being investigated in the DoE Nuclear Energy Programs; i.e the Generation-IV and Advanced Fuel Cycle Initiatives and the Advanced Nuclear Fuel Development Program. In order to insure a close integration of our work on the definition of the tokamak neutron source requirements with this nuclear technology being developed being in the Nuclear Energy Program, we have 1) interacted with other Georgia Tech faculty who are working on advanced fuels and actinide separations systems; 2) interacted with scientists at the Argonne and Oak Ridge National Laboratories and at the Idaho National Engineering Laboratory who are working on the GEN-IV, AFCI and Advanced Fuels programs; and 3) involved these people in a series of student-faculty conceptual design projects of transmutation reactors to insure the compatibility of the tokamak neutron source concepts being developed under this grant with nuclear technology being developed under the GEN-IV, AFCI and Advanced Fuels Programs. This work is documented below and summarized in the attached paper.

Documentation of work on transmutation tokamak neutron sources under Grant 54350

a. W. M. Stacey, "Capabilities of a D-T Tokamak Fusion Neutron Source for Driving a Spent Nuclear Fuel Transmutation Reactor", Nucl. Fusion, 41, 135 (2001).
b. W. M. Stacey, J. Mandrekas, E. A. Hoffman, et al., "A Fusion Transmutation of Waste Reactor", Fusion Sci. Technol., 41, 116 (2002).
c. W. M. Stacey, J. Mandrekas, E. A. Hoffman, et al., "A Fusion Transmutation of Waste Reactor" *Fus. Eng. & Des.*, 63-64, 81 (2002).
d. E. A. Hoffman and W. M. Stacey, "Nuclear and Fuel Cycle Analysis for a Fusion Transmutation of Waste Reactor", *Fus. Eng. & Des.*, 63-64, 87 (2002).
e. A. N. Mauer, W. M. Stacey, J. Mandrekas and E. A. Hoffman, "A Superconducting Tokamak Fusion Transmutation of Waste Reactor", Fusion Sci. Technol., 45, 55 (2004).
f. E. A. Hoffman and W. M. Stacey, "Nuclear Design and Analysis of the Fusion Transmutation of Waste Reactor", Fusion Sci. Technol., 45, 51 (2004).
g. J. Mandrekas, L. A. Cottrill, G. C. Hahn and W. M. Stacey, "FTWR-AT: An Advanced Tokamak Neutron Source for a Fusion Transmutation of Waste Reactor", Georgia Tech report GTFR-167 (2003).
h. E. A. Hoffman and W. M. Stacey, "Comparative Fuel Cycle Analysis of Critical and Sub-critical Fast Reactor Transmutation Systems", Nucl. Technol., 144, 83 (2003).
i. W. M. Stacey, "Transmutation Missions for Fusion Neutron Sources", Fusion Engr. Des., to be published (2004).
j. W. M. Stacey, V. L. Beavers, W. A. Casino, et al., "A Sub-Critical , Gas-Cooled Fast Transmutation Reactor (GCFTR) with a Fusion Neutron Source", Nucl. Technol., to be published (2004).
k. W. M. Stacey, J. Mandrekas and E. A. Hoffman, "Sub-critical Transmutation Reactors with Tokamak Fusion Neutron Sources", Fusion Sci. Technol., to be published (2005).

Attachment 1

SUB-CRITICAL TRANSMUTATION REACTORS WITH TOKAMAK FUSION NEUTRON SOURCES

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ABSTRACT

The principal results of a series of design scoping studies of sub-critical fast transmutation reactors (based on the nuclear and processing technology being developed in the USDoE Generation IV, Advanced Fuel Cycle and Next Generation Nuclear Plant programs) coupled with a tokamak fusion neutron source (based on the ITER design basis physics and technology) are presented.

I. INTRODUCTION

For many years there has been a substantial R&D activity devoted to closing the nuclear fuel cycle. During the 1990s this activity emphasized the technical evaluation of reducing the requirements for long-term geological high-level waste repositories (HLWRs) for the storage of spent nuclear fuel (SNF) by transmutation (fission) of the plutonium and higher transuranics in the spent fuel discharged from fission power reactors[1-8]. Recycling of this SNF in thermal spectrum fission power reactors, the most obvious option, was found to not significantly reduce the HLWR requirements[1,2], because the destruction of transuranics (by neutron fission) would be offset by the production of more transuranics by neutron capture transmutation of the isotope ^{238}U that constitutes about 95% of (slightly enriched) thermal reactor fuel. Repeated recycling of the SNF in special purpose fast spectrum reactors was found to be more effective, but with the net destruction rate of transuranics still limited by the requirement for the presence of ^{238}U to provide a negative reactivity coefficient for safety and by a safety-related limit on the transuranics loading. There is a potential to relax these two safety-related limits if the reactor is operated sub-critical, with a neutron source making up the neutron deficit to sustain the neutron chain reaction. A general consensus emerged from these studies that significantly higher transuranics net destruction rates could be achieved in sub-critical reactors[1,2].

The accelerator community was quick to recognize the opportunity to use a D+ accelerator with a spallation target as a neutron source for a sub-critical transmutation reactor. Almost all of the studies in the 1990s of sub-critical transmutation reactors were based on an accelerator-spallation neutron source[1-8]. In the USA, these studies and the supporting R&D development were organized by DoE under the Accelerator Transmutation of Waste (AWTR) Program[6], which has now evolved into the Advanced Fuel Cycle Initiative[9] (AFCI).

The USDoE Generation IV (GEN-IV) nuclear reactor development activity[10] envisions that the pacing item for the development of a transmutation reactor--the development of the spent fuel processing technology--should be sufficiently advanced by about 2020 that the detailed design of a critical fast transmutation reactor and the associated processing facility could be started, which would enable the entire system to be brought online in about 2030. The roadmap[6] for developing sub-critical transmutation reactors driven by accelerator-spallation neutron sources also envisions such a reactor coming online in about 2030.

A sub-critical transmutation reactor (using the same nuclear and separations technology) driven by a tokamak fusion neutron source could be brought online somewhat later. The pacing items in bringing online a tokamak neutron source to drive a sub-critical transmutation reactor would be the operation of ITER (or a similar facility) as a prototype and the operation of a set of fusion technology test facilities needed to develop component reliability. ITER is scheduled to operate from 2015 to 2035. Component test facilities could be upgraded or constructed to operate before and in parallel with ITER, so it would be plausible to begin detailed design of a tokamak neutron source in about 2025. Construction of a sub-critical reactor using the same fast reactor technology developed for critical reactors and a tokamak fusion neutron source could then begin as early as about 2030, leading to initial operation in about 2040.

The fusion community has been rather slower in examining the opportunity of using a fusion neutron source for a sub-critical transmutation reactor, with only a few studies[11-16] through the end of the 1990s. Since that time we have undertaken at Georgia Tech a series of studies[17-24] of coupling a tokamak fusion neutron source based largely on ITER design basis physics and technology[25] with a

sub-critical transmutation reactor based on the nuclear and processing technology being developed in the USDoE GEN-IV, AFCI and NGNP programs[9,10,26].

II. THE FTWR AND GCFTR STUDIES

We have examined sub-critical transmutation reactors based on two of the nuclear technologies being developed in the GEN-IV studies. The Fusion Transmutation of Waste Reactor (FTWR) series of studies was based on a fast-spectrum reactor using a metal fuel consisting of TRU (transuranics) alloyed with zirconium in a zirconium matrix and cooled by a liquid metal (Li17Pb83 eutectic), which also served as the tritium breeder. The ongoing Gas Cooled Fast Transmutation Reactor (GCFTR) series of studies is based on 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 and then by a shield to protect the magnets from radiation damage and heating, as indicated in Fig. 1 for the initial FTWR design.

A design objective was to use near-term nuclear technology being developed in the DoE Nuclear Program (GEN-IV, AFCI, NGNP) and near-term fusion technology being developed in the ITER Project. The ANL metal fuel, liquid metal cooled reactor designs[8] were adapted to accommodate a different coolant and TRU-Zr fuel for the FTWR designs. The fast, gas-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[26] was adapted to accommodate TRU-oxide fuel for the GCFTR.

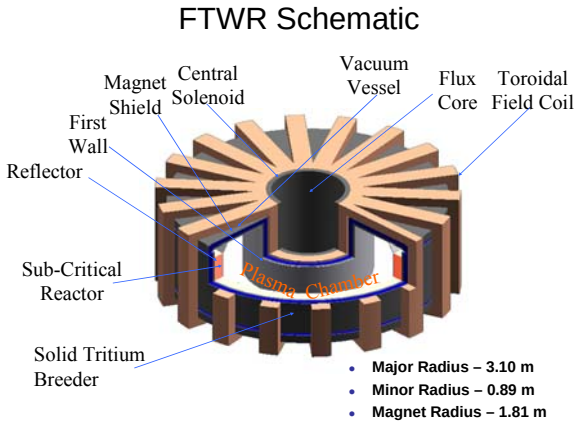

Fig. 1 Tokamak Fusion Transmutation of Waste Reactor

The fusion technology was based on the ITER design[25]. The superconducting magnet design was based directly on the ITER superconducting magnet system. The first-wall and divertor designs were based on adaptations of the ITER designs to accommodate different coolants. The reference materials compositions for the FTWR and GCFTR designs are given in Table I.

TABLE I Reference Materials Composition of FTWR and GCFTR

Component	FTWR	GCFTR
Reactor		
Fuel	TRU-Zr metal in Zr matrix	TRU TRISO/SiC matrix (option BISO/Zirc-4 matrix)
Clad/structure	FeS/FeS	Zirc-4/FeS
Coolant	LiPb	He
Trit. Breeder	LiPb	LiO_2
Reflector	FeS, LiPb	FeS, He
Shield	FeS, LiPb, B_4C, ZrD_2, W	W, B_4C, He
Magnets	NbSn,NbTi/He (OFHC/LN_2)	NbSn, NbTi/He
First-Wall	Be-coated FeS, LiPb	Be-coated FeS, He
Divertor	W-tiles on Cu-CuCrZr, LiPb	W-tiles on Cu – CuCrZr, He

A series of design studies was performed for the FTWR. The objectives of the original FTWR study[18] were to achieve minimum size by using liquid nitrogen cooled Cu magnets, to achieve electrical power breakeven (Q_e = 1), and to achieve an adequate transmutation rate to dispose of the spent nuclear fuel being generated by three 1000 MW_e LWRs. The second FTWR-SC study[19] was a modification of the FTWR design to replace the Cu magnets with superconducting magnets and to

provide enough shielding to make them lifetime components. The core radius became larger as a result, and the power density was held constant so that the FTWR nuclear and thermal core design[18,24] and fuel cycle analysis[18,23] could be simply scaled up by volume. The third FTWR-AT study[20] investigated the reduction in size that could be achieved in a superconducting design by using advanced tokamak physics; again the same core power density was used.

The GCFTR series of studies is now in progress. The objectives of the first GCFTR study[22] were to achieve > 90% burnup of transuranics in the coated fuel particles without reprocessing the coated TRU pellets, achieve an adequate transmutation rate to dispose of the spent nuclear fuel being generated by three 1000 MW_e LWRs, and to achieve net electric power while avoiding the very high temperatures (and associated materials requirements) characteristic of other gas-cooled reactor designs. During the later stages of the GCFTR study it became apparent that the superconducting magnet thicknesses could be reduced, and the preliminary GCFTR-2 study was performed to assess the effect on the design.

The major dimensions of the various design concepts are given in Table II. The plasma-related parameters for the FTWR and GCFTR designs are given in Table III.

TABLE II Dimensions (m) of FTWR and GCFTR Designs

Parameter	FTWR[a]	FTWR-SC[b]	FTWR-AT[c]	GCFTR[d]	GCFTR-2[d]
Major Radius[e], R_0	3.10	4.50	3.86	4.15	3.70
Fluxcore, R_{fc}	1.24	1.10	0.65	0.66	0.66
CS+TF, Δ_{mag}	0.57	1.68	1.20	1.50	1.13
Refl+Shld, Δ_{rs}	0.40	0.65	0.90	0.86	0.82
Plasma, a_{plasma}	0.89	0.90	1.10	1.04	1.08
Core					
Inner Radius, R_{in}	4.00	5.40	5.00	5.25	4.84
Width, W	0.40	0.40	0.40	1.12	1.12
Height, H	2.28	2.28	2.28	3.00	3.00

[a]ITER physics, LN_2 Cu magnets, PbLi/TRU-metal reactor[18]; [b] ITER physics, ITER SC magnets, PbLi/TRU-metal reactor[19]; [c] AT physics, SC magnets, PbLi/TRU-metal reactor[20]; [d] ITER physics, ITER SC magnets, He/TRU-TRISO reactor[22]; [e] Includes gap, first-wall, scrape-off layer and items below.

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. 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.

III. TRANSMUTATION REACTOR CORES

III.A. FTWR

The fuel is a transuranic zirconium alloy (TRU-10Zr) dispersed in a zirconium matrix 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 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 blanket design studies[8]. The same pin and assembly geometry was used, with the exception that the length of the assembly was increased to 228 cm. Table IV 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, as shown in Fig. 2.

The total coolant mass flow rate required to maintain T_{in} = 548 K and T_{out} = 848 K is 51630 kg/s. The required pumping power is 130 MW, the majority of which is needed to overcome MHD losses.

TABLE III Tokamak Neutron Source Parameters for Transmutation Reactors

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

[a–d] same as Table II; [e] ITER design parameters. (Ref. 25); [f] bootstrap current calc. using ITER scaling, then required CD effic. calculated.

TABLE IV FTWR Fuel Assembly Design

Pin Diameter (cm)		0.635
Clad thickness (cm)		0.05588
Pitch		Triangular
Pitch to Diameter		1.727
Pins per assembly		217
Structure Pins		7
Fuel Smear density		85%
Hexagonal Assembly Pitch		16.1
Assembly Length (cm)		228
Assemblies		470
Pumping Power (MW)		130
Volume %	Fuel	17.01
	Structure	10.44
	Coolant	69.55
Materials	Fuel	TRU-10Zr/Zr
	Structure	FeS
	Coolant	Li17Pb83

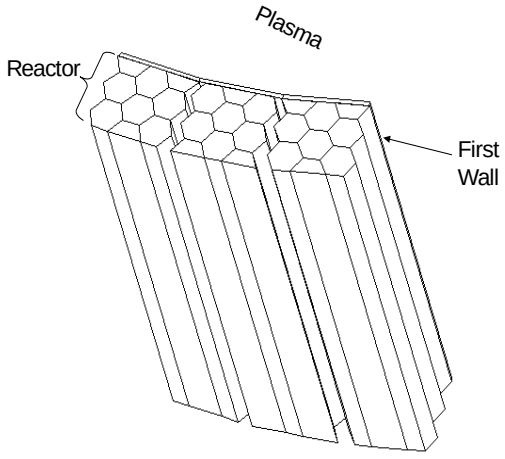

Fig. 2 FTWR Transmutation Reactor Core Configuration Outboard of Plasma Chamber

III.B. GCFTR

Design concepts were developed for a TRISO (tri-material isotropic) particle and for a BISO (bi-material isotropic) particle, as shown in Fig. 3.

The TRISO particle has a TRU kernel (300 μm diameter) surrounded by a 50% porous buffer layer (100 μm) of ZrC to allow for fission product recoil and to accommodate fission product gas buildup, followed by a structural layer (20 μm) of pyrolytic carbon which prevents chlorine attack of the kernel during the coating process and contains the fission products, followed by a structural layer (25 μm) of SiC which shrinks under irradiation to provide an inward pressure to counteract the fission product gas pressure buildup, followed by an outer pyrolytic carbon layer (35 μm) to prevent interaction of the SiC

with any metallic cladding material. The BISO particle has a similar kernel and buffer layer followed by a (25 μm) pyrolytic carbon structural layer and then by a (35 μm) ZrC structural outer layer.

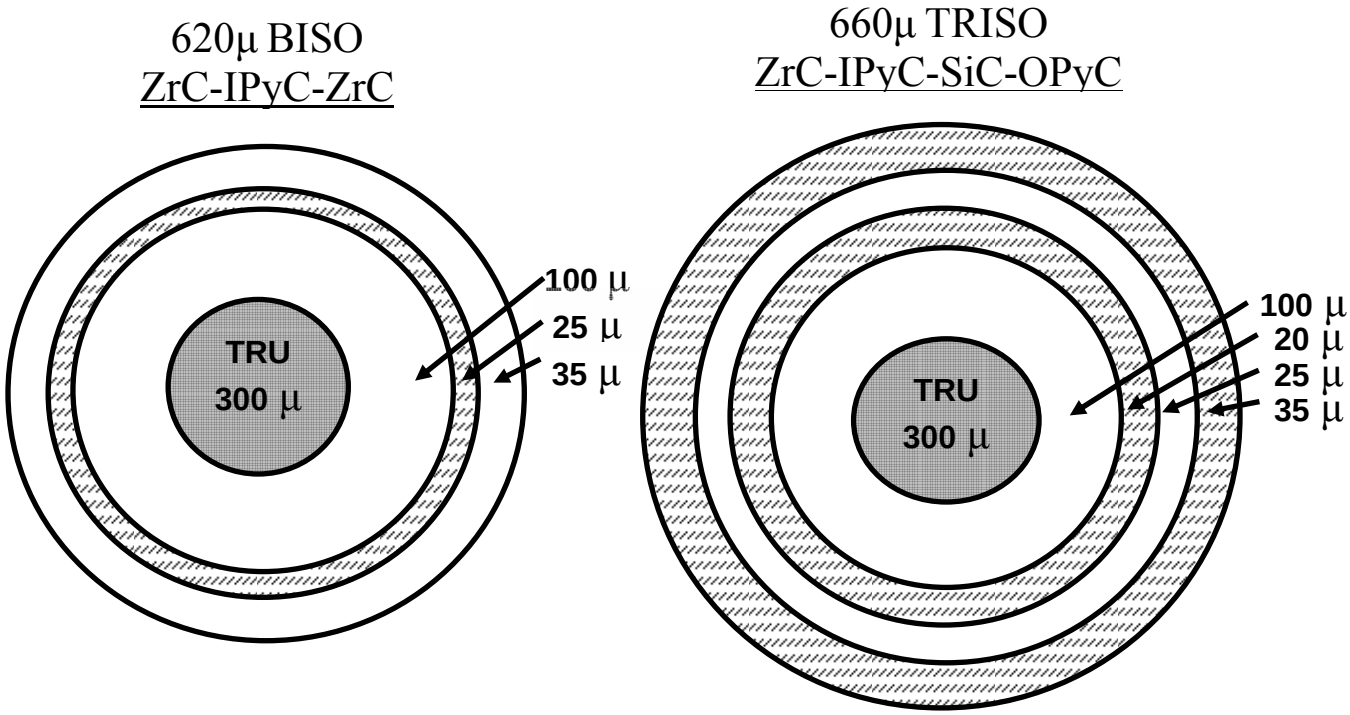

Figure 3: BISO and TRISO coated fuel particles

It is an objective to achieve very high burnup before loss of integrity of the coated fuel particle or degradation of fission product gas containment becomes unacceptable. The TRISO and BISO particles are predicted to reach 155 MPa at 90% FIMA and 180MPa at 99% FIMA for the maximum predicted fuel centerline temperature of 560 ^{o}C. The operational pressure limit due to the compressive yield strength of SiC for the TRISO particle is 345 MPa, and the similar limit for the BISO particle is 352 MPa. These limits correspond to fuel centerline temperature limits of 1700 and 1520 ^{o}C at 90% FIMA and 99% FIMA, respectively, for the BISO particle; and to fuel centerline temperature limits of 1690 and 1510 ^{o}C at 90% FIMA and 99% FIMA, respectively, for the TRISO particle.

A thermal analysis was performed for Zirc-4 clad pins in which the BISO fuel particles were uniformly homogenized in the Zirc-4 matrix material. A configuration with 207200 fuel pins 0.60 cm in radius with a gap of 0.005 cm and a 0.057 cm thick cladding was chosen for the analysis. For 3000 MW_{th} total reactor power uniformly distributed in the fuel pins, the volumetric heat source is q''' = 42.2 MW/m^3. With a He mass flow rate of 2870 kg/s, the He coolant entered at 280 C and exited at 481 C, the maximum clad temperature was 513 C (well below the 1845 ^{o}C m. p. for Zircaloy), the maximum homogenized fuel centerline temperature was 560 C, well below the 2000+ ^{o}C melting point for TRU-oxides, and the He pumping power was 0.15 MW. A He coolant v/o ≥ 25% would be adequate for heat removal under normal operating conditions.

IV. FUEL SEPARATION AND FABRICATION SYSTEMS

IV.A. FTWR

The processing system for the FTWR will be identical to the waste processing system being developed for the ATW system[27]. The waste processing system consists of three basic components. The first is a uranium extraction system (UREX) that will separate the bulk uranium and fission products in the SNF from the transuranic elements. The transuranic elements and the rare earth fission products will then be transferred to a pyro-metallurgical system (Pyro-A) that will separate the rare earths from the transuranic elements and convert the latter to a metallic form for fuel manufacturing. The discharged FTWR fuel will be sent to a separate

pyro-metallurgical system (Pyro-B) where the residual actinides will be recovered. The recovered materials from Pyro-A and Pyro-B will be blended together and manufactured into new fuel elements for the FTWR.

The UREX system is assumed to remove 99.995% of the uranium and all of the fission products that are not rare earth elements. The Pyro A system is assumed to remove 95% of the rare earth fission products and recover 99.9% of the transuranic elements. The Pyro B system is assumed to remove 95% of the rare earth fission products, remove 100% of all other fission products, and recover 99.9% of the transuranic elements. In addition to the recovery fractions, the total fraction of transuranics that end up in the waste stream is a strong function of fractional burnup achieved during each residence in the FTWR. For the FTWR, each MTU of SNF will result in 70 g of transuranics in the waste stream.

IV.B. GCFTR

Aqueous systems for separating the TRU in LWR spent nuclear fuel (SNF) and systems for fabricating it into coated particle fuel have been identified for the GCFTR. The uranium (99.995%) is first removed from the SNF using a UREX process. The remaining 0.005% of the uranium, the TRU and the fission products are then treated with a TRUEX process and a TRU/lanthanide separation step to remove virtually all of the fission products, which are sent to a high-level waste repository. The TRU emerging from the TRUEX process (including 0.005% of the uranium and virtually all of the transuranics) is then fabricated into coated TRU fuel particles. The heavy metal composition of the 'TRU' emerging from this process is (U—0.43%, Np—4.32%, Pu—84.91%, Am—10.21%, Cm—0.13%).

The fabrication process starts with evaporation of the TRU stream, which is then passed through a calciner to form a mixture of transuranic oxides. Finally, a ZrC buffer layer and the pyrolytic carbon and ZrC (BISO) or pyrolytic carbon and SiC (TRISO) layers are coated onto the particles. Less than 0.1% TRU loss is assumed during the fabrication process.

V. FUEL CYCLE ANALYSIS

The composition changes in the fuel cycle were calculated with the REBUS fuel cycle code[28]

V.A. FTWR

The reference fuel cycle assumes that the FTWR fuel will remain in the reactor for 4 cycles of 623 days each and then be reprocessed, blended with 'fresh' SNF and fabricated into new fuel elements for re-insertion into a FTWR. A beginning-of-cycle (BOC) transuranic loading of 27 MTU will produce k_{eff} =0.95, the largest value during the cycle. Over the 40 FPY plant life of the first generation of FTWRs, the original charge of LWR feed will be reprocessed 5 times.

The initial charge of the reactor and the first reload batch will require approximately 3500 MTU of LWR SNF to manufacture these fuel elements. Following this, approximately 190 MTU of LWR SNF will be processed in each subsequent 623-day cycle. A first generation FTWR 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 used 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 over their entire life. Repeated recycling of the discharged transuranics from FTWRs in successive generations of FTWRs will ultimately result in the destruction of 99.4% of the transuranics discharged from LWRs.

V.B. GCFTR

An emphasis in the GCFTR investigation was achieving sufficiently high (>90% FIMA) TRU burnup that the coated fuel particles can be burned and then removed from the reactor and directly deposited in a waste repository without the necessity of reprocessing. To this end, we again examined a multi-batch fuel cycle in which the reactivity decrease (from k = 0.95 at BOC) associated with fuel burnup was partially offset by an increase in neutron source strength over the burn cycle.

For the reference 5-batch, 600 day burn cycle, 8.2 year fuel cycle, the BOC TRU loading was 36 MT for the TRISO fuel and 47 MT for the BISO fuel. For both fuels, the BOC k_{eff} = 0.95 and neutron source $P_{fusion} \approx 40$ MW, and the end of cycle $k_{eff} \approx 0.81$ (0.87 for BISO) and neutron source $P_{fusion} \approx 170$ MW (107 MW for BISO). About 23% of the BOC TRU loading is fissioned in an 8.2 year fuel cycle. The fuel would have to be resident in the core for about 10 such fuel cycles to achieve 90% TRU burnup.

V.C. Performance

The FTWR and GCFTR cores are designed to operate at a nominal fission power level of 3000 MW_{th}, which corresponds to the fission of 1.1 metric

tons of TRU per EFPH. A typical 1000 MW_e LWR produces 0.36 metric tons of TRU per EFPY. Hence, one FTWR or GCFTR would be able to 'support' (burn the TRU discharged from) three 1000 MW_e LWRs.

The FWTR and GCFTR also produce electrical power. The original FTWR with Cu magnets has a large ohmic heat removal power requirement and was designed for electrical breakeven, but superconducting FTWRs would produce net electrical power, as the GCFTR does. Using a Brayton cycle with 32% thermal-to-electrical energy conversion efficiency to convert the 3000 MW thermal power, the gross electric power production of a GCFTR would be 1024 MW_e. The electrical power requirements for the operation of the GCFTR are 305 MW_e, leading to an electric power amplification factor of Q_e = 3.4 and a net electric power production of 719 MW_e.

V.D. Deployment

Availability of the transmutation reactor will determine the annual transmutation rate, hence the number of transmutation reactors needed to service the USA LWR fleet. The projected SNF transmutation rate is 100A MTU per year for both the FTWR and the GCFTR, where A is the availability. (The other design variants with somewhat higher power would have somewhat higher transmutation rates.) At the present level of nuclear power production in the US, about 100 LWRs produce about 2000 MTU of SNF per year. Thus, 20/A transmutation reactors would be needed to handle the annual SNF production, assuming the present level of nuclear power continues indefinitely. Operating at 50% availability, 40 sub-critical reactors would accomplish this transmutation mission. At 75% availability, only 25 would be needed.

VI. COMPONENT LIFETIMES

The design lifetime of the GCFTR is 40 years at 75% availability, or 30 EFPY. The magnet systems, shields, reflectors, etc. are designed as lifetime components. However, the reactor fuel and structure, the first-wall of the plasma chamber and the divertor will have to be replaced one or more times over the 30 EFPY because of radiation damage.

It is envisioned that the coated fuel pellets will be imbedded in a matrix material and clad in Zircalloy-4 fuel elements and arranged in fuel assemblies constructed of ferritic steel. The fuel elements will be left in the reactor for five consecutive 600 EFPD cycles, which requires that the clad not fail in this "residence" time, during which it will accumulate a fast (E > 0.1 MeV) neutron fluence of 4.2×10^{22} n/cm^2. We have not been able to determine the radiation damage lifetime of Zircalloy-4, but it is widely used as cladding in nuclear reactors.

The structural material of the fuel assembly will accumulate a fast neutron fluence of 1.9×10^{23} n/cm^2 over the 30 EFPY design lifetime. The estimated[29] radiation damage lifetime of ferritic steel is 80-150 dpa, or $1.5\text{-}3.0 \times 10^{23}$ n/cm^2, implying that the core fuel assembly structure may need to be replaced once over the 30 EFPY lifetime of the GCFTR.

When the fuel is removed from the reactor after its residence time, the cladding will be replaced, and the matrix material (SiC or Zircalloy-4) will be replaced if necessary, but the coated fuel pellets will be blended with "fresh" fuel pellets and re-fabricated into fuel elements to be re-inserted into another GCFTR. The objective is to repeatedly recycle the fuel pellets until they reach > 90% FIMA, without reprocessing. The fast neutron fluence will be 4.1×10^{23} and 8.2×10^{23} n/cm^2 at 90% and 99% FIMA, respectively. A fluence lifetime in this range is then a requirement of the coated particle fuel development program.

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 of the GCFTR. The radiation damage limit of the ferritic steel first-wall structure is $1.5\text{-}3.0 \times 10^{23}$ n/cm^2 , which implies that it will be necessary to replace the first-wall 2-4 times over the 30 EFPY lifetime of the GCFTR. Erosion of the divertor by the incident plasma ion flux will necessitate several replacements over the 30 EFPY lifetime of the GCFTR.

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^{10} rads for ceramic insulators).

VII. TRANSMUTATION MISSION IN THE FUSION PROGRAM

A transmutation reactor can be driven by a tokamak fusion neutron source based on physics (H, β_N, Q_p, etc.) similar to or less demanding than that used for the ITER design, except for the need to achieve a higher bootstrap current fraction and/or higher current drive efficiency. This tokamak neutron source can be constructed with the fusion technology being developed for ITER, but will need

to achieve greater availability, hence have greater component reliability, than ITER. Achieving higher availability, which will require various component test facilities, must be addressed in the fusion development program, but would have a higher near-term priority if the transmutation mission were undertaken.

The reactor technology for the sub-critical reactor driven by the fusion neutron source would be adapted from the reactor (nuclear, fuel, cooling, separations, materials) technologies being developed in the nuclear program (e.g. GEN–IV, AFCI, NGNP), but these technologies must be modified to provide for the tritium breeding requirement. A fusion nuclear technology program would have to be revived with this goal. There is a need to develop a long-lived structural material, primarily for the fuel assemblies of the sub-critical reactor, but also for the first wall of the fusion neutron source.

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 V.

Table V Requirements for a Tokamak Neutron Source, Electric Power Reactor and DEMO

Parameter	Transmutation	Electric Power[a]	DEMO[b]
Confinement $H_{IPB98}(y,2)$	1.0	1.5-2.0	1.5-2.0
Beta β_N	< 2.5	> 5.0	> 4.0
Power Amplification Q_p	< 3	> 25	> 10
Bootstrap Current Fraction f_{bs}	0.2-0.5	0.9	0.7
Neutron wall load (MW/m^2)	≤ 1.0	> 4.0	> 2.0
Fusion Power (MW)	≤ 200	3000	1000
Pulse length/duty factor	long/steady-state	long/steady-state	long/steady-state
Availability (%)	> 50	90	< 50

[a] ARIES studies (Ref. 30); [b] DEMO studies (Ref. 31)

VIII. CONCLUSIONS

A sub-critical transmutation reactor, based on adaptation of nuclear and separations technology presently being developed in the DoE Nuclear Energy Program to accomodate tritium breeding, and driven by a tokamak D-T fusion neutron source, based on the physics and technology presently being developed in the DoE Fusion Energy Sciences Program, could be online in 2040. The tokamak neutron source, which would be about R = 4 m in major radius and produce < 200 MW of D-T fusion power, could be designed on the basis of the existing plasma physics and fusion technology databases, with only a few modest extensions. The pacing items for the neutron source would be operation of a prototype plasma (e.g. ITER) experiment and component test facilities to gain the experience necessary to achieve > 50% availability in operation of the fusion neutron source.

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C. WORK IN SUPPORT OF FIRE

1. Background

Over the duration of the Fusion Ignition Research Experiment (FIRE) study, the Georgia Tech Fusion Research Center performed physics simulations utilizing unique computational tools and analysis capabilities that have been built up over more than a decade of such work. These capabilities include: 1) a 1½-D plasma transport code with multiple impurity species and charge states, that has been used to develop the impurity-seeded radiative mantle power exhaust concept for ITER, in supporting analyses for the 2002 SNOWMASS workshop, and for FIRE; and 2) a global profile-averaged core power balance code that has been widely used for POPCON analysis of FIRE and in supporting analyses of other proposed Burning Plasma experiments for the SNOWMASS workshop.

2. FIRE Performance Evaluation Studies

We performed 0-D simulations using our profile-averaged tokamak power balance code to evaluate the performance of FIRE under various operating scenarios and to compare FIRE against other candidate burning physics experiments (BPX).

In Fig. 10, a POPCON plot for the reference FIRE design parameters is shown. The thick black line represents the boundary of the operating space which is defined by $Q > 5$, $P/P_{LH} > 1$, $P_{aux} < P_{aux}$ (max), $\beta_N < 2.0$ and $n/n_{Gr} < 1$, where P/P_{LH} is the ratio of the output power to the L-H threshold power, P_{aux}(max) is the maximum available auxiliary power for the device, β_N is the normalized beta and n/n_{Gr} is the Greenwald density limit fraction.

To explore the sensitivity of the operating space to the confinement scaling itself, POPCONs were constructed using various extensions of the IPB98(y,2) scaling. In Fig. 11, a POPCON for the FIRE reference assumptions is shown, using Cordey's extension of the IPB98(y,2) scaling (*J.G. Cordey, et al., 28th EPS Conf. on Contr. Fusion and Plasma Phys., 2001*) which attempts to account for the effects of triangularity, proximity to the Greenwald density limit and density peaking.

The results of our FIRE performance evaluation studies have been published in various project reports and presented at international meetings[1-3].

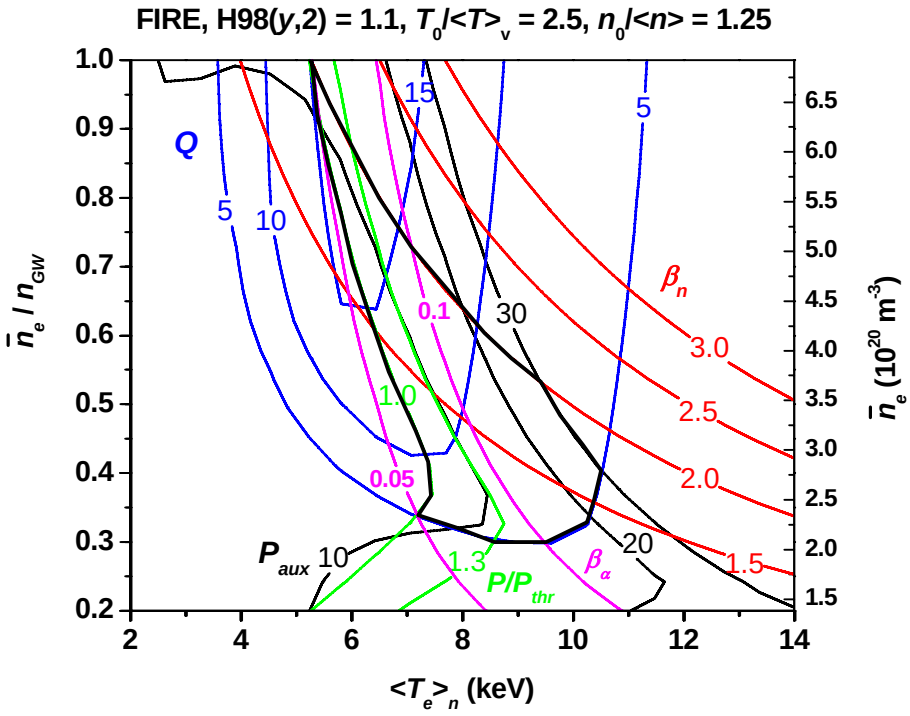

Figure 10: FIRE POPCON for the reference design assumptions.

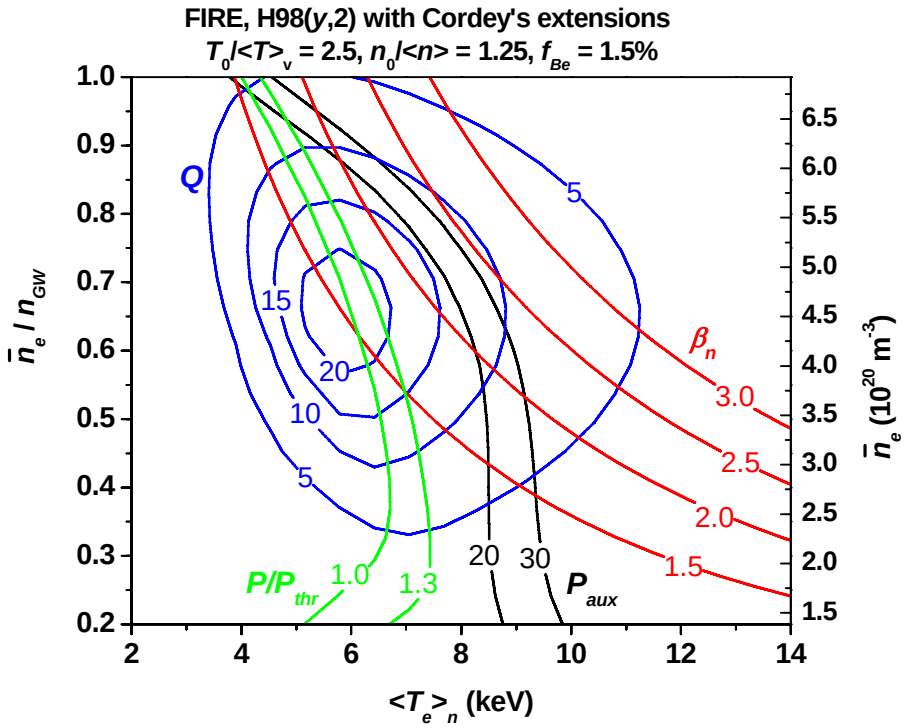

Figure 11: FIRE POPCON using the IPB98(y,2) scaling with Cordey's extensions.

3. 1½-D Transport Simulations

While 0-D simulations provide us with a valuable insight into the operating space of each reactor design and its sensitivity to several uncertain parameters, time-dependent simulations of the reference operating scenario with 1½-D transport codes employing theory-based transport models are still necessary to assess the performance projections of FIRE and other BPXs. We performed such simulations for the FIRE reference operating scenario with our GTWHIST 1½-D transport code[4] and using the latest version of the GLF23 transport model. In Fig. 12, the time histories of various power balance quantities are shown. We found that the performance of the reactor was very sensitive to the pedestal temperature. A pedestal temperature of about 4 keV was required to achieve $Q = 10$ at a Greenwald density limit fraction of 0.75.

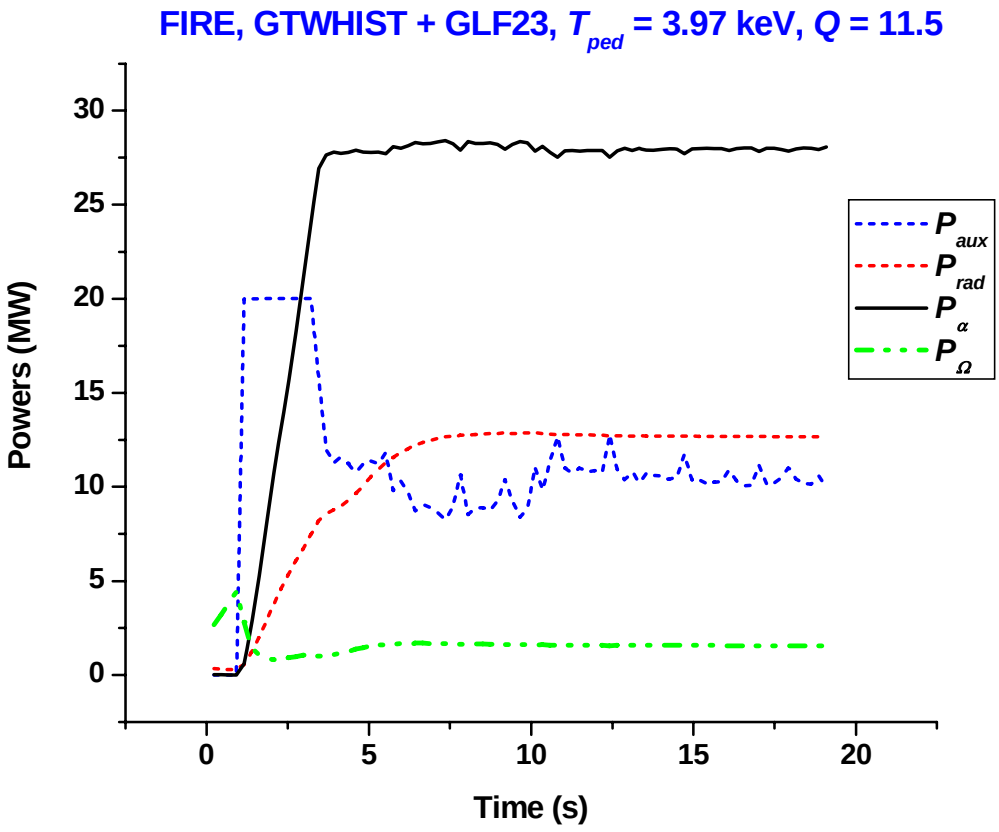

Figure 12: Time history of various power balance parameters of the FIRE reference scenario using the GLF23 transport model.

4. Impurity Transport Simulations for FIRE

We performed transport simulations with our 1½-D main plasma – multi charge state impurity transport code GTWHIST, in order to evaluate the impact of impurity seeded operation on the performance of FIRE.

While the new FIRE divertor design[5] can withstand the anticipated heat loads from the plasma core during the standard ELMy H-mode operation of the device, enhanced radiation from seeded impurities from the plasma mantle and the divertor is expected to be necessary during the higher power Advanced Tokamak (AT) operating mode in order to maintain a flexible operating space.

As a first step, the entire* FIRE reference operating scenario was modeled with GTWHIST and compared to the reference TSC simulation[6]. The results of this benchmarking simulation are shown in Fig. 13, where time histories of various global power quantities are plotted. A fixed-shape transport model normalized to yield an H-factor of about 1 relative to the ITER IPB(y,2) global confinement scaling was adopted for these simulations.

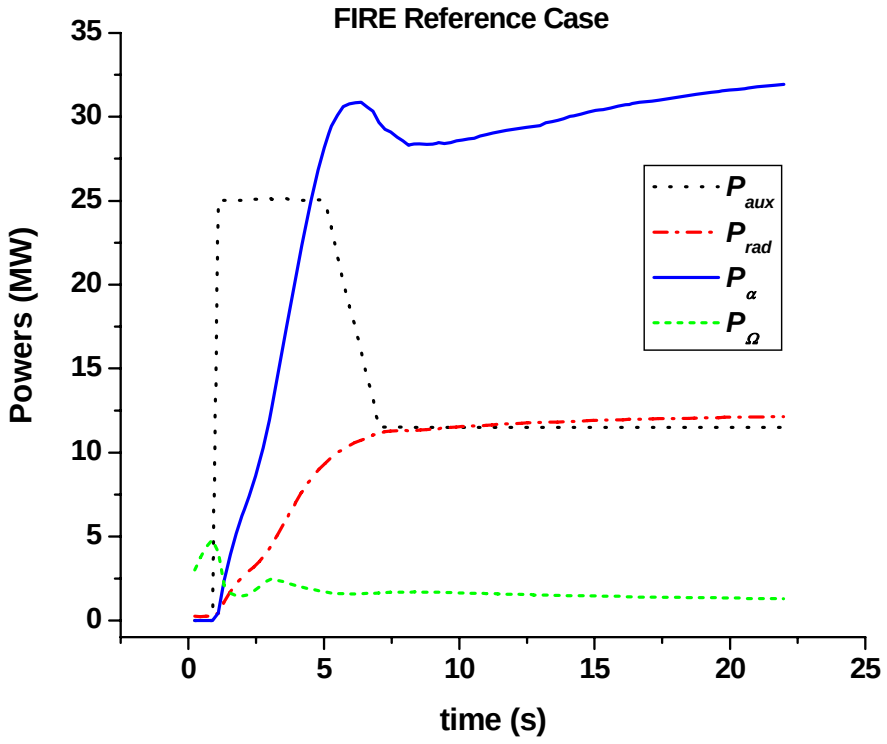

Figure 13: GTWHIST evaluation of the time history of various global power balance parameters for the FIRE reference case.

Following the establishment of the reference discharge, Argon impurities were injected at the edge of the device and their evolution and contribution to the power balance were followed using the multi-charge state impurity transport capabilities of the GTWHIST code. A fixed diffusion coefficient of 0.5 m^2/s for all impurity charge states and no inward pinch have been assumed in these simulations. The profiles of the various Ar charge states are shown in Fig. 14, for a 0.3% global Ar concentration.

* Since the MHD part of the GTWHIST code supports fixed-boundary configurations only, our simulation starts when the plasma geometry and fields (major and minor radii and toroidal magnetic field) are at their reference values, corresponding to about 4 seconds in the TSC simulation.

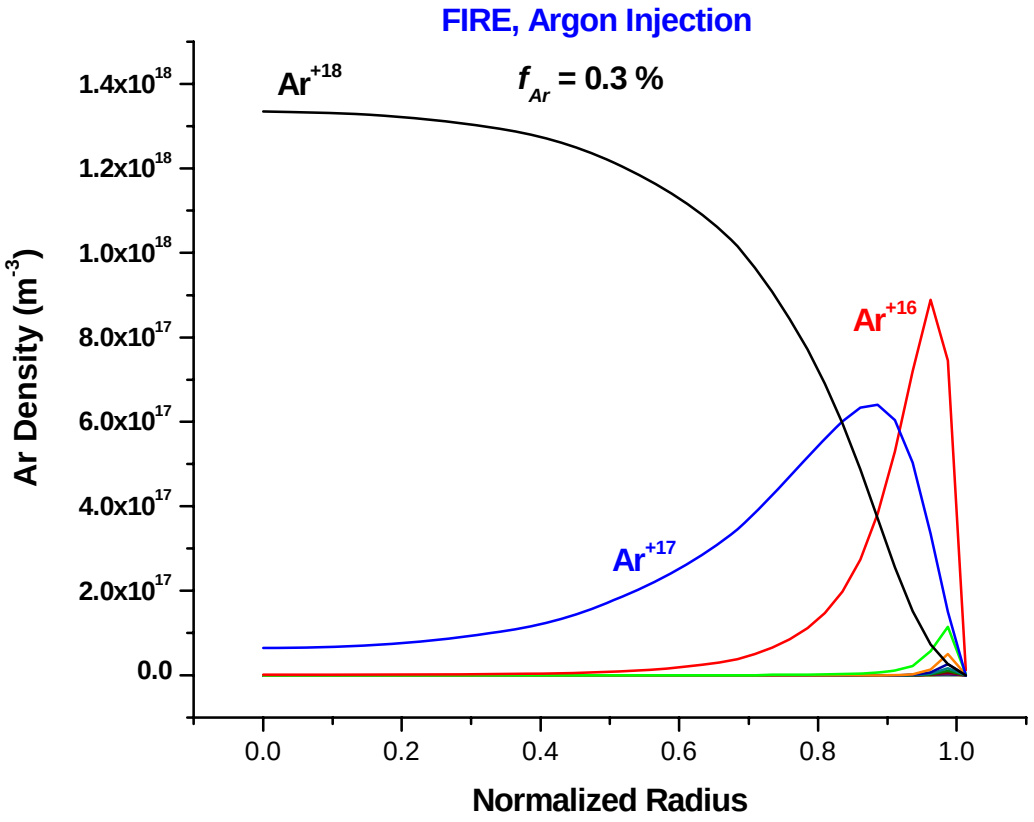

Figure 14: Profiles of Argon charge states following Ar injection.

As expected, Ar is almost fully ionized in the plasma core, while the highly radiating Lithium-like and Helium-like charge states are concentrated in the plasma edge. Our simulation predicts that for the reference concentration of 0.3%, the total radiated power by the Ar impurities (including bremsstrahlung and line radiation) is 45.2 MW, which is about 20%-30% higher than the predictions of earlier 0-D (fixed profiles) simulations. This suggests that lower Ar concentrations may be adequate to meet the needs of the FIRE design.

In addition to the determination of the radiating properties of the seeded Ar impurities, our simulations identified a number of critical issues that must be addressed before impurity seeding can be safely adopted as part of the reference operating scenario of FIRE. These include: a) the potential of edge thermal instabilities following Ar injection which were observed in several of our simulations and which can collapse the edge temperature profile and, eventually, terminate the plasma; b) the sensitivity of our predictions to the edge temperature assumptions, underlying the need for a realistic and accurate pedestal boundary condition model; c) the

importance of the edge ion and electron thermal transport assumptions; and d) the possibility of core impurity accumulation due to neoclassical effects arising from peaked density profiles.

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D. NTCC PARTICIPATION

1. Introduction

The Georgia Tech Fusion Research Center has been participating in the National Transport Code Collaboration (NTCC) project since 1999. During this period, we have reviewed several submitted modules and have submitted two modules of our own.

2. Submitted Modules

NBEAMS module (submitted in 1999)

We have submitted the module NBEAMS, which contains routines for the calculation of neutral beam heating and current drive parameters in tokamak plasmas. It calculates the NB deposition profile, profiles of the NB heating power deposited to the background ions and electrons, various NB current drive quantities and, optionally, several beam-target fusion quantities.

The NBEAMS module was originally developed by the author for the ITER systems code SUPERCODE[1] and it was used extensively during the ITER CDA and EDA design activities. The

calculation is approximate (it is based on the diffuse beam approximation and it assumes a simplified flux surface geometry) but computationally fast. Extensive benchmarks with more detailed codes (ACCOME, NFREYA, TRANSP) have consistently shown excellent agreement

Before submission to the NTCC library, the module was updated, tested and extensively modified to conform to the NTCC standards. The module is intended to be used in transport simulations where a realistic NB heating profile is desired, but the full accuracy of more detailed (and computationally expensive) codes such as the NUBEAM Monte Carlo fast ion package (also submitted to NTCC) is not necessary.

GTNEUT module (submitted in 2004)

We recently submitted the GTNEUT module. This module contains our 2-D neutral transport code GTNEUT[2] which is based on the Transmission & Escape Probability (TEP) method[3]. While other neutral modules have been submitted to NTCC (NUT, FRANTIC), the computational speed of GTNEUT and its ability to handle complex geometries, like the ones encountered at the edge of tokamak plasmas, make it an ideal tool for edge plasma and core fueling simulations. The code has been extensively benchmarked against Monte Carlo and experiment[4-6].

3. Reviewed Modules

We have reviewed five NTCC modules and are in the process of reviewing a sixth. Each review usually consists of ensuring that the module conforms to the NTCC standards, followed by building, testing and installing the module on various computer platforms (usually workstations running different versions of the UNIX operating system). In addition to the test programs supplied by the module developers, we often write our own testing routines to exercise the capabilities of the module and to test various usability and implementation issues. During the review process, we communicate with the module developers to resolve any problems which we identified, and offering suggestions for improvements. Following each review, a standardized evaluation form is filled out and submitted to the NTCC Committee chairman with our comments and recommendation, and then the module is put to a vote by the chairman for the NTCC committee members.

We have reviewed the following modules, which are now accepted as part of the NTCC module library:

FPREPROC (reviewed and approved in 2000)

The Fortran Pre-processor module *FpreProc* (submitted by Doug McCune of PPPL) is a set of PERL scripts that pre-process Fortran (f77 and f90) code using GNU's *gcc* compiler. It provides a simple and elegant code pre-processing mechanism for specifying conditional compilation and/or compile time macro expansion, greatly aiding the ability to maintain from a single source a code which compiles and runs correctly on many types of target architectures. The advantage of *FpreProc* over comparable tools such as *fpp* is that it should work the same on all systems (vendor-supplied Fortran preprocessors can be system and architecture dependent).

PORTLIB (reviewed and approved in 2000)

The *PortLib* portability tools library (submitted by Doug McCune of PPPL) contains a number of useful routines and functions that perform certain system tasks. Some of these tasks are very common (e.g. elapsed CPU time, access to command line arguments, access to the shell, etc.) and can be found in almost every code. While most Fortran compilers provide access to such routines, the interface is not standard forcing the developer to re-write part of his/her code and use conditional compilation every time the code is ported into a new environment The *PortLib* library provides a standardized interface to these routines, greatly simplifying the porting of codes to new platforms.

XPLASMA (reviewed and approved in 2002)

The XPLASMA module (submitted by Doug McCune of PPPL) is a set of routines that provides a representation standard for MHD equilibria in axisymmetric plasmas using spline interpolation. The biggest advantage of XPLASMA is the elimination of the need that physics modules share the same MHD representation and grids as the main transport code. This makes it easier to port physics modules into transport codes, since one does not have to write interface code (a non-trivial task usually) to translate from one representation to another (e.g. from an inverse equilibrium representation to an *R,Z* representation and vice versa). In addition, the XPLASMA module is an integral part of the Monte Carlo fast ion code NUBEAM which has also been submitted to NTCC.

NUT (reviewed and approved in 2003)

NUT (submitted by P.M. Valanju, University of Texas). NUT is a fast, semi-analytic algorithm for 3-D neutral transport in 3-D plasmas and can be a useful module for calculating fueling of fusion reactors.

LSC module

The Lower Hybrid Simulation Code module **LSC** (developed by David Ignat and submitted by Doug McCune, PPPL), has been submitted to NTCC and is considered a high priority module, since there is a need for plasma heating source modules.

Our work with the LSC module has been more than a typical NTCC review, since the current version of the module is not up the NTCC standards. Besides coding and portability issues–which are not that hard to rectify–the module is missing a number of potentially significant physics effects, the most important of which is the absence of trapped electron effects.

We have a developed a routine to calculate trapped electron effects for LH current drive based on the work by Ron Cohen[7]. Additionally, a trapped electron correction routine from the ACCOME code based on the adjoint technique and developed by Karney and Fisch has been provided to us by Paul Bonoli from MIT. The revised LSC code will be re-submitted to NTCC after benchmarking with other codes such as the ACCOME code have been completed.

CYTRAN (reviewed in 2004, pending final approval by the NTCC committee)

The **CYTRAN** module (submitted by Wayne A. Houlberg, ORNL) calculates the radial profile of synchrotron/cyclotron radiation loss (or gain) for toroidal plasmas. While cyclotron radiation is rather small in present-day experiments—at least compared to bremsstrahlung and impurity radiation—and is usually neglected in most numerical simulations, its strong dependence on plasma temperature and magnetic field strength will increase its importance in next-generation burning plasma experiments. Cyclotron radiation exhibits strong non-local effects where the wall-reflected radiation is often reabsorbed in the plasma edge giving rise, in effect, to a local *heating* rather than *cooling* term in the outer plasma. This means that the commonly used approach in a number of radial transport codes of taking a global formula for the cyclotron radiation loss (e.g., Trubnikov's expression) and applying it locally is questionable at best. A true radial transport approach is needed to accurately account for the radial profile of the cyclotron radiation cooling or heating rate. CYTRAN is such a routine and has been used in the WHIST code and its derivatives for a couple of decades.

FRANTIC (under review)

The FRANTIC module (submitted by Doug McCune of PPPL) calculates neutral particle transport in cylindrical plasmas based on the semi-analytic technique by S. Tamor[8]. This module is currently under review.

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