A Possible Scenario to Commercial Tokamak Power Reactors Based on the Results of the UWMAK-I and II Conceptual Design Studies

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June 27, 1975
(revised November 25, 1975)

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FUSION TECHNOLOGY INSTITUTE

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Table of Contents

		Page
I.	Introduction	1
II.	Summary of Major Features of the UWMAK-I and UWMAK-II Tokamak Fusion Reactors	1
III.	Listing of the Most Critical Problems of UWMAK Type Reactors . .	19
IV.	Possible Scenario for Solving Critical Technology Problems and Achieving Commercial Fusion Reactors	29
	Appendix A - Abbreviations and Names for Tokamak Experiments . .	42
	Appendix B - Letter Designations for Tokamak Test Reactors and Tokamak Power Reactors	43

Abstract

The basic design features of the conceptual tokamak design studies, UWMAK-I and UWMAK-II, are reviewed with an eye towards discussing the major technological problems presented by such systems. We then describe an interesting scenario of test reactors which might allow an orderly solution to the general technology problems we have identified.

I. Introduction

Recent studies have revealed many exciting possibilities for the use of tokamak fusion reactors. However, these studies have also added new problems to the "old" list of difficulties that must be solved before economical fusion systems can be realized. The purpose of this paper is to highlight the results of extensive studies by the University of Wisconsin Fusion Reactor Study Group concerning two large (5000 MW_{th}) DT Tokamak reactors, UWMAK-I[1,2] and UWMAK-II.[3]

We first review the basic features of both conceptual reactor designs and then highlight the problems which we have found are critical to reaching a large scale fusion reactor economy based on these reactor concepts. We limit our remarks in this paper to the major problems than present "go" or "no go" situations. We then exercise our literary options and propose a possible scenario or sequence of test reactors which might allow an orderly solution to the general technology problems we discuss. For clarification, the reader is cautioned to note that this scenario has no official basis and represents only the view of the authors.

II. Summary of Major Features of the UWMAK-I & II Fusion Reactors

Tables 1 through 12 are given here to establish a common ground for a discussion of these two reactors. More extensive descriptions are given in several papers and comprehensive reports.[1-8]

The first point is to recognize the similarities of both reactors. They are fueled with deuterium and tritium and produce 5000 MW_{th} during the plasma burn time of 90 minutes. Both reactors have a double null poloidal divertor and are heated to ignition by a combination of ohmic heating and neutral beams.

Fueling is accomplished by solid DT pellets. Both reactor designs use 316 stainless steel as the blanket-shield structural material and a mixture of steel, Pb, and B_4C in the shield region. The anticipated lifetime of the 316 SS is 2 years in both designs and this value is determined by a loss in ductility. The superconducting magnets are TiNb cryogenically stabilized with copper. The maximum magnetic field in the coils is 86 kG and the magnetic field on axis is 38 kG for I and 36 kG for II. The energy storage unit to drive the air core transformer uses superconducting magnets with a specially designed switching sysyem. A sodium secondary loop is used in both reactors and serves the auxiliary role of a "thermal flywheel" to level the electrical load delivered to the turbine. A steam turbine is used in both systems. The estimated cost of both reactors is found to be in the $950-970 kWe range and the cost of electricity generation is 21 mills/kwh in UWMAK-I and 25 mills/kwh in UWMAK-II.

There are also many differences between these reactor designs. The plasma physics scaling laws in I were taken to be a modified form of neoclassical scaling whereas in II, trapped particle modes, and particularly the trapped ion mode, were assumed to govern transport in the reactor regime. Microinstability scaling laws result in a increased energy load to the divertor (from 250 MW_{th} in I to 750 MW_{th} in II). A carbon curtain is used in UWMAK-II to protect the plasma from impurities and the first wall from surface erosion. The current rise phase is 10 seconds in II whereas it was 100 seconds in I. This is largely because the divertor coils are inside the TF coils in II and require only a 1 MW-hr energy storage unit. In UWMAK-I, these coils were outside the TF coils and windings are outside the TF coils in both designs.

The breeding material is $LiAlO_2$ and the coolant is He in UWMAK-II whereas liquid Li serves both functions in UWMAK-I. Beryllium is required to achieve a breeding ratio of greater than 1 in UWMAK-II whereas without beryllium, the breeding ratio is 1.49 in UWMAK-I. The maximum structure temperature in II is 680°C compared to 500°C. This translates into a higher temperature coolant and eventually higher thermal efficiency (35% for II vs 32% for I).

There are 24 TF coils in II versus 12 in I to reduce the field ripple at the plasma surface to less than 1%. The shape of the TF coils is a constant tenstion "D" shape in I and an extended, non-constant tension "D" shape in II. The latter design permits the removal of individual blanket modules without having to move a TF coil. Other differences between the two designs can be found by examining tables 1 through 12.

Table 1

General Parameters and Reactor Dimensions

	Units	I	II
Thermal Power	MW_t	5000	5000
Electrical Power	MW_e	1473	1716
Primary Coolant	-	Li	He
Blanket Structure Material	-	316 SS	316 SS
Power Cycle	-	Li/Na/Steam	He/Na/Steam
Breeding Material	-	Li	$LiAlO_2$
Breeding Ratio	-	1.49	1.06-1.19
Plasma Radius	m	5	5
First Wall Radius	m	5.5	5.5
Major Radius	m	13	13
Blanket Thickness	m	0.73	0.89
Shield Thickness	m	0.78	0.99
Maximum Core Diameter	m	21.5	28.3
Maximum Reactor Height	m	23.4	33.6
Maximum Reactor Diameter	m	43.0	52
Total Reactor Weight	Tonnes	60,226	61,168

Table 2

Parameter	Unit	I Modified-Neoclassical	II Trapped Ion Mode
Plasma Current	MA	21	14.9
Fuel Density	m^{-3}	8×10^{19}	7.3×10^{19}
Electron Density	m^{-3}	8.6×10^{19}	7.7×10^{19}
Ion Temperature	keV	11.1	13.2
Electron Temperature	keV	11	12
Energy Confinement Time	sec	14.2	3.6
Particle Confinement Time	sec	14.2	8.7
Safety Factor, q(a)	-	1.75	2.3
Poloidal Beta	-	1.07	2.3
Toroidal Beta	-	0.052	0.065
Fractional Burnup	%	7.2	4.85
Magnetic Field on Axis	kG	38.2	35.7
Poloidal Magnetic Field	kG	8.4	5.96
Plasma Shape (nominal)	-	'circular'	'circular'
Plasma Radius	m	5	5
Major Toroid Radius	m	13	13
Aspect Ratio	-	2.6	2.6
Plasma Volume	m^3	6,415	6,415
Total Chamber Volume	m^3	7,700	7,700
First Wall Area	m^2	2,823	2,823
Divertor	-	'double null'	'double null'
Burn Time	sec	5400	5400
Recharge Time	sec	390	330
Total Cycle Time	sec	5790	5730
Initial Heating	-	ohmic	ohmic
Secondary Heating	-	Neutral Beams	Neutral Beams
Number of Neutral Beams	-	20	16
Energy of Neutral Beams	keV	500	750
Current per Injector	A	1.5	16.7
Total Power of Inj. Beams	MW	15	200
Time of Injection	sec	11	10
Time from Initiation of Cycle to Steady State Operating Conditions	sec	120	30
Fueling Mechanism	-	pellets	pellets

Table 3

Blanket and Shield Structure

Parameter	Unit	I	II
	Blanket		
Material	-	316 SS	316 SS
First Wall Thickness	mm	2.5	10.5/1.5*
Structural Material in Breeding Zone	%	5	10
Metallic Temperature Range	°C	359-500	450-654
Thermal Stress - 1st Wall	psi	8424	960
Coolant Stress Level	psi	4572	7143
Number of Cycles/Year (80% P.F.)	-	4350	4400
Total Structural Weight	Tonnes	6,250	6376
First Wall Protection	-	none	C curtain
	Reflector		
Material	-	316 SS	Carbon
Minimum Distance from 1st Wall	cm	51	33
Temperature Range	°C	359-670	685-935
Total Reflector Wt	Tonnes	4210	1188
	Shield		
Materials	-	Pb/B_4C/316 SS	Pb/B_4C/316 SS
Coolant	-	He	He
Coolant Pressure	psi	735	735
Temperature Range	°C	50-200	50-200
Total Structural Wt	Tonnes	21,450	29,738

*See reference 3 for first wall structure

Table 4

Primary Coolant Parameters

Parameter	Unit	I	II
Coolant	-	Li	He
First Wall Heat Flux	W/cm^2	22.6	3.42
Neutron Wall Loading	MW/m^2	1.25	1.16
Inlet Temperature	°C	359	371
Outlet Temperature	°C	489	650
Pressure	psi	670	750
Coolant Velocity in Blanket	cm/sec	4	800-11800
Mass Flow Rate	kg/hr	3.24×10^7	1.5×10^7
Weight of Coolant in Blanket	Tonnes	1159	4
Pumping Power	MW_e	32	294*

*Thermal pumping power

Table 5

Tritium, Breeding and Inventory Parameters

Parameter	Unit	I	II
Breeding Medium	---	Li	$LiAlO_2$
Breeding Ratio	---	1.49	1.06 - 1.19*
Lithium -6 Concentration	%	7.42(nat)	90
Lithium Inventory	Tonnes	1159	90
Mass of Breeder Material	Tonnes	1159	688
Tritium concentration in Breeder	appm	12.5	1.5
Maximum Temperature of Breeding Material	°C	489	1097
Minimum Temperature of Breeding Material	°C	359	521
Tritium Separation Technique	---	Y metal	O_2 in He
T_2 inventory-blanket/primary coolant	kg	8.7	0.12-0.725
" " -divertor Li	kg	0.008	0.008
" " -vacuum pumps	kg	0.005	0.125
" " -separation unit	kg	1.73	3.92
" " -fueling - 1 day	kg	8.4	12.85
" " -storage - 10 days (burnup only)	kg	6.72	6.23
" " -Total	kg	25.56	23.13-23.63
Heat Exchanger Area (IHX)	m^2	19000	43200
" " " (Steam generator)	m^2	37,056	34000
T_2 leakage	ci/day	10	1
Neutron multiplying medium	---	none	Be
Mass of neutron multiplier	Tonnes	none	433
Maximum temperature-multiplier	°C	---	595
Minimum " - "	°C	---	581
Burn-up of neutron multiplier	% / year	---	0.2

*The range given is based on several neutronics calculations. The value 1.19 is based on 1-D cylindrical transport calculations assuming 5% structure (equivalent) in the breeding zones. The value of 1.06 is based on detailed 3-D Monte Carlo calculations and a rigorous representation of the blanket.

Table 6

Neutronics Information

Parameter	Unit	I	II
Neutron Wall Loading	MW/m^2	1.25	1.16
Energy per Neutron	MeV/n	20.08	21.59
Maximum Nuclear Heating - First Wall	W/cm^3	6.0	4.86
Maximum Gamma Heating - First Wall	W/cm^3	6.8	6.67
Neutron Attenuation through Blanket	MeV/MeV	0.0126	0.012
" " " " and Shield	MeV/MeV	1.5×10^{-5}	8.84×10^{-7}
Total Nuclear Heating in Magnet	Watts	3840	256
Radioactivity - 1st wall[(a)]	Ci/cm^3	90.1	56.8
Total Radioactivity - 1st Wall[(a)]	Ci	1.09×10^9	2.29×10^9
Total Radioactivity in Reactor	Ci	3.8×10^9	5.81×10^9
Afterheat - 1st Wall[(a)]	$Watts/cm^3$	0.66	0.23
Total afterheat - 1st Wall[(a)]	MW	7.5	13
Total Afterheat in Reactor	MW	29	65

[(a)] $t = 0$, $t_{irrad.} = 2$ year

Table 7

Radiation Damage Conditions

Parameter	Unit	I	II
1st Wall Material	---	316 SS	316 SS
Neutron Wall Loading	MW/m^2	1.25	1.16
Damage Rate[(a)]	dpa/yr	11.6	10.7
Helium Production Rate [(a) (b)]	appm/yr	298	264
Hydrogen Production Rate [(a)(b)]	**appm/yr**	611	563
Max. Swelling Rate - 1st Wall[(a)]	% year	0.13	~0.1
Maximum Temperature - 1st Wall	°C	500	540
Charged Particle Flux - D [(c)]	ions/cm^2/sec	6.4×10^{13}	1×10^{13}
" " " - T	" " "	6.4×10^{13}	1×10^{13}
" " " - He	" " "	4.9×10^{12}	5.1×10^{11}
Erosion Rate - Sputtering[(d)]	mm/yr	0.2(max) 10^{-5}(min)	0.12 (max) 10^{-5}(min)
" " - Blistering	" "	0.24	0 (steel)
Damage in Magnet	dpa/yr	6×10^{-5}	2×10^{-6}
Assumed 1st Wall Lifetime	years	2	2
Assumed Failure Mechanism - 1st Wall	----	loss of ductility	loss of ductility
Replacement rate in blanket (total)	tonnes/yr	433	1240
Reflector Material	---	316 SS	Carbon
Damage Rate in Reflector	dpa/**yr**	~1	1.8
Maximum Temperature - Reflector	°C	670	935
Helium Production Rate - Be (100% P.F.)	appm/yr	---	3040
Swelling of Be	%/yr	---	~5-10

Table 7 (cont)

Parameter	Unit	I	II
Temperature range - Be	°C	----	581-595
Damage rate in $LiAlO_2$	dpa/yr	----	Not Avail.
Helium production rate - $LiAlO_2$	appm/yr	----	~30,000
Swelling - $LiAlO_2$	%/yr	----	~10-20
Temperature Range - $LiAlO_2$	°C	----	521-1097

[a] Based on cross sections used for UWMAK-II , 100% P.F.

[b] Including contribution from impurities.

[c] To first solid member of reactor (carbon curtain).

[d] Using highest reported neutron sputtering values.

Table 8

Magnet Parameters

Parameter	Unit	I	II
Toroidal Field Coils			
Number	----	12	24
Superconductor	----	TiNb	TiNb
Stabilizer	----	Cu	Cu
Cu/Superconductor ratio	Vol. ave	34	36
Support Material	----	316 SS	316 SS
Max. Current Density	A/cm^2	1318	2200
Max. Magnet Field	kG	86.6	83
Max. Stress in Support Material	MPa	276	276
Fraction of Yield Stress	----	0.67	0.67
Maximum Magnet Bore	m	21.5	28.3
Maximum Magnet Thickness	m	0.93	0.98
" " Width	m	1.7	0.95
Total wt per coil	tonnes	872	710
Total wt TF coils	tonnes	10904	16996
Stored Energy	GJ	158.4	223.2
Ohmic Heating Coils			
Number		10	18
Superconductor		TiNb	TiNb
Stabilizer		Cu	Cu
Cu/Superconductor ratio	Vol. ave	30	30
Support Material	----	304 SS	304 SS
Max. Current Density	A/cm^2	3460	2200
Max. Magnetic Field	kG	75.9	57.2

Table 8 (cont)

Parameter	Unit	I	II
Maximum stress in support	MPa	276	138
Fraction of yield strength	----	0.67	0.33
Maximum bore	m	16.08	16.44
Maximum magnet thickness	m	0.995	0.2
Maximum magnet width	m	1.07	1.32
Maximum wt/coil	tonnes	108	86.3
Total wt OH coils	tonnes	1718	1118
Stored Energy (max)	GJ	13.2	1.9
Divertor Coils			
Number	--	8	12
Superconductor	--	NbTi	NbTi
Stabilizer	--	Cu	Cu
Cu/NbTi ratio	Vol. ave	115	115
Support material	----	304 SS	304 SS
Maximum current density	A/cm^2	4730	2200
Maximum magnetic field	kG	75.9	52.7
Maximum stress in support material	MPa	276	138
Fraction of yield strength	--	0.67	0.33
Maximum bore	m	44	38
Maximum magnet thickness	m	1.85	0.88
Total Wt. Divertor Coil	tonnes	8500	2165
Stored Energy (max)	GJ	67.5	6.96

Table 9

Power Cycle Parameters

Parameter	Unit	I	II
Primary Coolant	-	Li	He
IHX Material	-	304 SS	Incoloy-800 304 SS
IHX Area	m^2	1900	43200
IHX Tube Wall Thickness	mm	0.89	
Secondary Coolant	-	Na	Na
Na Flow Rate	kg/hr	1.18×10^8	0.51×10^6
Na Inlet Temperature - IHX	°C	336	322
Na Outlet Temperature - IHX	°C	456	567
Pressure	psi	35	35
Total Wt Na(1)	Tonnes	17826	8400
Steam Generator Material	-	Croloy-2 1/4 Inconel-600	Croloy-2 1/4 321 SS
Steam Generator Area	m^2	37,056	34,054
Steam Generator Tube Wall Thickness	mm	0.89	
Energy Storage Method for Power Load Leveling	-	Na Flywheel	Na Flywheel
Power Generating Method	-	Steam	Steam
Inlet Temperature Steam	°C	218	399
Outlet Temperature Steam	°C	404	510
Pressure (Steam)	psi	2000	2400
Flow Rate	kg/hr	7.56×10^7	6.2×10^6
Gross Power Output(2)	MW_e	1681	1814
Auxiliary Power Required	MW_e	208	98
Net Power Output	MW_e	1473	1716
Overall Efficiency	%	32	36

(1) Including Thermal Flywheel

(2) Averaged Over Entire Burn Cycle

Table 10

Materials Resource Requirements[a]

Element	Tonne/MW_e I 1475 MW_e	Tonne/MW_e II 1710 MW_e
Al	0.68	2.80
Be	-	0.58
B	1.07	2.50
C	0.30	2.86
Cr	10.55	5.51
Nb	0.09	0.09
Cu	7.42	6.65
He	0.09	0.05
Fe	86.87	63.02
Pb	13.9	11.6
Li	1.15	4.95[b]
Mn	1.13	0.75
Hg	0.002	-
Mo	0.70	0.59
Ni	7.95	5.24
Na	12.13	5.74
Si	-	0.007
Ti	0.05	0.06
Y	0.003	0.002
Zr	0.07	0.06
Total	144.2	113.1

(a) Total plant requirements including replacement amounts.
(b) Natural Li equivalent

Table 11a

Economic Comparisons ($\$/kW_e$)

Account Title	I	II
Direct Costs		
Non Depreciating Assets		
Land and Land Rights	0.8	0.7
Depreciating Assets		
Special Materials	19.2	3.4
Physical Plant		
Structures and Site Facilities	94.8	94.5
Reactor Plant Equipment	388.9	453.3
Turbine Plant Equipment	115.6	93.6
Electric Plant Equipment	96.9	49.3
Miscellaneous Plant Equipment	6.4	11.2
Subtotal Physical Plant	702.6	701.9
Indirect Costs		
Construction, Facilities, Equipment and Service	16.5	14.2
Engineering Services	32.9	28.4
Other Costs	51.3	53.1
Interest During Construction	148.2	146.7
Subtotal-Indirect	248.8	242.4
Total Dollars per Kilowatt Generated	971	948

Table 11b

Economic Comparisons ($)

Account Title	I	II
Direct Costs		
Non Depreciating Assets		
Land and Land Rights	$1,200,000	1,200,000
Depreciating Assets		
Special Materials	28,290,000	5,820,000
Physical Plant		
Structures and Site Facilities	139,807,000	161,590,000
Reactor Plant Equipment	573,636,000	775,179,000
Turbine Plant Equipment	170,580,000	160,000,000
Electric Plant Equipment	142,859,000	84,218,000
Miscellaneous Plant Equipment	9,410,000	19,110,000
Subtotal Physical Plant	1,036,292,000	1,200,100,000
Indirect Costs		
Construction Facilities, Equipment and Services	24,300,000	24,300,000
Engineering Services	48,500,000	48,500,000
Other Costs	75,600,000	90,800,000
Interest During Construction	218,618,000	250,923,000
Subtotal	367,018,000	417,000,000
Total Plant Investment	1,432,800,000	1,621,640,000
Cost per Kilowatt Generated	971	948

Table 12

Cost of Generating Electricity

Cost Item	Unit Cost (mills/kw-hr) I	II
Operations and Maintenance	1.1	4.6
Fuel	0.01	0.01
Return on Capital	20.1	20.2
Total	21.2	24.7

III. Listing of Most Critical Problems of UWMAK Type Reactors

A thorough discussion of all the technologies that must be mastered for successful operation of UWMAK type reactors is beyond the scope of this article. However, the five or six most troublesome problems in each area that present the most serious obstacles to routine operation of DT tokamak fusion reactors are listed. Table 13 has been constructed to summarize these problem areas and we will discuss each category briefly.

A. Plasma

Plasma physics remains the primary problem of fusion research and the major experiments through 1980 are all aimed at developing a better understanding of plasma behavior at densities and temperatures similar to those required in reactors. From a reactor viewpoint, we need to know more about plasma startup, heating, and D-T burning. In turn, this means we need to understand the transport scaling laws characteristic of reactor grade plasmas and the methods by which long burn times can be achieved.

The startup problem will be characteristic of all high current tokamaks and relates to the power requirements during the current rise phase. To minimize the power requirements, one would like a controlled $\frac{dI_p}{dt}$ on the order of 1-10 MA/sec, and preferably on the lower side. Proposals for accomplishing this include moving limiters, programmed verticle fields to move the plasma away from a fixed limiter, and a moving magnetic separatrix. The ATC device provides information about programmed vertical field effects on the plasma but the true tests of these methods will come on machines like PLT, PDX, and the larger, 1980 experiments.(See Appendix A for list of plasma experiments)

Auxiliary heating appears a certain requirement for reaching ignition condition and adiabatic compression, neutral beam heating, and RF heating are primary candidates. Compression has been demonstrated on ATC and beams have been used on both ATC and ORMAK with success. Less experience is available

Table 13

Summary of Major Problem Areas that Need to be Understood before Successful Operation of UWMAK Type Reactors

General Area	Problem Area
1.) Plasma	a.) Start Up
	b.) Auxiliary Heating (either neutral beams or RF)
	c.) Scaling Laws
	d.) Impurity Control
	e.) Particle Removal (divertor?)
	f.) Fueling
	g.) Plasma Burn Dynamics and Control
2. Materials (non Radiation Damage)	a.) Fabrication of large (12-15 m diameter) vacuum tight chambers in the field
	b.) Long fatigue life of welded structures
	c.) Liquid metals in magnetic fields (pumping power and laminar flow)
	d.) Sintering of solid breeders
	e.) Reserves of neutron multiplier, Be
3. Materials (Radiation Damage)	a.) Develop alloys that can retain ductility and long fatigue life with high helium contents
	b.) Alloys which are resistant to dimensional changes due to swelling or creep
	c.) Materials to protect the first wall surface from plasma ions
	d.) Remote methods for disassembly and assembly of radioactive structures
	e.) Understanding of high fluence (10^{21}-10^{22} n/cm^2) 14 MeV neutron displacement damage in CTR blanket materials
4. Breeding and Neutronics	a.) Methods for extracting tritium from lithium at the ~1 appm level
	b.) Control of tritium release through blanket, shield, heat exchanger and power cycles
	c.) Demonstration that tritium can be collected and re-injected with a turn around time of 1 day or less
	d.) Experimental verification of 3-D calculational methods for T_2 breeding and radiation damage calculations
5. Magnets	a.) Fabrication of large bore (15-20 meter) diameter superconducting magnets
	b.) Verification of magnet stability under CTR conditions
	c.) Energy storage, switching and recovery mechanisms for plasma start up

Table 13 (con't.)

General Area	Problem Area
6. Power Cycle	a.) Method to even out energy production during time between burns b.) Demonstrate plant factors of 70-80%
7. Environment	a.) Disposal of large volumes of radioactive and contaminated structural components b.) Demonstrate tritium release rates of $\lesssim$ 10 curies/day/1000 MW_e c.) Substitutes for large demands of scarce elements d.) Successful containment and control of very large masses of liquid metals at high temperatures e.) Demonstrate that diversion of tritium is not a problem
8. Economics	a.) Ability to produce power at costs less than competitors (i.e. LMFBR, other advanced systems that are not fuel limited) b.) Achieve maintenance record that allows reliable operation

with RF heating at frequencies scalable to reactor applications. Adiabatic compression may find use in reactors although space limitations will restrict the attainable compression ratio. Neutral beam heating will be studied extensively on future machines and some of the main questions related to plasma stability under high power injection and the development of efficient, effectively steady-state, sources at beam energies of 200 keV to 700 keV. Such beams will be required for optimizing beam-plasma interactions and for beam penetration.

Relatively long burn times (>100 seconds and hopefully >1000 seconds) are desirable for an economic power reactor. From the UWMAK studies, the requisite transformer action appears to be within reach and the main problems are associated with scaling laws, impurity control, particle removal, and fueling. Impurity control is essential if long burn times are to be achieved since enhanced radiation especially from high Z impurities can cause an ignited plasma to become unignited and require, at a minimum, external energy injection for continued operation. Divertors are one possibility and a combination of divertors and low Z liners is another. Some experiments like PDX, ISX, DITE, ASDEX, and TEXTOR (if all are built) will begin to address these questions. Unfortunately, none of these machines will have a plasma current exceeding 0.5 MA and it is not clear that the edge conditions will be characteristic of much larger tokamaks.

Refueling a plasma which has a relatively short particle confinement time (like 10 seconds in a 15 MA reactor) is critical for long burn times. Yet this remains an area where little, if any, experimental information is available.

Finally, it is clearly necessary to understand the transport phenomena taking place in large tokamaks and to know the effective particle and energy containment times. Theoretical predictions assuming microinstabilities have been made on the basis of relatively crude estimates for the thermal conductivity and the diffusivity but no strong experimental confirmation is available.

B. Materials (non-Radiation Damage)

Roughly 95% of the materials in the nuclear island will not be subject to radiation damage. These materials (mainly metallic alloys) are in a rather large structure (15-20 m diameter) which must maintain vacuum tightness during extended high temperature operation and will be subject to large, periodic stresses and strains during the startup and burn phases. This is especially true if liquid metals are used for coolants. At times, these components will be so large and complex that they must be fabricated and assembled in the field. This will require materials which are not sensitive to the environment during welding and which can retain good pre-operational ductility.

In the area of breeding, there are two problems associated with low tritium inventory blankets. The first lies with the use of solid breeders, especially ceramics. These usually rely on small particle size to allow T_2 diffusion into a carrier gas. Unfortunately, ceramics have low thermal conductivities which promote high temperatures (and high temperature gradients) and sintering into large particles. The latter event can lead to a large T_2 inventory. The second problem is the need for a neutron multiplier to compensate for low lithium inventories. This means that Be must be used (neglecting fissile or fertile isotopes) and the use of large amounts of Be could place severe demands on the world reserves, perhaps exceeding the world supply.

C. Materials (Radiation Damage)

This is probably the most difficult problem to be solved after the plasma physics. The main problems will be to develop alloys which can retain reasonable (>1-2%) high temperature uniform ductility and fatigue life (>10,000 cycles) after experiencing 14 MeV neutron fluences of $>10^{21}$ n/cm^2. The generation of high internal helium concentrations (up to a few hundred atomic parts per million per year) tends to aggravate this situation. It will be necessary to develop metals and nonmetals (graphite, ceramics, etc.) which can retain their dimensional stability under the same irradiation conditions as described above. It is also quite necessary to subject potential CTR materials to high ($>10^{21}$ n/cm^2) fluences of 14 MeV neutrons to make sure that there are no new 'surprises' in the basic damage state.

The irradiation of the CTR blanket will induce large amounts ($\sim 10^9$ curies) of activity in the first hour of irradiation. Many components will undoubtedly fail for nuclear and non-nuclear reasons and must be changed quickly and efficiently. This will require extensive methods and equipment to be developed to perform complex cutting, bending and welding operations remotely. Finally, the high flux of charged particles and photons from the plasma will cause part of the first wall surface to be sputtered or blistered away, not only causing the wall to be thinned but also contaminating the plasma. Methods for protecting the first wall from the energetic plasma particles (neutrals as well as ionized species) must be found. The use of carbon curtains(9), honeycomb structures(10) or neutron spectral shifters(11) may have merit here.

D. Breeding and Neutronics

The process of breeding tritium at low temperatures in lithium bearing compounds is well established. However, very little experience has been obtained from high temperature systems where the breeder is liquid metal or a ceramic compound. The problem lies in being able to limit the equilibrium concentrations of tritium to ~1 kg which means that concentrations of ~1 appm must be kept. Methods for extracting tritium at such low concentrations on a routine basis must be demonstrated. Even at such low concentrations, diffusion of tritium through pipes, valves, welds, or even through metallic components themselves must be controlled. Another area for concern is whether one will be able to inject, partially burn, collect, extract, refabricate and re-inject the tritium atoms in a turn around time of ~1 day or less. If the turn around time is much longer, the tritium inventory could exceed tens of kg's (~10^8 curies).

Probably the only major unsolved problem in the neutronics area will be the experimental verification of T_2 breeding, nuclear heating and radiation damage predicted by sophisticated 3-D codes. Such tests are essential if we are to rely on such calculational methods to predict accurately what will happen in the complex geometrical structures of a tokamak reactor.

Magnets

Perhaps the next most critical technological problem to be solved after plasma physics and radiation damage is the design, fabrication and operation of large superconducting magnets. For the UWMAK series this means maximum bores of 22-28 m and structures which weigh ~700-900 tonnes each when all of the support material is taken into account. The difficulty in removing or repairing such a magnet after it has become activated means that initially, great care must be taken to insure against failure. Long term stability and sufficient safety margins must be established before an electric utility would risk spending 1-2 billion dollars on such a reactor.

A significant problem is the transfer of large amounts of stored energy (~$10^3 MW_e$) in a few seconds to the OH coils. This energy must not only be stored efficiently to avoid unreasonable demands on local electrical grids but it must also be recovered with a high enough efficiency to reduce the total costs of the system. Superconducting energy storage and homopolar generators may help in this regard.

Power Cycle

One unique problem of large (~1000-2000 MW_e) Tokamak reactors is to find a way to deliver a constant flow of energy to the electrical grid. The problem stems from the pulsed nature of tokamak operation. The down period can last as much as several minutes and suitable methods for thermal energy storage must be developed to offset this. Thermal flywheels are one way this may be accomplished but they require large storage tanks and large inventories of liquid metals (~2 tonnes per MW_e).

Power reactors must also demonstrate that they can deliver power 70-80% of the time or the electrical generating costs will rise to very high values. Successful operation of hundreds of electrical, thermal and mechanical systems in a fusion reactor will have to be documented before commercial operations can commence.

Environment

The two most serious problems related to environmental considerations will be the successful containment of large inventories of tritium and the ultimate disposal of large volumes (~$100 m^3$/yr) of highly radioactive structural materials. The tritium inventory in tokamak reactors will exceed 10 kg (10^8 curies) even if solid breeders are used in the blanket. Holding the release of this activity to ~11 curies/day (a factor of 10^{-8}) will require the development of countless procedures, equipment and detection methods before licensing of DT fusion reactors will be a routine matter.

The removal and disposal of defective blanket components or other activated CTR structural materials must also be closely scrutinized. It has been suggested that the use of Al could mitigate this problem because of the short half life associated with the main transmutation products. Unfortunately, Al also will have an extremely short mechanical life in the reactor because of its susceptability to helium embrittlement in typical CTR neutron environments. Vanadium alloys may be a partial answer to this question but its in-service life in the presence of small amounts of oxygen and carbon is also questionable and an industry does not exist to produce V alloys in thousands of metric tonne quantities required for a mature fusion reactor economy.

Fusion power may also be subject to criticism if it places large demands on scarce elements in the world. The widespread use of Be is a good example of this problem. It has been calculated that if as much as 10^6 MW_e were supplied by UWMAK-II type reactors in the year 2020 (~30% of U.S. projected demand) it would require 10 times the world's known reserves <u>and</u> resources. Significant demands on the refractory metals could also occur and local shortages of some rather basic elements, such as Cr, Mn, or Ni, could develop. Careful studies are required to identify these problems early in reactor design and there is the need to vigorously investigate the use of more abundant substitutes.

The potential for liquid metal fires and the subsequent spread of radioactive isotopes must also be carefully studied for fusion reactors. The UWMAK-I and II reactors each have between 10^4 and 2×10^4 tonnes of liquid metals at temperatures from 350-650°C. It is hoped that the LMFBR programs around the world will demonstrate safe handling of large masses of liquid metals by the year 2000, but the added feature of high magnetic fields and significant tritium concentrations in fusion systems may complicate this situation even further.

Finally, the possibility that tritium could be stolen or diverted to terrorist groups for subversive purposes must be considered. As yet, a "clean" fusion bomb (without using a fissionable trigger) has not been produced. However, if laser or relativistic electron beam implosion techniques advance sufficiently, it may be possible to set off a limited nuclear explosion without possessing fissionable material. The consequences of even a limited explosion, or the threat of one, cannot be taken too lightly.

Economics

Ultimately, fusion power will be important only if it can generate electricity at a cost that is less than or comparable to the costs of competing forms of advanced energy systems (LMFBR, central station solar power, coal). It is important however that fusion costs be compared with other sources of central station electric power generation that are <u>not</u> fuel limited. For example, the present generation of light water fission reactors are not truly competitive with fusion because the long term ^{235}U supply is inadequate. Thus, as stated above, fusion costs must be competitive with those of the LMFBR, central station solar, and possibly coal but not with hydroelectric, natural gas plants, oil fired plants, or LWR's. Closely related to this will be the examination of maintenance records for the first few fusion reactors and the plant factors that they achieve. Such operating experience cannot be simulated in separate component tests; the whole reactor must operate in one piece to answer the detractors of any new power source.

IV. Possible Scenairo for Solving Critical Technology Problems and Achieving Commercial Fusion Reactors

The complexity and magnitude of the problems facing commercialization of fusion reactors means that many reactors and at least 20 years will be required to address the areas discussed in the earlier sections. We will briefly discuss here a possible scenario that might be use to insure that the problems are solved in a timely fashion. We will address the period from 1980 to 2020 in our remarks.

a.) Past and Present Tokamak Research

The first work on tokamaks took place in the Soviet Union. The first device to give encouraging results was T-3. Shortly thereafter the ST tokamak at PPPL and T-4 in the Soviet Union raised the operating current to 100-200 kA. The TFR experiment in France has now achieved 400 kA. Many devices by several countries have been constructed in the intervening years so that in 1975 there are two machines (PLT and T-10) that will operate at about 1 MA plasma current. These last machines should have plasmas operating well into the trapped electron regime and bordering on the trapped ion regime.

Non circular cross sections are being addressed by General Atomic with Doublet-II-A and D-III. Impurity control will be investigated with DITE at Culham, PDX at PPPL and with the ASDEX machine at Garching. Plasma wall interactions are the main purpose for machines like ISX at ORNL and TEXTOR, proposed by Julich. Neutral beam heating has been studied on ORMAK and ATC and there are plans for near term studies on T-11, DITE, TFR and PLT.

Future machines will go to higher current operation such as TFTR (2.5 MA) and JET (3-5 MA). The TFTR and T-20 (6 MA) will be the first to use a D-T mixture and produce 14 MeV neutrons. However, none of these devices will have used superconducting magnets nor will they demonstrate burn times long enough for refueling or radiation damage to be significant problems. A summary of the past, present and near term fusion devices is given in Appendix A.

b.) Possible Scenario for Future Reactors

Let us now consider the type of reactors that are needed to produce a commercial tokamak reactor. Table 14 lists reactors and some (certainly by no means all) of the problems each must substantially solve during the operation of that device. We have listed an estimate of the plasma current for each machine as a measure of its operating level and a summary of abbreviations used in this discussion is given in Appendix B. Figure 1 is a graphic representation of this scenario.

The primary goal of the scenario is a commercial fusion reactor. A secondary goal can be the development of a fusion-fission reactor system and the analysis suggests the machines required to demonstrate the economic feasibility of this additional objective. It is an interesting result that the reactors suggested are relevant to either objective so that a final decision between these goals, if that is required, need not be made until around 1990.

The facilities have been classified into three broad categories, plasma physics devices, machines which simultaneously extend plasma physics and have a significant technological purpose, such as the development of divertors for impurity control or the advancement of neutral beam technology, and facilities with mainly technological objectives such as a superconducting magnet development program or a tritium facility. An abbreviated list of the plasma experiments operating or scheduled for 1975 and those machines planned for 1975-80 period are the basis for the four large tokamak devices scheduled for the 1980-83 period. These four machines, the U.S. tokamak fusion test reactor (TFTR),

POSSIBLE SCENARIO TO COMMERCIAL TOKAMAK REACTORS

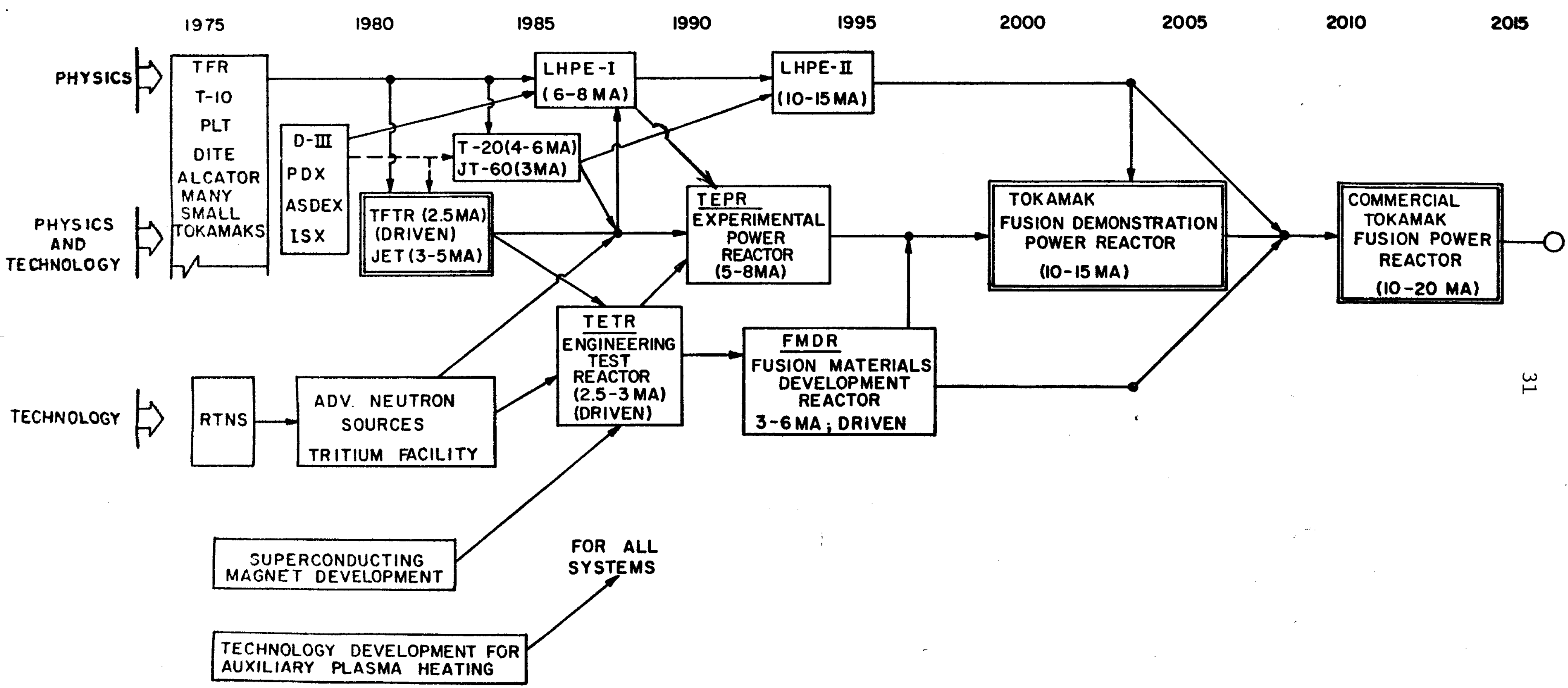

FIGURE 1

Table 14

Function of Tokamak Reactors in Current Scenario to a Commercial UWMAK-Type Reactor

Reactor	Immediate Supporting Devices	Year of Operation	Current MA	Major Purpose
TFTR (Driven) Tokamak Fusion Test Reactor	PLT, T-10, ORMAK ATC, PDX, ISX	1981	2.5	D-T burning T_2 handling Scaling Laws Handling Radioactive Components
TETR (Driven) Tokamak Engineering Test Reactor(12)	D-III, PDX, PLT, TTA, RTNS, TFTR, JET	1985-87	2.4	Test Materials to 10^{21} n/cm^2 (E = 14 MeV) Fueling S/C Magnets Limited T_2 Breeding Neutronics Test Remote Handling Blanket Design Tests Performance Test of Plasma Operation Required for Hybrids
LHPE (Lawson to Ignition) Large Hydrogen Plasma Experiment-I	TFTR	1985-87	6-8	Achieve Classical Lawson Conditions Demonstrate Heating Demonstrate Fueling Startup Achieved Particle Removal (Divertor) Test Physics Scaling Laws at High Current
EPR (Ignition) Experimental Power Reactor	LHPE, TETR, TFTR, JET, T-20, JT-60	1988-92	6	Limited Electrical Generation High Temperature Operation Fabrication of CTR Vessel Components in Field Structural Material Test(Fatigue) Demonstrate Safe Handling and Pumping of Liquid Metals in CTR Environment Reliability of S/C Magnets Remote Assembly and Disassembly BR~1 and Controlled Tritium Release in Nuclear Island
LHPE-II	LHPE-I	1990-94		Achieve Reactor Grade Plasma Plasmas Typical of Full Scale Fusion Reactors. Plasma Density and Temperature Exceed the Requirements for Ignition.

Table 14 (cont.)

Reactor	Immediate Supporting Devices	Year of Operation	Current MA	Major Purpose
FMDR (Driven) Fusion Materials Development Reactor	TETR	1990-1992	3-5	Understand Radiation Damage to $10^{22} n/cm^2$ (14 MeV neutrons) Develop Ductile, Swelling and Fatigue Resistant Materials for DPR Systems Show Reasonable Disposal Techniques for Damage First Walls Complete T_2 Cycle and Containment
DPR-I (Ignition)	TETR, EPR, LHPE-I & II	1997-2005	10-15	First Continuous Production of Elec. Demonstrate Ability to Produce Economically Competitive Power Plant Factor of $\gtrsim$ 70% Show Environmental Acceptability in all Areas Including Waste Disposal, T_2 Release, Scarce Resource Usage and Potential Diversion Threats
CPR-I (Ignition)	EPR, FMDR, LHPE-II, DPR-I	2005-2015	15-20	The Type of Reactor a Utility Would Buy in the Open Market

the Joint European Torus (JET), the Soviet tokamak T-20, and the Japanese tokamak JT-60, then provide the starting point for the remainder of the discussion.

Recent indications are that both JET and TFTR should operate around 1980 but that T-20 and perhaps JT-60 will not begin operation before 1982 or possibly 1983. These experiments are aimed at understanding the physics of burning plasmas at conditions approaching breakeven and/or ignition. These reactors must also address the problem of handling radioactive components and will, to a limited degree, encounter tritium handling problems. Auxiliary devices such as D-III should answer the physics questions related to non-circular cross sections, and DITE, PDX, ASDEX and ISX, should contribute to our understanding of the impurity problem, particle removal, and impurity control.

Non plasma facilities should also be in place by about 1980 to support post-1980 devices. At the very least, there must be solid target neutron production facilities to gather information at fluences up to $10^{20} n/cm^2$ (14 MeV) so that we can safely design the next higher power series of D-T experiments. A superconducting magnet test facility should be constructed in the 1978-82 time period to provide design and fabrication experience to the 1985 machines. (Such facilities are planned for construction as part of the present U.S. program.) Finally, a separate tritium chemistry facility designed to find ways of extracting T_2 in low (~1 appm) concentrations from large amounts of hot lithium systems must be in place. This facility should also address the question of tritium extraction from solid breeders (especially, the effect of sintering and T_2 holdup) while demonstrating methods of rapid handling for T_2 turn around.

From this point, it is possible that one can proceed ahead directly to an experimental power reactor EPR designed with the objective of generating limited net electric power, even if it is only during the plasma burn period. However, such a reactor will have a plasma current of at least 6 MA with a circular cross section and possibly a higher current in the noncircular case. This represents a significant extrapolation, particularly from the TFTR plasma. In addition, the EPR will be extending technology in a significant way through the use of superconducting magnets, reasonably sophisticated blankets, and high voltage neutral beams.

In the scenario developed here, we consider the possibility of constructing two large reactors in the mid-1980's which separately extrapolate plasma physics and reactor technology. The two machines are a Tokamak Engineering Test Reactor[12] (TETR) and a Large Hydrogen Plasma Experiment. The TETR is to extrapolate engineering technology while the LHPE is to extrapolate the plasma physics. Importantly, TETR need not extrapolate our plasma physics knowledge beyond that expected around 1980 while the LHPE need not extrapolate many areas of technology which should be available in 1980. The physics base for TETR are the PDX, D-III, TFTR and JET experiments.

The TETR can be a relatively small machine with essentially the same plasma physics parameters as the TFTR but with a noncircular (2 to 1) cross section. Its purposes are to advance materials testing, blanket design, tritium breeding, and neutronics techniques required for the first Experimental Power Reactors (EPR's). One would hope to test several alloys in a TETR under realistic conditions (temperatures, stresses, plasma radiation, and cyclic application) up to 10^{21} n/cm^2. Such a reactor could also build on the information developed in the Tritium Test Facility (TTF) so that a limited amount of breeding in both solid and liquid systems could be demonstrated. This breeding would verify complex 3-D neutronics calculational procedures that will ultimately be employed as final calculations for designs of blanket assemblies. The TETR would also be the first system to build on superconducting magnet test experience and would provide a realistic test for smaller (4-5 m bore) S/C magnet assemblies. Superconducting coils are necessary on TETR to achieve reasonable burn times and a good duty factor. Burn times of 30 to 60 s also means fueling will be an important element of reactor operation.[12]

The role of the LHPE would to be achieve the classical Lawson criteria, demonstrate viable heating and fueling techniques for a tokamak reactor and successfully solve the startup problem. Energy storage, switching and recovery could be part of that system. This system could also use the large S/C magnets provided by the superconducting magnet development program. The LHPE provides a device for studying the physics of large high temperature plasmas and allows one to develop the detailed knowledge required for operation of the EPR and follow on machines.

Importantly, the cost of both the LHPE and TETR together should be less than or about equal to the cost of a circular EPR at 6 MA.. The reason is TETR is probably only half the major radius and half the current of an EPR and the LHPE does not have the added complexities brought on by D-T operation.

Both the TETR and LHPE would provide the basis to build the first EPR in about the 1990 period. This device would be the first large scale reactor which might require fabrication of realistic reactor materials in the field (TETR is half the size of EPR or less and the LHPE could use materials not necessarily suited for D-T environments.) The EPR would not necessarily be a materials radiation damage test reactor but it would provide the first complete high temperature test of reactor materials in a moderately neutron damaging environment. Pumping of hot liquid metals in high magnetic fields could be demonstrated as well as showing that such large masses of liquid metals can be safely handled in a CTR environment. (One might choose to dump the heat generated in the TETR and use well established H_2O cooling techniques).

The radioactivity levels will be quite high in an EPR and remote handling techniques for assembly and disassembly would be required. This reactor should also breed significant amounts of T_2 and demonstrate that amounts up to several kg's can be safely handled. The release of T_2 to the environment must be carefully studied in this machine.

Finally, the physics base from JET, TFTR, T-20, JT-60, LHPE, and TETR should allow long pulse operation in EPR, a reasonable duty factor, and allow the device to generate net electrical power. However, the device need not have a high plant factor. The intention here is more than one of overall proof in principle of the engineering feasibility rather than economics.

The operation of TETR in the 1980's should provide the necessary information to build the major materials test facility for the scenario illustrated on Fig. 1 and we will refer to that reactor as the Fusion Materials Demonstration Reactor (FMDR). The function of this reactor will be to develop ductile, low swelling and fatigue resistant structural materials that can operate in fluences of 10^{22} n/cm^2 (14 MeV). It must also simulate realistic fusion environments so that the design of the Demonstration Power Reactor can proceed in a confident manner. The FMTR would not be a net energy producer because of the necessity for many test loops and even low temperature operation of some of the more permanent structural components. It may also be a consumer of tritium because of the very specialized nature of its test loops although some limited high temperature breeding could take place to test breeding compounds and the effect of high tritium concentration on materials performance. A considerable amount of information on high temperature operation can also be obtained in test loops at high neutron fluxes. The FMTR would also be expected to test materials up to 3×10^{22} n/cm^2 (14 MwV) for the Commercial Power Reactor of the 1990's.

The operation of TETR in the 1980's in the driven mode described in ref. 12 also provides the basic test facility required to build a demonstration fission-fusion hybrid system. The two component mode of high powered deuterium beam injection into a tritium target is optimum for neutron production[13] and therefore optimum for TETR and a hybrid. Thus, a TETR can also be viewed as the plasma feasibility test for the hybrid in the sense that it can demonstrate long burn times (30-60s) and reasonable duty factors (75-85%) required for a hybrid. As such, a hybrid demonstration reactor prototypical of a commercial system could be constructed in the early 1990's and this option is illustrated on Fig. 2. Since the tokamak demonstration hybrid reactor (TDHR) is driven and achieves high neutron fluxes, this facility might also be used as the fusion materials demonstration reactor. However, the FMDR deals with such a critical problem in fusion technology that we do not advocate combining its purpose with that of the TDHR. The TDHR must demonstrate that fission-fusion systems can perform their function, whether that function is electricity production, fissile fuel production, or both, in an economically competitive manner. It must also demonstrate environmental acceptability of both the fusion related aspects, such as tritium handling, tritium management, and long term solid radioactivity waste disposal, and the more familiar aspects of fission systems.

If the fission-fusion option has shown itself to be viable, the TDHR in the early 1990's would likewise provide the requisite base for the construction of the first commercial hybrid systems by around the year 2000. It seems clear that tokamak hybrid reactors, requiring less demanding plasma performance, can

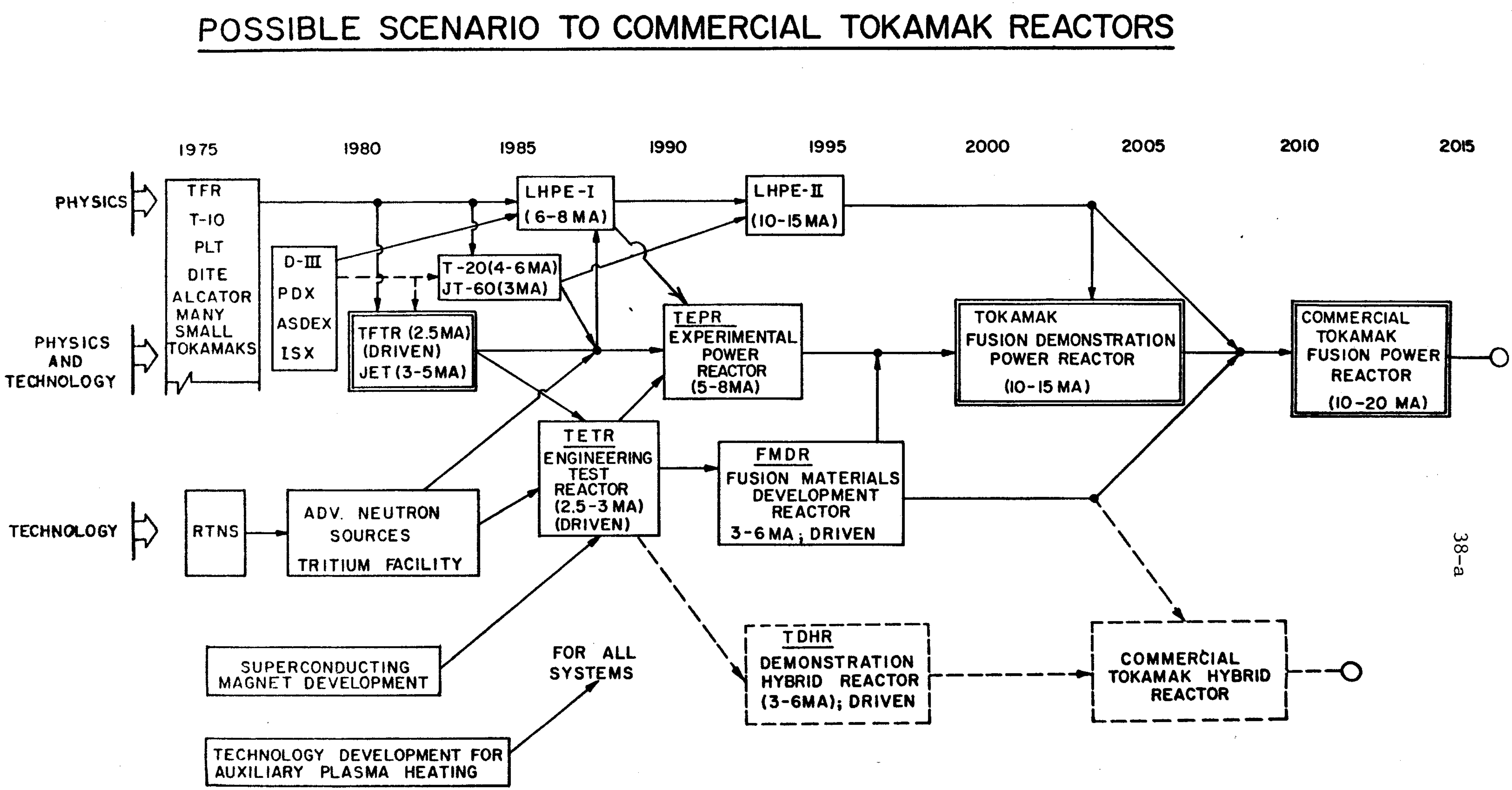

FIGURE 2

The original scenario with the hybrid reactor development option indicated by the dashed line. If this option is chosen, the time frame for development of pure fusion might be extended. This has not been examined.

prove the physics basis early and can achieve commercialization more rapidly _if_ the technical complexity of adding fission blankets does not extend the time necessary to develop a reliable and economically competitive reactor system.

Returning to the basic pure fusion scenario on Fig. 1, a second large plasma experiment in H or D is required following LHPE-I to improve the plasma physics base for the pure fusion DPR and CPR. This device would probably have a plasma current in the 10-15 MA range and extensive auxiliary heating to reach the density-temperature regime of full scale reactor plasmas. With auxiliary heating, this should provide all the physics scaling laws required for DPR's and CPR's. Alpha heating can be simulated with auxiliary beam heating and will also be studied in the TETR and EPR. As with LHPE-I, operating in H or D will significantly reduce costs by eliminating the complexities and auxiliary facilities required to handle D-T operation.

The EPR, TETR and LHPE-I provide the base for the fusion demonstration power reactor. This is the last reactor before a commercial plant and must be a net energy producer. It must _demonstrate_ that it can produce electricity continuously to the grid and competitively with other advanced sources of central station electric power, including environmentally acceptable coal plants. It must demonstrate plant factors of approximately 70% and it must be environmentally acceptable in all ways. This means that methods of solving the scarce elements requirements, tritium diversion, long term radioactive waste disposal and in-service routine release of radioactivity must all be in hand. The maintenance record must be reasonable and no major technological problems must stand in the way of constructing a truly _Commercial Power Reactor_ (CPR) in the early 21st century.

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Appendix A

Abbreviations and Names for Tokamak Experiments

Device	Name	Plasma Current (MA)	Laboratory and Country
T-3	Tokamak-3	0.07	Kurchatov (U.S.S.R.)
T-4	Tokamak-4	0.12	Kurchatov (U.S.S.R.)
T-10	Tokamak-10	0.8	Kurchatov (U.S.S.R.)
T-20	Tokamak-20	6	Kurchatov (U.S.S.R.)
ST	Symmetric Tokamak	0.07	Princeton Plasma Physics Lab (U.S.A.)
ATC	Toroidal Compressor	0.06-0.14	Princeton Plasma Physics Lab (U.S.A.)
PLT	Princeton Large Torus	1.4	Princeton Plasma Physics Lab (U.S.A.)
PDX	Poloidal Divertor Experiment	0.5	Princeton Plasma Physics Lab (U.S.A.)
TFTR	Tokamak Fusion Test Reactor	2.5	Princeton Plasma Physics Lab (U.S.A.)
D-IIA	Doublet-II-A	0.13	General Atomic Co. (U.S.A.)
D-III	Doublet-III	5.0	General Atomic Co. (U.S.A.)
ORMAK	Oak Ridge Tokamak	0.12	Oak Ridge National Lab (U.S.A.)
JFT-2		0.1	Tokai (Japan)
JT-60	Japan Tokamak	3.3	Tokai (Japan)
TFR	Tokamak Fontenay-aux-Roses	0.4	Fontenay-aux-Roses (France)
DITE	Divertor-Injected-Tokamak-Exp't	0.28	Culham (U.K.)
JET	Joint European Torus	3.6	Not determined

Appendix B

Letter Designations for Tokamak Test Reactors and Tokamak Power Reactors

TFTR	Tokamak Fusion Test Reactor
TETR	Tokamak Engineering Test Reactor
LHPE	Large Hydrogen Plasma Experiment
EPR	Experimental Power Reactor
FMDR	Fusion Materials Demonstration Reactor
DPR	Demonstration Power Reactor
CPR	Commercial Power Reactor
UWMAK	University of Wisconsin Tokamak (Conceptual Reactor Design)