CONFERENCES AND SYMPOSIA

FUSION REACTOR DESIGN IV

Report on the
Fourth IAEA Technical Committee Meeting
and Workshop,
Yalta, USSR,
26 May – 6 June 1986

M.A. ABDOU
University of California, USA

E. BERTOLINI
JET Team, Abingdon, UK

R. HANCOX
UK Atomic Energy Authority, UK

W.J. HOGAN
Lawrence Livermore National Laboratory, USA

A.I. KOSTENKO
D.V. Efremov Scientific Research Institute
of Electrophysical Apparatus, USSR

G.L. KULCINSKI
University of Wisconsin, USA

R.L. MILLER
Los Alamos National Laboratory, USA

V.V. ORLOV
I.V. Kurchatov Institute of Atomic Energy, USSR

F. RAU
Max-Planck-Institut für Plasmaphysik,
Fed. Rep. of Germany

I.N. SVIATOSLAVSKY
University of Wisconsin, USA

J. KUPITZ
International Atomic Energy Agency, Vienna

ABSTRACT. The International Atomic Energy Agency convened, in the framework of its Fusion Technology and Engineering Programme, the 4th Technical Committee and Workshop on Fusion Reactor Design and Technology at Yalta, USSR, from 26 May – 6 June 1986. This report contains all summaries of sessions that were organized during the workshop. The papers presented at the meeting are being published by the Agency in its Proceedings Series.

INTRODUCTION

(J. Kupitz)

The long term objective of fusion power reactor development is to bring to the world a new source of unlimited energy. While essential progress has been made during the last few years there is still a great deal of research and development needed in areas such as plasma physics, nuclear technology, confinement systems, materials, fuel cycle, radiological safety. The IAEA convened the fourth TCM and Workshop on Fusion Reactor Design and Technology in Yalta in order to

- review and assess the current status and recent progress made in fusion reactor experiments, design and technology
- identify the areas in which further work needs to be done to advance towards the goal of a commercial fusion power reactor
- identify the critical issues in fusion reactor design and technology that will be important during the next five years.

The first IAEA meeting in this series [1] was held in January 1974, at the Culham Laboratory, UK. The second IAEA Technical Committe Meeting and Workshop on Fusion Reactor Design [2] was held in Madison, Wisconsin, USA, in October 1977, and the last meeting [3] was held in Tokyo, Japan, in 1981. These meetings were extremely fruitful in assessing the state of the art of fusion reactor design and also in indicating the appropriate orientation of R and D needs for the future [4].

During the Yalta meeting, 80 participants from ten countries or international organizations participated and presented a total of 70 papers. The meeting was – as it was in the past – subdivided into two parts. In the first week, papers were presented on experimental results from existing machines, on the status of major fusion reactor projects, reactor design studies, plasma engineering aspects, nuclear and energy technologies, materials development, and on safety aspects. In the second week, workshop sessions were convened by groups on specific aspects of fusion reactor development. The purpose of these sessions was to discuss the major changes and advances that have taken place during the past five years and to identify critical issues in reactor design and technology during the next five years. Each session formulated recommendations for future activities. The following workshops were held:

(1) Near term tokamaks, chaired by G.L. Kulcinski and E. Bertolini
(2) Long term tokamak reactors, chaired by R. Hancox
(3) Stellarators, chaired by I.N. Sviatoslavsky and F. Rau
(4) Magnet systems, chaired by A.I. Kostenko
(5) Inertial confinement fusion, chaired by W.J. Hogan
(6) Hybrid fusion fission reactors, chaired by V.V. Orlov
(7) Alternative fusion concepts, chaired by R.L. Miller
(8) Fusion nuclear technologies and materials, chaired by M.A. Abdou

After the meeting a scientific tour was offered to all participants through laboratories in Moscow and Leningrad engaged in the development of nuclear fusion.

REFERENCES

[1] Fusion Reactor Design Problems (Proc. IAEA Workshop Culham, 1974), Nucl. Fusion Special Suppl. 1974, IAEA, Vienna (1974).
[2] Fusion Reactor Design Concepts (Proc. 2nd IAEA Workshop Madison, 1977), IAEA, Vienna (1978); also CONN, R.W., FRANK, T.G., HANCOX, R., KULCINSKI, G.L., SCHMITTER, K.H., STACEY, W.M., Jr., Nucl. Fusion **18** (1978) 985.
[3] Fusion Reactor Design III (Proc. 3rd IAEA Workshop, Tokyo, 1981), IAEA, Vienna (1983); also ISO, Y., STACEY, W.M., Jr., KULCINSKI, G.L., KRAKOWSKI, R.A., CARLSON, G.A., YAMANAKA, C., CASINI, G., IGATA, N., Nucl. Fusion **22** (1982) 671.
[4] For a survey of present R&D, see World Survey of Activities in Controlled Fusion Research, Nuclear Fusion Special Supplement 1986, IAEA, Vienna (1986).

1. NEAR TERM TOKAMAK REACTORS

(G.L. Kulcinski, E. Bertolini)

1.1. Introduction

The period between the Tokyo (1981) and the Yalta (1986) Conferences on the same topic has been filled with a substantial design effort for the next major technology step in tokamaks. At the Culham Meeting (1974), there were no near term tokamak concepts discussed but at the Madison Meeting (1977) no less than a dozen near term reactors were discussed. The number of concepts was reduced by the time of the Tokyo Meeting to four: INTOR, NET, FER, and FED. At the Yalta Conference, the number of major near term tokamak reactors has remained at four (OTR, NET, FER, TIBER-II), with an additional major effort in the USA on the Compact Ignition Tokamak (CIT). The CIT can be more readily classified as a physics device as can IGNITOR, whose conceptual design is currently underway, but since neither design was presented in detail in Yalta, only brief references will be made to them when appropriate.

Concerning INTOR, it has continued to play its key role of focusing international co-operation on a near term tokamak reactor, allowing the four main national projects to exchange views and improve designs.

The level of effort and resources committed to the Near Term Tokamaks has greatly increased in the past four years, and it was determined that more than 200 full time person years are now being devoted to this subject worldwide.

Because of the cost and magnitude of such a next step, there has been considerable discussion about international collaboration on these devices, at least in sharing information on the design solutions. Such a maturization of this phase of fusion research reflects the growing commitment to fusion by the nations of the world.

1.2. Activities in Near Term Tokamak Reactor design since 1981

Since the Tokyo Conference, the FED study in the USA was discontinued and replaced with a less ambitious device called TFCX. This study was also discontinued in 1985 and replaced with an even more near term device, the CIT physics test facility. However, a more aggressive tokamak technology test facility called TIBER was also initiated in 1985 in the USA. The USSR has begun a major design study in 1983 of the OTR hybrid tokamak and continues to develop that concept. The Japanese discontinued the R project in 1985 and now concentrate on the FER. The NET (a European Community project) design effort has been more continuous, and it has proceeded along the same general guidelines that were originally proposed in 1978 and expanded with serious design activities in 1983.

Device	Design origin	Date of major design activities	Number of full time equivalent scientists
INTOR	EC, Japan, USA, USSR	1979	50[a]
FER	Japan	1980	50
NET	EC	1983	80
OTR	USSR	1983	60
TIBER-II	USA	1985	10

[a] In addition to those working on national near term tokamak designs.

1.3. Status of tokamak physics experiments

The highlights of experimental research in fusion power since 1981 have been the successful completion and the start of operation of the TFTR in the USA (25th December 1982), JET in the European Community (25th June 1983) and JT-60 in Japan (8th April 1984). These devices are the so called 'Large Tokamaks', because their parameters (Table 1-I) are approaching those required to demonstrate reactor relevant plasma performance. In the USSR, T-15, in an advanced status of construction, is due for operation some time in 1988.

With the experimental programmes in their early stages, the results obtained have already substantially extended the plasma parameter range existing in 1981 (Table 1-I). Thus far, peak ion temperatures have been measured to be ~8 keV in JET, global energy confinement times $\tau_E \approx 0.85$ s have been achieved with Ohmic heating in JET, $n_e\tau_E$ values of 10^{20} $m^{-3}\cdot s$ with pellet injection in TFTR, and fusion product values of $n_i\tau_E T_i$ in excess of 10^{20} $m^{-3}\cdot s\cdot keV$ have been measured with pellet injection and 10 MW of neutral beams also in TFTR.

The construction of these facilities has required a considerable development in technologies, some of them clearly relevant to the next generation of 'reactor-like' experiments. Examples are the multimegawatt, long pulse, neutral beam lines, ICRF generator antenna and LHRF generator launcher systems, the multipellet injection equipment, the safe control of elongated plasmas and magnetic limiter configuration, the high and low 'Z' materials, development of diagnostics compatible with moderate DT neutron activation, development of remote handling techniques, etc.

While TFTR plans a limited number of D + T pulses, JET has been conceived, designed and constructed for a sustained D-T operation of several thousand pulses. This has required a detailed neutron activation analysis and remote handling equipment development, leading for the first time to proven tokamak assembly details and maintenance tools in fusion research.

The size of these experiments has required an involvement of industry to such a degree that it may be considered a meaningful exercise for the next step. It has also become apparent that the design concepts, technical solutions adopted and selected technologies (not simply overall dimensions) may play a leading role in determining costs.

The main goal is now to increase, with these experiments, the fusion product $n\tau T$ by a factor of ten or more. The key issue is to find ways to counteract the degradation of the energy confinement time with plasma heating. Heating experiments in ASDEX and PDX divertor have suggested the existence of high modes of operation with poloidal divertor configurations. JET, TFTR and JT-60 are planning to increase the engineering capability of the tokamak system (plasma current and/or magnetic field) substantially above the design values, while JET is also planning magnetic limiter configuration experiments up to 4 MA of plasma current. In Europe, the construction of a new generation of medium size tokamaks (TORE Supra, ASDEX Upgrade, FTU Upgrade and COMPASS) is being pursued to address specific physics issues and these devices will be completed between 1987 and 1989. It is worth mentioning that T-15 and TORE Supra should provide some data base for the next step superconducting coil technology.

1.4. Progress since 1981

The working group analysed twelve areas with respect to A) Trends and B) Advances made since 1981. The trends were considered to be new ideas or changes in the direction of research since 1981. The advances include mainly experimental, but to some extent also theoretical, data obtained since 1981. The results are summarized below.

Significant change	Heating theory and facilities Startup procedures and current drive Mechanical configuration Maintainability
Some change	Magnets Blanket/shields Diagnostics
Not much change	Physics Impurity control Fuelling First wall/divertor/limiter plates Materials

A brief description of the trends and advances is shown in Table 1-II without detailed documentation.

TABLE 1-I. CURRENT DATA BASE FOR LARGE TOKAMAKS

Parameter/data	TFTR	JET	JT-60	T-15
Major radius (m)	2.5	2.96	3.1	2.43
Minor radius (m)	0.85	1.25	0.95	0.7
Elongation	1	1.78	1	1
Plasma volume (m^3)	35	150	60	25
Toroidal field (T)	5.2	3.45	4.5	4.5
Type	Cu	Cu	Cu	Superconducting
Plasma current (MA)	2.5	4.8	2.7	2.0
Heating	NI	NI ICRF	NI LHRF ICRF	NI ECRF
Experimental data (June 1986)				
Electron density (m^{-3})	4×10^{20}	7×10^{19}	5.7×10^{19}	–
Maximum ion temperature $\hat{T}_i$ (keV)	10	8	1	–
Confinement time (s)	0.5	0.85	0.4	–
Impurity (Z_{eff})	1.2	2	1.2	–
D–D neutrons (s^{-1})	4.5×10^{15}	5×10^{15}	–	–
$\hat{n}_i \hat{T}_i \langle \tau_E \rangle$ ($m^{-3} \cdot keV \cdot s$)	1.9×10^{20}	1×10^{20}	1.8×10^{19}	

TABLE 1-II. PROGRESS SINCE 1981

Physics:

1. Trends

   Assumptions on confinement scaling law and β-limits
   Provide flexibility ('margin') to accommodate uncertainties

2. Advances

   New experimental data on β-limits and τ_E; theoretical and computational results

Impurity control:

1. Trends

   Poloidal divertor (single and double null) and pumped limiter

2. Advances

   New experimental data on poloidal divertor
   Experimental data on pumped limiter
   Two-dimensional physical computational models
   Experimental validation of the computational models
   Results of physics–nuclear–engineering analyses of divertor plates and pumped limiter

Heating:

1. Trends

   ICRH and other RF methods
   Neutral beams

2. Advances

   Experimental data with high power ICRH
   Physical modelling of ICRH for reactor
   Multi-megawatt antenna construction studies
   Experimental data with high power ECRH
   ECRH design studies for special purposes
   Multi-megawatt NBI line development up to 160 kV
   Development of negative ion sources

Fuelling:

1. Trends

   Pellet and gas puffing

2. Advances

   New experimental data with pellet injection
   Pellet injector development

First wall/divertor plates/limiter:

1. Trends

   Austenitic stainless steel, H_2O cooled for first wall;
   W armour of H_2O cooled CV heat sinks for divertor plates;
   low Z materials for limiter and walls

2. Advances

   Lifetime analysis models with disruptions
   Study results of first wall and divertor plate concepts with disruptions

Startup and current drive:

1. Trends

   Startup scenario
   Current drive for steady state operation and for current density profile control

2. Advances

   Experimental data with non-inductive and hybrid scenarios
   Physical modelling of hybrid scenario for reactor
   Lower hybrid current drive experiments
   Consideration of ICR/LHR/ECR/NBI and synchrotron power for current drive in reactor

Blanket/shield:

1. Trends

   H_2O/He coolant
   Be multiplier for solid breeder
   Ferritic stainless steel structure
   Outboard breeding blanket, test module

2. Advances

   Expanded data base for solid breeder (T-recovery, damage, . . .).
   International joint experiments under way and to be started for solid breeder (IEA (BEATRIX), US/JAPAN,. . .)
   Expanded data base for SS (HT9 . . .) (HFTR/ORR) (30–50 dpa), . . .)

Mechanical configuration:

1. Trends

   Variety of removable component structures using radial straight motion or oblique motion of multisegmented removable sector, . . .
   Common cryostat for all SC coil
   Semi-permanent shield structure

2. Advances

   Simplification of maintenance
   System integration design
   Development of oblique replacement concept
   Improvements of calculation for plasma position control with passive structures and active coil, and of their integral configuration design

Maintainability by remote handling:

1. Trends

   Studies for improvement in mechanical configuration design
   Replacement of damaged internal components (FW, divertor, . . .) instead of in situ repair/maintenance
   Accessibility to the torus after reactor shutdown
   Limited hands-on capability

2. Advances

   Development in JET
   Development in other areas, such as fission reactor, accelerator, . . .

Magnets:

1. Trends

   Reduction of PF coil requirements by assisted startup
   Nb_3Sn conductor for PF coils
   Forced flow cooling
   Extension of radiation damage limits in existing superconductors

2. Advances

   Expanded data base from LCT, 12 T programmes, SUPRA-II, TRIAM-1M, T-15, SULTAN
   Design and construction of pulse coil (~30 MJ)

Materials:

1. Trends

   SS structural
   H_2O coolant

2. Advances

   (See materials section)

Diagnostics:

1. Trends

   Diagnostics for plasma control and all systems of reactor suitable for operation in reactor conditions

2. Advances

   Identification and requirements to diagnostics
   Construction of diagnostics compatible with moderate neutron activation and remote handling capability (JET)

1.5. A comparison of the five international Near Term Tokamak Reactor (NTTR) studies

The five main NTTR studies currently being conducted are summarized below and in Tables 1-III and 1-IV. These summaries are current up through June of 1986 but the reader should realize that the designs will evolve as time progresses.

TABLE 1-III. THE INTERNATIONAL NTTR STUDIES – GENERAL PARAMETERS

	OTR[i] (USSR)	FER (Japan)	TIBER-II[i] (USA)	NET[c] (EC)	INTOR[p] (IAEA)
Fusion power (MW)	520	297	290	600(900)	570
Major radius (m)	6.2	5.2	3.0	5.18(5.4)	4.9
Q	$10 \rightarrow \infty$[k]	∞	6.2[j]	∞	∞
External heating power[a] (MW)	50[f]/$\leqslant$30[g]	50[m]	46.8[d]	50[e]	40[e]
Burn pulse length (s)/duty cycle	600/0.90	2000/0.87	steady state	600/0.9 (400/0.9)	200/0.7
Peak/average neutron wall load ($MW\cdot m^{-2}$)	1.2/0.8	0.88/0.68	2.0/1.3	1.0[b] (1.4[b])	1.3[b]
End of life fluence goal ($MW\cdot a\cdot m^{-2}$)	5	0.3	3.0	0.8[n]	0.3–3
Final phase availability goal (%)	60–70	–[h]	30	25	25
Final phase tritium consumption ($kg\cdot a^{-1}$)	18	–[h]	4.8	7.7	6.1
Final phase tritium breeding ratio	~1.05	0	–[h]	~0.4	>0.6

[a] Absorbed by plasma.

[b] Average only.

[c] Plasma phase-I parameters shown in parentheses.

[d] For current drive.

[e] For heating to ignition.

[f] During startup.

[g] During burn phase.

[h] To be determined.

[i] Preliminary parameters.

[j] Approximately 47 MW of current drive power is dissipated during the steady state burn.

[k] Ignition margin not determined at this time.

[m] 50 MW of RF for heating to ignition; 10 MW of LH for current maintenance during transformer recharge.

[n] 0.8 $MW\cdot a\cdot m^{-2}$ is the fluence goal; machine will be designed for a limit of 3 $MW\cdot a\cdot m^{-2}$.

[p] INTOR phase 2A, Part II (1985).

1.5.1. TIBER-II (USA)

1.5.1.1. Mission statement and objectives

A compact, non-inductive, current driven, steady state ETR for the qualification and testing of reactor relevant physics and components in a reactor relevant configuration in order to meet the basic performance requirements of the next step. This includes:

- (i) testing of components
- (ii) testing of materials (to moderate fluences of ~3 $MW\cdot a\cdot m^{-2}$)
- (iii) testing of tritium and energy extraction under reactor conditions
- (iv) demonstration of (remote) maintainability
- (v) demonstration of plant reliability/availability
- (vi) demonstration of safe, reliable and environmentally acceptable operation.

Current US studies are conducted with the philosophy that the ETR engineering and construction project will be an international effort.

1.5.1.2. Physics basis

Operating mode: steady state, current driven
Confinement scaling: Kaye-Goldston with H mode enhancement of 1.3 for divertor operation
Beta scaling: $\langle\beta_{crit}\rangle = 4\ I/aB$
$\langle\beta_{max}\rangle = 0.75\ \langle\beta_{crit}\rangle$
Density limit: Either $\langle\beta_{max}\rangle$ limit at T_i or Greenwald limit, where $\langle n_{e,max}\rangle \approx 0.75\ \kappa\langle J\rangle$
Current drive: ECRH + LH (neutral beams considered as a backup)
Disruption and profile control: Multifrequency ECRH ($d\omega/\omega \sim 20\%$) used to control J(r) and stabilize sawteeth at q = 1 surface and disruptions at q = 2 surface.

TABLE 1-IV. THE INTERNATIONAL NTTR STUDIES – PHYSICS PARAMETERS

	OTR[n] (USSR)	FER (Japan)	TIBER-II[n] (USA)	NET[h] (EC)	INTOR[p] (IAEA)
Major radius R_0 (m)	6.2	5.2	3.0	5.18 (5.4)	4.9
Minor radius a (m)	1.5	1.12	0.834[a]	1.35 (1.7)[b]	1.2
Elongation κ	1.5	1.5	2.18[a]	2.18 (2.18)[b]	1.6
Triangularity δ	0.3	–[s]	0.55[a]	0.65 (0.65)[b]	–[s]
B_0/B_{max} (T)	5.6/11	5.3/12	6/14	5/11.4 (4.8/11.4)	5.5/11
Plasma current (MA)	8.2	5.9	10.0	10.8 (15)	8.0
Beta scaling factor[j] g	3.1	4.0	4.0	3.5	4.0
$\langle\beta\rangle$	0.03	0.04	0.06	0.056(0.065)	0.049
$\langle T_i\rangle$ (keV)	7.5	10	16.7[g]	10	10
$\langle n_e\rangle$ (10^{20} m^{-3})	1.7	1.43	1.22	1.56	1.4
Murakami parameter[m] M (10^{19} m$^{-2}\cdot$T^{-1})	16	14	N/A[c]	16	see (q)
Energy confinement scaling	T-11 (T-10)	H mode[l]	Kaye–Goldston[d]	ASDEX H mode	see (r)
Ignition margin at $\langle T_i\rangle$	1.1	≥1	N/A[k]	2.9 (4.3)	≥1
q_{MHD}	2.1[f]	2.3[i]	3.9[i]	2.1[f](2.2[f])	1.9[f]
Impurity control	Single null poloidal divertor	Single null poloidal divertor	Double null poloidal divertor	Double null poloidal divertor	Single null poloidal divertor
Toroidal field ripple at plasma edge (%)	1.0	0.75	0.94	1.5 (2.8)	1.2

a Shape at 95% flux surface.

b Shape at null point.

c Uses Greenwald limit, where $\langle n_{max}\rangle = 0.75\ \kappa\langle J\rangle$.

d With H mode enhancement of 1.3 for divertor operation.

f Equivalent cylindrical safety factor.

g Electron temperature for TIBER-II is 23.5 keV.

h Plasma phase-I parameters shown in parentheses.

i q_{edge}

j Troyon coefficient g, where $\langle\beta_{crit}\rangle = g\ I/a\ B$.
For TIBER-II, $\langle\beta_{max}\rangle$ is set at 0.75 of $\langle\beta_{crit}\rangle$.

k TIBER-II is current driven and, therefore, runs in a driven mode with ~48 MW of dissipated RF power; it operates under a controlled ignition burn condition with $Q \approx 6.2$.

l Assumed to be ASDEX H mode; $\tau_E = 2$ s.

m $M = \langle n_e(10^{19}\ m^{-3})\rangle\ R_0(m)/B_0(T)$.

n Preliminary parameters.

p INTOR Phase 2A, Part II (1985).

q Not quoted; INTOR does suggest that current experimental limits of 10 to 16 are too stringent for reactors by a factor of 1.5 to 2.

r INTOR requires $\tau_E \approx 1.4$ s for ignition. Kaye–Goldston scaling with no H mode enhancement would yield ~0.7 s; ASDEX H mode would yield ~4 s.

s Not defined.

1.5.1.3. Phases of operation

Phase	I	II	III
Duration (calendar years)	1	2	10
Emphasis	Hydrogen checkout	DT operation verification	Engineering testing
Availability (%)	15	10	30
Annual tritium consumption (kg/yr)	–	1.6	4.8
End of life peak 14 MeV fluence ($MW\cdot a\cdot m^{-2}$)	–	~0.4	~6.4[a]

[a] Goal is ≥3 $MW\cdot a\cdot m^{-2}$.

1.5.1.4. Tritium supply philosophy

The only assured commercial supply of tritium is from Ontario Hydro of Canada. Depending on their commercial sales to other non-fusion users, TIBER-II can burn tritium at the rate of ~2.2–3.7 $kg\cdot a^{-1}$ beginning in 1999, in order to reduce the Canadian inventory to zero at the end of a ten years' operating life. At current commercial prices (~10 M$ per kg), operating costs would be tens of M$ per year. To offset some or all of this cost, we may wish to provide several low technology, low temperature baseload tritium breeding blankets in the machine. We do not rely on any tritium availability from the reactor relevant test blankets in the machine. Tritium supply philosophy is currently under discussion for TIBER-II.

1.5.1.5. Testing philosophy

For main objectives, see 1.5.1.1. above. Detailed testing philosophy is still under study.

1.5.1.6. Plasma heating

See heating philosophy under physics basis above. Current drive powers: LH ≈ 9.5 MW (edge absorbed); ECRH ≈ 37.3 MW (19 MW edge + 18.3 MW core).

1.5.2. NET (EC)

1.5.2.1. Mission statement and objectives

To demonstrate the feasibility of fusion energy production in a plant which integrates the essential technologies of a reactor, as the *only* intermediate step between JET and the DEMO. Specific mission objectives are:

(i) controlled ignition
(ii) extended (up to 1000 s) and reproducible burn pulses
(iii) selection and qualification of design concepts which also meet the basic performance requirements of DEMO
(iv) testing of materials and components in an integrated fusion reactor environment
(v) testing of tritium and energy extraction under reactor relevant conditions
(vi) demonstration of maintainability of a fusion reactor
(vii) demonstration of plant reliability at levels relevant to DEMO
(viii) demonstration of the safe and environmentally acceptable operation of a fusion reactor-like plant.

1.5.2.2. Physics basis

Operating mode: inductive drive, pulsed; burn pulse length ~600 s

Confinement scaling: $\tau_E(s) \approx 0.1\, I(MA)\, R_0(m)$ (ASDEX H mode).

Beta scaling: $\langle\beta_{crit}\rangle = 3.5\, I/aB$
$\langle\beta_{max}\rangle = \langle\beta_{crit}\rangle$

Density limit: from beta limit at T_i = 10 keV or Murakami limit where $\langle n_{e,max}\rangle = 16\, B/R_0$

1.5.2.3. Phases of operation

Staged operation, from a maximum of physics capability and flexibility (plasma phase I to optimize plasma performance) to a maximum of NET's technological potential (phases III and IV).

The plasma phase I is an initial stage involving only the basic machine and the plasma facing components, with an ignition margin 50% higher than in the reference engineering configuration.

Phases	Plasma I	II	III	IV (optional)
Duration (calendar years)	2	4	7	to be defined on the basis of machine reliability
Fluence[a] ($MW\cdot a\cdot m^{-2}$)	–	0.1	0.7	(~2)
Integral operation time (days)	–	70	300	(~700)
Number of pulses ($\times 10^4$)	1	4	5	(~10)
Breeding ratio required	–	–	0.3–0.4	(~0.8)

[a] Based on an average wall loading of 1 $MW\cdot m^{-2}$.

1.5.2.4. Tritium supply philosophy

The tritium supply will be from external sources during phases I and II. During phase III (and IV) in situ breeding with a TBR of 0.3–0.4 (0.8) will be required. In particular, a total of ~6 kg from blanket and ~20 kg from external sources are needed for phases I, II, and III.

1.5.2.5. Testing philosophy

During phases I, II and III, the basic machine and the auxiliary equipment will have been subject to 10^5 operating cycles during 13 calendar years. Extensive experience and reliability data will be obtained during prolonged operation (up to one year) at 25% average availability.

Materials samples and full blanket segments will have been irradiated to 10–15 DPA in a fusion environment. At least two blanket concepts will be tested in parallel.

Typical continuous operation to characterize a blanket concept at one operating condition: 10 days.

Integral operating time for full characterization of a blanket concept: 3 months (~0.2 $MW\cdot a\cdot m^{-2}$).

Blanket, first wall and associated components are removed from the top using an oblique withdrawal method.

1.5.2.6. Plasma heating

50 MW of RF power are provided for heating to ignition.

1.5.3. FER (JAPAN)

1.5.3.1. Mission statement and objectives

(i) To demonstrate a self-ignited, long burn plasma;
(ii) to demonstrate the engineering feasibility of a fusion reactor;
(iii) to demonstrate essential reactor relevant technologies;
(iv) to take a reasonable step from present experiments and technologies and to be flexibly designed to permit staged progress to reach objectives.

1.5.3.2. Physics basis

Operating mode:	pulsed, quasi-steady state (200 s burn) inductive drive (non-inductive current drive during startup and recharge of transformer)
Confinement scaling:	H-mode (assumed ASDEX H-mode); $\tau_E \approx 2$ s
Beta scaling:	$\langle\beta_{crit}\rangle = 4$ I/aB $\langle\beta_{max}\rangle = \langle\beta_{crit}\rangle$
Density limit:	Murakami limit, where $\langle n_{max}\rangle = 14$ B/R_0
Safety factor:	$q_I = 1.8$ ($q_{edge} = 2.3$)
Impurity control:	poloidal divertor, single null
TF ripple:	<0.75%

1.5.3.3. Phases of operation

The objectives of FER are expected to be achieved under staged operation. This comprises the first stage, which will be devoted to plasma performance tests, and the second stage to engineering tests such as breeding blanket test. The details are under investigation.

1.5.3.4. Tritium supply philosophy

For an end-of-life fluence of 0.3 MW a·m^{-2}, and a fusion power of 300 MW, the total tritium consumption is ~5 kg. The tritium supply will be available from external sources; no in situ breeding is planned for tritium supply purposes.

1.5.3.5. Testing philosophy

The lifetime fluence of 0.3 MW·a·m^{-2} is determined as the minimum reasonable level for engineering testing of components; this would exclude irradiation effects on structural materials.

1.5.3.6. Plasma heating

Fifty MW of RF heating (ICRF or LHRF) is the baseline option; high energy NBI is a backup option. LH (~10 MW) is used for current maintenance during transformer recharge. ECRH is used for startup assist.

1.5.4. OTR (USSR)

1.5.4.1. Mission statement and objectives

(i) To demonstrate safe and reliable electricity generation and nuclear fuel production by a *hybrid* fusion test reactor with complete tritium self-supply in the final phase;
(ii) to acquire experience in the design, construction and maintenance under operating conditions close to those in a commercial reactor;
(iii) to serve as a basis for scientific and engineering research in fusion reactors;
(iv) to test materials and verify principal engineering solutions for a commercial reactor. (See also 1.5.4.5 below.)

Although OTR is presently conceived as a hybrid ETR, the Soviets fully appreciate that any international ETR project would be concerned with a pure fusion device.

1.5.4.2. Physics basis

Operating mode: pulsed inductive drive, burn pulse length ~600 s
Confinement scaling: $\tau_E \propto \langle n\rangle q_I R_0^{2.75} T_i^{-1} a$ (T-10 scaling)
Beta scaling: $\langle\beta_{crit}\rangle = 3.1$ I/aB
$\langle\beta_{max}\rangle = \langle\beta_{crit}\rangle$
Density limit from beta limit at $T_i = 8$ keV or Murakami limit, where $\langle n_{max}\rangle = 16$ B/R_0
Pressure profile is MHD stable and $q_I = 2.1$

1.5.4.3. Phases of operation

Three phases of OTR operation are planned:

Phase 1 – physics and startup experiments with hydrogen plasma.
Phase 2 – burn and ignition studies (including some engineering experiments) with deuterium–tritium plasma.
Phase 3 – full-scale, life-time tests at a first wall load of ~0.8 MW·m^{-2}.

Phase	1	2	3	total
Duration (calendar years)	4	4	7	15
Average availability	0.1	0.1–0.3	0.6–0.7	
Fluence (MW·a·m^{-2})	–	0.8	4.3	~5
Integrated operating time (days)	~150	~360	~1800	~2300
Number of pulses (X 10^4)	~2	~5	~23	~30
Total tritium consumption (kg)	–	28	128	158
Annual tritium consumption (kg·a^{-1})	–	7	18	

1.5.4.4. Tritium supply philosophy

For phases 2 and 3, the reactor will be configured for self-sufficiency in tritium. However, in the initial stage of operation of phase 2, supply from an external source could compensate tritium inventory accumulation in the reactor (in blanket, in tritium system components, including storage, in pumps, etc.) totalling approximately 5 kg. In the beginning of phase 3, an additional amount of 2 kg tritium should be supplied from the external source to compensate for increased tritium storage (equivalent to about one month's consumption). A breeding ratio of ~1.05 will be required in the final phase. The large tritium consumption (18 kg·a^{-1}) in the final phase is due to the high assumed availability (~60–70%).

1.5.4.5. Testing philosophy

Investigations on OTR are scheduled to be performed in three phases, each one being addressed to definite priority directions (see above). In the startup phases of 1 and 2, some physics as well as engineering experiments with hydrogen and deuterium–tritium plasma are planned in regimes considerably different from the base regime of the reactor.

In phase 3, the main objectives are:

(i) feasibility demonstration of reliable and safe electricity and nuclear fuel production by a fusion reactor with the complete self-supply with tritium; tritium and plutonium generation rates in the final phase are estimated at $\sim$18 kg$\cdot$a^{-1} and $\sim$200 kg$\cdot$a^{-1}, respectively;
(ii) maintenance experience under operating conditions close to those in a commercial reactor;
(iii) testing of materials and verification of principal engineering solutions for commercial fusion power reactors. Demonstrated availability at this phase should be increased to 0.6 – 0.7.

At the same time, OTR should be used as a testing facility for lifetime evaluation of alternative technologies, materials and structures in a series of test channels, in particular of those contacting the plasma.

1.5.4.6. Plasma heating

Power during startup to ignition (6 – 10 s) is 50 MW. The contingency power during burn (600 s) should ignition not be reached <30 MW.

1.5.5. INTOR (IAEA)

1.5.5.1. Mission statement

INTOR is viewed as the single major experiment in the tokamak programme between the present generation of large tokamaks (TFTR, JET, JT-60, T-15) and the generation of demonstration reactors (DEMOs). The programmatic objectives for INTOR are:

(i) Demonstration of a plasma physics performance which can be extrapolated to DEMO conditions, and, in particular, the containment of a controlled D–T plasma for long pulse lengths at optimum plasma parameters;
(ii) testing and development of reactor materials and components, and demonstration of their operation at high availability and reliability under conditions approaching those required for DEMO;
(iii) demonstration of the integration of all necessary components into an overall reactor system which can be safely and remotely maintained; and
(iv) investigation of electricity generation and tritium breeding in INTOR in a local structure which is prototypical of DEMO.

1.5.5.2. Technical objectives

The technical objectives of INTOR have been developed to support the achievement of the programmatic objectives, while being consistent with the anticipated technical basis for the design and construction of such an experiment to initially operate in the mid-to-late 1990s. These technical objectives are:

A. Reactor-relevant mode of operation

1. Ignited D–T plasma
2. Controlled burn pulse of >100 s
3. Reactor-level particle and heat fluxes ($P_n \geq 1$ MW$\cdot$m^{-2})
4. Optimized plasma performance
5. Duty cycle ≥70%
6. Availability 25%

B. Reactor-relevant technologies

1. Superconducting toroidal and poloidal coils
2. Plasma composition control (e.g. divertor)
3. Plasma power balance control
4. Plasma heating and fuelling
5. Blanket heat removal and tritium production
6. Tritium fuel cycle
7. Remote maintenance
8. Vacuum
9. Fusion power cycle

C. Engineering test facility

1. Testing of tritium breeding and extraction
2. Testing of advanced blanket concepts
3. Materials testing
4. Plasma engineering testing
5. Electricity production
6. Fluence $\sim$3 MW a$\cdot$m^{-2}.

1.6. Critical issues for Near Term Tokamaks

The work on the next step machine will move, in the next four years, from a conceptual design study to engineering design. In this framework, it is necessary to develop detailed designs of critical components which can verify the design philosophy, and to test prototypes, thereby getting into a position to produce a more reliable costing of the entire tokamak assembly. This approach should allow the scientific community to check, with sufficient confidence, that no technological issues in addition to the ones already identified will stand in the way of a complete NTTR design.

1.6.1. Critical issues for physics

The measured value of the global amplification factor, Q, should be of the order of unity in D–T operation before starting construction of the next step (Q is the ratio of the total nuclear power generated to the total external power supplied to the plasma). Lower experimental values of Q would require significantly higher ignition safety margins in the design of the next ignited tokamak ($Q \sim \infty$). However, values of $Q \sim 1$, extrapolated from D–D operation, could allow the start of a next step construction, provided a somewhat higher safety margin is built into the design.
Other physics issues include:

Understanding and control of disruptions
Study of confinement with alpha particle heating
Study of β stability limit (plasma shape optimization)
Study of density limit
Current density profile control

Operation at $q < 2$
Improvement of heating methods
Plasma edge physics, plasma–wall interaction and impurity control
Improvement of current drive efficiency (desirable but perhaps not essential for next step machines).

1.6.2. Critical issues for engineering and technology

The most crucial issue is the maintenance scheme, which affects the whole machine design and cost. In fact, the maintenance in the reactor hall will probably have to be carried out in a fully remote fashion. Radial or vertical movement for maintenance of internal components is still an unresolved issue and must be clarified by a more detailed design of handling equipment, reactor building and layout of peripherals.

The design and technological development of the plasma-facing components (first wall, divertor plates and RF launchers) also must have a very high priority, simply because they have to be in position at the start of operation. The blanket segment design can be developed on a longer time-scale, but its basic architecture must be defined in the early stage of overall machine design.

The shielding inside the toroidal coils should be more than sufficient to protect the coils from radiation damage because correcting errors, after the DT operation has begun, may prove to be very costly.

Containment of tritium and radioactive debris must be performed during scheduled operation, and all possible efforts should be made to minimize the spread of contamination, in case of accident.

Magnetic field effects on personnel are not an issue at the present time because the coils can be de-energized, if required, in a short time. The control of burning plasma is very important. The available codes must be improved and validated in the present experimental machines to avoid the disastrous consequences of rapid plasma dump (disruption).

The diagnostics for plasma control, suitable for operation in an active environment, must be developed well ahead of time. It is not considered essential to have a high global tritium breeding ratio (TBR) in these machines; however, the capability of achieving the required TBR in a DEMO reactor shall be demonstrated in a test blanket with a local breeding ratio of at least one.

High speed pellet injectors for fuelling and for plasma control have to be developed very early in the research programme in order to provide reproducible plasma conditions.

Finally, advanced superconducting cables for high current density and magnetic fields up to 12 T shall be tested. Special reactor components such as first wall and breeder structural material, coatings for plasma–surface interaction, ceramic insulators for electrical purposes, tritium breeding materials and neutron multipliers as well as tritium permeation barriers must be the product of intense research programmes around the world in the next five to ten years.

1.6.3. Important objectives for the next step in large tokamak reactors

The workshop agreed to eight major objectives for the next step in the tokamak programme:

1. The next generation of tokamak devices must represent a significant step forward to gather key information on the design, construction and operation of an environmentally safe fusion reactor for commercial energy production. The term 'safe' here is defined as intrinsically safe against meltdown, and should possess the following properties:
   low afterheat;
   no necessity of active cooling during an accident (only 'natural' radiative and conductive cooling);
   effective confinement of tritium and activated debris during operation and maintenance.
2. The next step should demonstrate 'long burn' (>10 min) capabilities.
3. With respect to engineering, it should:
   have all components required in a commercial reactor;
   be suitable for testing of materials and have sound engineering solutions for a reactor;
   work at a plasma power density $>1.5\ \mathrm{MW\cdot m^{-3}}$;
   work at a wall loading $>1\ \mathrm{MW\cdot m^{-2}}$.
4. The possibility of tritium self-supply should be demonstrated, although not necessarily achieved.
5. The neutron fluence must allow performance tests of components ($>0.5\ \mathrm{MW\ a\cdot m^{-2}}$) capabilities.
6. Safe and reliable operation and maintenance with the required remote handling techniques shall have to be demonstrated.
7. During an operating period of approximately one year, availabilities of at least 25% should be demonstrated.
8. All the above information should be provided during an operating time of approximately ten years.

1.7. Future plans

The working group compared the various design, construction and operation schedules for the four major NTTR devices; a summary is shown in Fig. 1.1. It was felt that the European

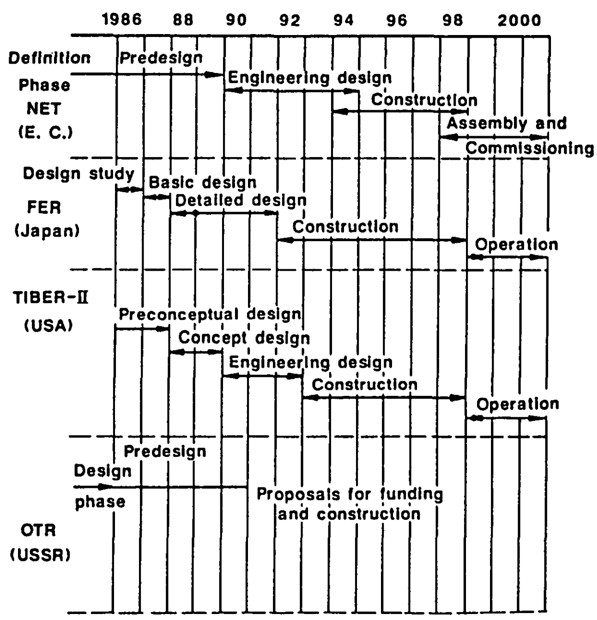

FIG. 1.1. Schedule for major next step tokamaks around the world.

NET design was most developed at this time, followed closely by the Japanese FER concept. The Soviet OTR design was relatively new and the US TIBER-II design is just getting started.

However, it is curious to note that the EC, Japanese and US NTTRs are all scheduled to begin operation in the 1998 to 2000 time period. The Soviet OTR design process is dependent on a major decision in the 1990 period on whether to proceed with the construction of this device.

1.8. Recommendations

The working group decided on the following broad recommendations for the IAEA:

1. A regularly scheduled meeting for discussion of NTTR issues should be organized by the IAEA. This should be initiated relatively soon while all four design groups are in the preliminary stage.
2. A better liaison between next step and present large scale tokamak design groups should be encouraged.
3. The next reactor design meeting should be held in 1989 and not five to six years from now.

2. LONG TERM TOKAMAK REACTORS
(R. Hancox)

2.1. Introduction

This group covered both commercial and demonstration tokamak reactors, since these will have similar technologies and differ only in power output and the need to generate electricity economically. The discussion was primarily concerned with tokamak reactors, but since many nuclear aspects are common to all fusion reactors there was some general discussion of blanket designs and issues such as safety.

The first objective of the group was to review the reactor studies which have been undertaken since the Tokyo Meeting in 1981. However, it was noticeable that there had been few complete reactor design studies during the past five years. On the other hand, there have been several blanket comparison studies concentrating on blanket concepts for power reactors, together with parameter surveys and proposals for significant improvements or innovations in reactor design. Several of these reactor related studies were discussed.

The second objective of the group was to review those issues which are now considered important in relation to the eventual development of fusion power reactors. The main issues were the safety and environmental impact of fusion and the economics of fusion reactors. Other topics which were the basis of lively discussion were the physics and technological assumptions which should be used as the basis of reactor studies, and the preferred power output of demonstration and commercial fusion reactors.

In common with previous workshops, the participants in this group expressed the opinion that long term reactor studies were of critical importance both in guiding the work of the fusion community and in justifying the allocation of resources to the long term development of fusion power.

2.2. Design studies since 1981

The Japanese study, SPTR-P, of a commercial tokamak reactor has been completed since the 1981 workshop. Since then, no major commercial tokamak reactor study has been undertaken by any country although several smaller studies or blanket studies have been undertaken. The ANL DEMO study has also been completed since the 1981 workshop, and more recently a DEMO tokamak reactor study has been completed in the UK. In the USSR in 1984, the feasibility was analysed for using a fusion plant for hydrogen production. The four studies discussed at this workshop are described below. Some parameters from the four studies are given in Table 2-I. Commercial reactor studies based on confinement systems other than the tokamak such as MARS (mirror), TITAN (RFP), and Cascade (inertial) were dealt with by other groups.

2.2.1. SPTR-P (Japan)

Preliminary information on this study was presented at the 1981 workshop. The study was completed at the end of fiscal year 1982 [1]. Its main feature is that the 1000 MW(e) reactor is submerged in a water pool (10^4 tonnes), acting as shielding for the superconducting toroidal field coils. The water shield concept reduces the quantity of radioactive waste for disposal and overcomes radiation streaming.

The reactor uses the solid tritium breeder lithium oxide, and a modified austenitic steel for the first wall and blanket structural material. The water coolant is operated under conditions typical of pressurized water fission reactors. The use of lithium enriched to 30% ^{6}Li and of beryllium as a neutron multiplier gives local breeding ratios of 1.26 in the inboard blanket and 1.29 in the outboard blanket, and an overall breeding ratio of 1.05. RF power at a frequency of 2.5 GHz provides lower hybrid heating and steady state current drive.

The SPTR-P concept was developed without finding any fatal engineering problems. Particular advantages of the water pool design were that the weight of the reactor was markedly reduced, maintainability was improved, no solid shields were required around RF and exhaust ducts, and space around the reactor was used more effectively.

2.2.2. ANL Demonstration Tokamak Reactor (USA)

A Power Demonstration Tokamak Reactor study [2] was completed in 1982, and an update of the reactor parameters was performed in 1986. The update assumes that enhanced physics (higher beta in the second stability regime) will have been tested in previous devices for use in DEMO. The basic goal of the DEMO study was to provide a technical perspective and conceptual design of the tokamak that might follow an NTTR/INTOR type of test reactor. The role of such a device in the overall strategy to develop fusion power was considered. The effort was focused on designing the key features of such a device with the aim of providing design information for guiding the research and development programme. Whilst a reference conceptual design for the DEMO was developed, more emphasis was placed on exploring major design features.

The DEMO is expected to produce a few hundred megawatts of net electrical power and to demonstrate the ability to

TABLE 2-I. REACTOR PARAMETERS

	SPTR-P	ANL-D		Culham-D		HTR
Net electric power (MW)	1000	290				
Fusion power (MW)	3200			2000		
Gross thermal power (MW)	3700	1050				10 000
Neutron wall loading ($MW \cdot m^{-2}$)	3.3	1.8		2.6		5.0
Major radius (m)	6.9	5.2		6.8		8.75
Minor plasma radius (m)	2.0	1.3		1.6		2.9
Elongation	1.6	1.6		1.6		
Plasma current (MA)	16	9.0		9.3		
Toroidal β (%)	7.0	8.0		7.7		
Magnetic field, on axis (T)	5.2			6.0		
Magnetic field, max (T)	12	10		11		
Breeder	Li_2O	solid	solid		$Pb_{83}Li_{17}$	Li
Structure	PCA[a]	st. steel	steel[b]		steel[b]	
Coolant	H_2O	H_2O	He		He	H_2O
Outlet temperature (°C)	330	320	580		600	1000
Plasma heating	LH	REB		NBI		
				80 MW		
Current drive	LH	REB		–		
	80 MW					
Pulse length (s)	Cont.	Cont.	1000			

[a] Titanium modified stainless steel.
[b] Ferritic steel.

extrapolate to an economically competitive system. Specifically, the following objectives for the DEMO can be stated:

(i) Demonstrate a level of performance of all components in an integrated power plant system which is satisfactorily extrapolatable to a first commercial reactor;
(ii) demonstrate plant availability at a level which will satisfactorily extrapolate to a first commercial reactor;
(iii) demonstrate that a tritium breeding, power producing blanket can operate at conditions required for a commercial reactor;
(iv) demonstrate safe and environmentally acceptable operation; and
(v) demonstrate compatibility with electrical grid operations including off-normal conditions.

In summary, the DEMO study developed a reference conceptual design and focused on in-depth investigations of key issues for the reactor systems that require extensive development in parallel to and beyond NTTR/INTOR. These systems include non-inductive current drive, impurity control, first wall and breeder blanket, and reactor configuration and maintainability.

2.2.3. *Culham DEMO Tokamak (Euratom)*

This design study was undertaken by the United Kingdom Atomic Energy Authority during the period June 1983 to December 1984 as part of the European Fusion Technology Programme [3]. The primary objective of the study was to identify promising blanket designs that might be used in a future demonstration reactor, and as a result tritium breeding, electricity generation and maintenance were key features of the design. During the course of the study the interdependence of the blanket design with other reactor components made it necessary to consider other aspects of the reactor. Consequently, the study eventually considered first wall, divertor, magnetic field coil, tritium extraction, auxiliary heating and power cycle designs and layouts.

The reactor design was based on the INTOR pulsed tokamak with an increase in size and wall loading to raise the fusion power from 600 MW to 2000 MW, giving a total thermal power of 2600 MW and an electrical output of 1100 MW. Two different blanket designs were considered, one that utilized a liquid lithium-lead breeder (30% Li, 70% Pb) and the other a solid lithium metasilicate ceramic breeder. In both cases a honeycomb ferritic steel blanket structure was used in which the toroidal breeder elements were contained. Tritium extraction was provided by direct removal from the coolant using a molecular sieve bypass flow extraction plant. Other notable features of the design included a helium cooled copper first wall with radiating tungsten tiles and a liquid tin covered thin tungsten–rhenium divertor target cooled by high pressure water. The purpose of the copper first wall was to provide a degree of passive plasma stabilization, while the liquid tin divertor was designed to allow the divertor to operate for two years before replacement.

The study established that the critical factors that controlled the design were the neutron economy, which was important in ensuring a global tritium breeding ratio in excess of unity, high temperature operation, which was necessary to achieve reasonably high thermal efficiency, and high availability. Key issues that required further study included the toroidal field coil support structure, plasma passive stability, direct tritium extraction from the blanket coolant, liquid lithium lead corrosion, startup with high Z first wall material and the feasibility of liquid metal covered divertor designs. In general, it was concluded that the design options chosen provided a realistic basis for more detailed study, but that considerable research and development were required in many areas.

2.2.4. Fusion plant for hydrogen production (USSR)

The feasibility of a hydrogen producing plant based on a fusion tokamak reactor and high temperature electrolysis was analysed in 1984 [4]. The blanket neutronics were optimized to ensure that a considerable fraction of the thermal power was released within a tritium free high temperature zone, filled with zirconia pebbles. The fraction of power in this zone was shown to be about 30%, while the tritium breeding ratio exceeded unity. Liquid lithium was considered to be the breeding material, located in two zones on either side of the high temperature zone. The structural material for the lithium zones was molybdenum alloy, and for the high temperature zone zirconium. The maximum working temperature of the lithium was 1300°C, and of the high temperature hydrogen-production zone 1500°C. For the heat removal from the lithium zones, a system without any pumping equipment was envisaged, based on heat pipes. To prevent tritium permeation into the steam circuit a radiative heat exchanger was suggested with a cooled quartz tritium barrier and a vacuum gap.

The feasibility of the fusion reactor blanket parameters adequate for the direct steam heating and a high temperature electrolysis cycle has been demonstrated. The overall system efficiency with respect to the production of hydrogen was about 50%. This plant concept seems to offer a viable prospect for future power systems, although several material and other general issues of the nuclear fusion and high temperature electrolysis remain to be successfully solved.

2.3. Tokamak related design studies

An advanced tokamak reactor based on the Spherical Torus has been proposed in order to extend tokamak operation to the low aspect ratio ($R/a > 2$), paramagnetic regime. With first stability regime beta values approaching 30% according to Troyon scaling ($\beta \propto I/aB$), resistive coil reactor embodiments become feasible. This favourable beta performance has not yet been confirmed experimentally. A preliminary conceptual commercial reactor design study of a Spherical Torus device has been performed at Los Alamos [5] in the context of the FY-1985 Tokamak Power Systems Study. The design incorporates a double null, poloidal field divertor for impurity control; RF assisted startup with steady state current drive demountable resistive toroidal field coils to facilitate maintenance; and a flowing lithium–lead eutectic breeder/coolant in an out-board only blanket. Key parameters are summarized in Table 2-II for representative systems.

The following conclusions have emerged from the Advanced Tokamak Reactor study:

(i) A resistive coil tokamak at high beta gives a high mass power density (⩾150 kW(e) per tonne) and can be cost competitive with a superconducting concept (e.g. STARFIRE), despite higher recirculating power fractions.
(ii) Technology requirements are modest.
(iii) The approach has considerable flexibility in power output, configuration, impurity control and maintenance approach which has not been fully explored or exploited at this preliminary level of investigation.

TABLE 2-II. ADVANCED TOKAMAK REACTOR PARAMETERS

Net electric power (MW)	500	1000
Gross thermal power (MW)	2057	3710
Neutron wall loading ($MW \cdot m^{-2}$)	3.2	5.9
Major radius (m)	2.7	2.7
Minor plasma radius (m)	1.5	1.5
Plasma current (MA)	39.8	46.2
Toroidal beta (%)	29	29
Magnetic field, on axis (T)	4.1	4.8
Magnetic field, max. (T)	6.9	8.0

2.4. Blanket studies

2.4.1. Blanket Comparison and Selection Study (USA)

The USA completed a comprehensive Blanket Comparison and Selection Study [6], which was an evaluation of power reactor blanket designs and the status of blanket technology. It also served as an excellent basis for further development of blanket technology. This study provided an evaluation of over 130 blanket concepts for the reference case of an electric power producing, DT fuelled reactor in both tokamak and tandem mirror configurations. Based on a specific set of reactor operating parameters, the current understanding of materials and blanket technology, and a uniform evaluation methodology developed as part of the study, a limited number of concepts were identified that offer the greatest potential for making fusion an attractive energy source. Based on the systematic evaluation performed, the leading concepts were judged to be:

(i) Lithium breeder cooled by direct circulation, with vanadium structure: overall top rated concept for tokamak reactors. Advantages include advanced, low activation, high temperature structural alloy, and inherent simplicity of the self-cooled concept.
(ii) Lithium-lead breeder cooled by direct circulation, with vanadium structure: high ranking for tandem mirror reactor only.
(iii) Lithium oxide breeder, helium cooled, in ferritic steel structure: top rated solid breeder concept. Design features include lobular pressurized blanket modules with cool helium feed to first wall.
(iv) Lithium breeder, helium cooled, in ferritic steel structure: rated well below first three.

2.4.2. Tokamak Reactor Blanket Evaluation (Euratom)

A screening review of candidate blanket concepts and an evaluation study of their capability to meet commercial reactor requirements (full breeding capability, power production, tractable design) has been carried out [7], with a view to deriving consistent recommendations for the orientation of the European Fusion Technology Programme launched in 1982.

The methodology involved a classification of existing blanket concepts according to major design features (breeder, coolant, arrangement and direction of the cooling lines) and an assessment of the impact of the selected design options upon the major blanket characteristics such as the breeding ratio, the breeder temperature control and the overall engineering complexity.

The study generated data evaluation matrices for five major classes of blanket concepts and finally recommended three directions for the future engineering investigations and experimental work:

(i) The amount of work necessary to establish the confidence in lithium self-cooled blankets was estimated to be disproportionate to the budget available for Fusion Nuclear Technology.
(ii) A water cooled quasi-static $Li_{17}Pb_{83}$ tubular blanket, with several poloidal breeder rows, was proposed as the reference liquid blanket concept.
(iii) The association of helium cooling in a quasi-radial direction with the use of cladded pins was recommended for efficient control of the solid breeder operating temperature; breeding enhancement may be achieved by dispersing beryllium among breeder elements.
(iv) The present database on solid breeders was judged insufficient to confidently establish the capability of water cooled blankets to efficiently control the breeder temperature for a satisfactory tritium release. Experimental evidence that ceramics could be specially developed to release the tritium at about the coolant temperature (300–400°C) was estimated a prerequisite for future work in this area on water cooled solid blankets.

These guidelines are still broadly reflected in the continuing blanket design activity in Europe.

2.4.3. Power reactor blanket concepts (Japan)

Five blanket designs have been studied in the context of a power reactor whose parameters were similar to those of SPTR-P. The following conclusions were reached.

The water cooled lithium oxide concept with modified austenitic steel structure is a realistic option considering the present database. However, the maximum coolant temperature is restricted to a relatively low level, which causes the technological problem of breeder temperature control. Detritiation systems for the removal of tritium from the primary water coolant may have a cost impact if the tritium permeation rate is high.

The helium cooling concept has the advantage of high thermal efficiency. For a helium cooled blanket with lithium oxide breeder, a high tritium breeding ratio is possible by using a mixture of breeder and beryllium multiplier, which might offer the possibility of a net breeding ratio greater than unity with only partial blanket coverage of the tokamak.

The helium direct cooling blanket concept eases the problem of breeder temperature control and eliminates a tritium purge gas line. Its disadvantage is that the breeder packed tube length is required to be short to keep the coolant pressure drop low, and connections between coolant breeder tubes and coolant manifolds become rather complicated. The coolant pressure drop through the breeder packed tube or the tube connections should be designed to be acceptably low.

The liquid lithium self-cooled blanket concept will provide a simple blanket structure and have less neutron irradiation effects on breeder properties.

These are only a few examples for the engineering and technical issues covered in the study.

2.5. Reactor improvements

2.5.1. Tokamak Power Systems Studies (USA)

The last major conceptual design study of a tokamak power reactor in the USA was STARFIRE, which was carried out in 1979 and 1980. Since that time American studies have concentrated on NTTR, demonstration reactors, parametric systems studies, scoping studies and studies of selected critical issues such as pulsed versus steady state operation and blanket requirements. During this period, there have been many advances in tokamak physics and reactor technology, and there has also been a strong recognition that it is desirable to improve the tokamak concept as a commercial power reactor candidate. During 1984 and 1985, several organizations participated in the Tokamak Power Systems Study with the objective of developing ideas for improving the tokamak as a power reactor. Some highlights of this activity include the following:

(i) A range of reactor power output levels. This is important to improve flexibility in terms of siting, utility grid size, and market penetration.
(ii) Features which will reduce unit capital costs include increasing the mass power density of the reactor,

increasing the overall efficiency of the power conversion system and reducing reactor support subsystems. Important figures of merit for this area include blanket energy multiplication in the blanket and gross thermal efficiency.

(iii) Design simplification. This will reduce maintenance requirements and increase capacity factor. Examples of such design features include steady state operation, first wall or limiter for impurity control, reduced plasma shaping and control requirements, combined plasma startup, heating and current drive systems, large duct single pass liquid metal blankets, and mechanically integrated first wall/blanket/shield.

(iv) Enhanced safety and environmental features. Key items include achieving shallow land waste burial for all reactor components, and inherent safety features.

In an effort to improve the economics and environmental impact of a tokamak power reactor, several ideas were examined to increase the reactor's mass power density. A key feature is increasing the plasma beta. Several concepts were examined which include higher beta in first stability regime via very low aspect ratios or highly elongated plasma shapes and access to the second stability region with and without beam shaping of the plasma cross-section. Both copper and superconducting coil concepts were considered for the first and second stability regimes. Steady state operation was examined for a number of designs. Both fast wave current drive and electron cyclotron heated tokamaks were examined for the second stability regime. New ideas for impurity control were developed, including the concept of a helium pumping first wall. The major effort since the Blanket Comparison and Selection Study for improving fusion reactor blanket performance can be classified into the following areas: improvements in self-cooled liquid metal concepts provided by reduced magnetic fields due to higher beta operation and extensive use of electrically insulated walls; simplification of solid breeder concepts by innovative tritium recovery scenarios and improved economic performance by incorporating substantial amounts of beryllium as energy and neutron multipliers; expanded use of reduced activation materials; partial blanket replacement to minimize radioactive waste management requirements; and mechanically integrated first wall/blanket/shield that increase design simplicity and safety.

In general, a number of new concepts have been developed which substantially improve the potential commercial attractiveness of tokamak power reactors.

2.5.2. Innovative ideas (INTOR)

In January 1986 an INTOR related Specialists' Meeting was held on innovations which aim at significantly improving the reactor prospects of the tokamak concept. About half of the proposed innovations were recommended for further study with high priority. Whilst most of the innovative ideas could be applied directly to Next Step devices, others only have relevance to full scale electric power generating reactors. An outstanding example of the latter is power conversion by the MHD process. The idea consists of the combination of a number of steps: (i) the reactor will be operated at an elevated temperature such that about half of the fusion-alpha particle power will appear as synchrotron radiation; (ii) as coolant for first wall, blanket and shield a medium will be used appropriate for MHD power conversion; (iii) MHD power convertors, operating at rather practical temperatures of about 1000°C, are arranged in the shield area and use the toroidal magnetic field existing there anyway (some modification of the coils is necessary, though); (iv) The non-equilibrium ionization necessary at these operating temperatures is provided by the high-power synchrotron radiation (microwaves) transferred to the MHD chambers by waveguides. In this way the total reactor output power, including that of the neutrons, can be directly converted into electric power by the MHD process with its high conversion efficiency. A substantial saving in the balance of plant should thus be the consequence. An appropriate shaping of the inner first wall could be used for current drive by the same synchrotron radiation.

It is necessary to check this method in all details. If it turns out to be viable it has the potential to lead to a considerable reduction in cost of electricity. There is obviously no point, however, in incorporating this method into today's fusion devices.

2.6. **Tokamak parameter studies and economic issues**

Parameter studies have been undertaken over the past few years to identify

(1) the rough parameters of devices that form a logical sequence of steps to a power reactor;
(2) the effect of improvements that might reasonably be expected to come from future advances in plasma physics and technology; and
(3) the effect on power reactors of progress to date in plasma physics attainment, or of specific choices between design options.

Most of these studies choose cost as the figure of merit for optimization, but emphasize capital and generation costs to different extents. They also make somewhat different engineering and plasma physics assumptions and therefore

TABLE 2-III. LOGICAL SEQUENCES TO FUSION DEVELOPMENT

a. Euratom. Fusion systems based on same physics assumptions.

	R (m)	a (m)	P_n ($MW \cdot m^{-2}$)	P_{out} (MW(e))
NET	5.2	1.4	1.0	–
DEMO	8.2	1.9	1.8	670
Power reactor	9.3	2.4	2.2	1240

b. USSR. Hybrid reactors with superconducting coils (SC) or normal coils (NC).

	R (m)	a (m)	P_n ($MW \cdot m^{-2}$)	P_{out} (MW(e))
Engineering test reactor (NC)	3.9	1.1	1.5	–
Experimental reactor (SC)	6.2	1.5	0.8	80
Power reactor (NC)	6.4	1.7	1.3	1300

cannot easily be compared. Studies in the USSR have concentrated solely on the exploitation of the hybrid reactor concept, whereas elsewhere only pure fusion concepts are under study.

Two typical studies in the first category produce the results quoted in Table 2-III. They indicate the hardening of ideas on the objectives of each device, although these are only preliminary at present. The results illustrate two alternative strategies, the one building up the wall power loading to the reactor level, the other tackling the difficulties of higher wall loading in smaller devices to build up experience quickly. Another recent example of this type of study is the application of high field coils, superconducting or resistive, by MIT. The rationale for such an approach is the emphasis on strong Ohmic heating to decrease the requirement for auxiliary heating power and associated blanket penetration and the possibility for competitive reactor operation at low beta values with simplified poloidal field coil configurations. The concept has been applied to near term ignition experiments, engineering test reactor, and commercial reactor extrapolation. In particular, a commercial power reactor incorporating advanced niobium–tin or niobium nitride superconducting coil technology has been proposed. This study claimed that significant cost reductions and design simplifications may be attainable from the application of high field coils to commercial tokamaks.

The Tokamak Power System Study performed in the USA is the major example of the second category of study. It has identified the incentives for improving the tokamak by increasing mass power density, increasing plant efficiency and reducing plant support systems while at the same time enhancing the environmental and safety properties of fusion and keeping the design simple.

Studies in the third category have highlighted the incentive for improving the cost of tokamak reactors by improving beta levels within a given plasma configuration. They have also investigated the impact on the cost of present predictions of confinement time at ignition and thermal stability of reactor plasmas.

Parameter studies have also attempted to interrelate the mass power density in the fusion power core (i.e. power sent out/fusion power core mass) or mass expenditure (its inverse) to the cost of electricity or capital cost. This has led, in the USA, to the adoption of 100 kW(e)/tonne as a target measure of attractiveness of all fusion concepts. The rationale for emphasizing reactor systems with higher mass power density includes reducing the direct cost of the fusion power core (i.e. coils, blanket, and shield), and reducing the dominance of the fusion power core as a fraction of total plant costs in order to reduce the overall cost sensitivity to uncertainties in confinement physics and unit costs. Additionally, indirect but important cost benefits in terms of reducing construction time and costs, simplified repair and maintenance approaches and reduced development time and risk have been associated with higher power density systems.

There are however, some contrary viewpoints on the hardness of this target from other national groups and doubts concerning the wiseness of adopting such targets when so little is known in so many technological areas relevant to reactors, and when fatigue, lifetime, reliability and safety issues are inadequately considered in present economic modelling. Interestingly, most reactor studies since 1974 have produced costs of electricity in the range of 30 to 40 mills (1980)/kW·h, whereas the mass power densities of these designs have ranged between approximately 40 and 120 kW(e)/tonne.

In summary, therefore, these parametric studies highlight the different emphasis given by each national group to economic issues of reactors. In the USSR, the main line of development is the hybrid reactor, which is virtually now ignored as an option by other groups. In the USA there is a strong drive towards compactness and high power density in an attempt to obtain cost advantages against the competitors of fusion. In Euratom the issues of reactor cost competitiveness are considered to be of low credibility at this stage for reaching broad conclusions. These must wait for the development of reactor relevant design solutions on the next step machine.

2.7. Special topics

2.7.1. Commercial and demonstration reactor size

Fusion research and development should incorporate some flexibility in the plant output for commercial and demonstration plants. Two key uncertainties necessitating this flexibility are the grid or utility size, and the balance between the generating cost and total capital cost, i.e. the investment risk. These uncertainties arise because fusion lies significantly in the future and because the economic environments of utilities differ. The apparent range of appropriate targets for commercial plants is 600 to 1500 MW(e) with lower values more appropriate for smaller utilities emphasizing minimum investment risk and high values more appropriate for larger utilities emphasizing minimum generating costs.

Demonstration plant design requires sufficient output and technological similarity to expected commercial designs that the scale-up to commercial design will be adequately low risk. That is, the demonstration design should be nearly identical to commercial design in selection of design approach, technological requirements, material selection, and design margins. The scale-up in net power levels is less important than the technological scale-up, but it may give an overall picture of the difference between demonstration and commercial scale plants. Typical accepted power scale-up in fission technology experience is a factor of 3 to 5. For example, the demonstration fast breeder reactor Phoenix is 233 MW(e), which is being scaled to Creys-Malville at 1200 MW(e). Thus, if fission experience is found to be relevant to fusion, the appropriate demonstration plant output is probably in the range of 150 to 300 MW(e).

2.7.2. Safety issues in a commercial reactor design

Although various reference designs for fusion power facilities are still at the conceptual stage, there is now a general recognition by conceptual designers that safety and environmental factors must be addressed very early in the design process. Although more progress is needed, this approach promises to enhance the safety and economic attractiveness of fusion power. In the USA and Canada, this approach has been embodied in the concept of inherent safety in design which relies on the passive conductive and radiative heat transfer properties of materials comprising the blanket and first wall. This concept is viewed as desirable to ensure that fusion remains attractive compared with advanced fission power plant designs.

There is continuing effort to reduce system inventories of tritium in operating plant designs, such that the worst case releasable fraction does not exceed 100 to 500 g of tritium (as HTO). Depending on site conditions, meteorology, stack height, exclusion boundary, etc., acute releases of this magnitude would result in off-site doses of 1.0 to 25 rem. If severity of consequences is inversely proportional to frequency of occurrence, and if 25 rem is viewed as an upper bound, there is confidence that virtually all national nuclear regulators will license such fusion power facilities as acceptably safe.

Although some reactor designs may have the potential to release other radionuclides in quantity (e.g. ^{210}Po), tritium is generally seen as the dominant critical radionuclide.

With respect to chronic releases, there is a general consensus that routine site emissions in the range of 10 to 100 Ci/day (as HTO) are acceptable, since off-site doses are very likely to remain below 5 mrem per year. Again, variations depend on stack height, exclusion boundary, local agronomy, etc. There is general optimism that such limits can be met at operating power facilities without undue costs or design constraints.

Since the bulk of maintenance in a fusion reactor hall will be performed remotely, occupational radiation doses will likely result from exposures to tritium and gamma radiation sources in secondary areas such as heat exchanger rooms, pump rooms, maintenance and waste management areas. Activated corrosion products transported outside the reactor core by circulating fluids are the most likely potential source of penetrating radiation. With proper layout, isolation of hazards and control of radiation sources, there is confidence that occupational dose experience will be as good as, or possibly better than, what is presently considered good safety performance at operating nuclear facilities.

The goal of selecting low activation materials for incorporation into the reactor structure such that reactor decommissioning wastes will be low level continues to be of a high priority. The objective is to ensure that all wastes will meet the radiological criteria for shallow land burial.

There is continuing dissatisfaction with existing nuclear design codes with respect to their application in fusion power reactor design studies. Existing codes, developed for fission, do not allow exploitation of the many inherent safety advantages of fusion, and result in inappropriate cost loads on design and construction estimates. There is a feeling, particularly in the USA and Canada, that new design codes unique to fusion must be developed.

2.7.3. Assumed basis for reactor parameters

A subject on which there was no agreement within the group was the extent to which plasma physics or technological parameters could be extrapolated to take account of anticipated future improvements.

One view was that only present knowledge should be used as the basis of reactor studies. This choice leads to conservative designs which may turn out to be economically unattractive in comparison with fission reactors or other energy sources. However, there is a high certainty that the devices could be constructed. The opposite view was that optimism, based on past experience of steady improvements, was allowable. This choice might lead to attractive and competitive designs, although it is clear that they represent future expectations rather than present reality.

The clearest example of such a choice is the level of beta assumed for the plasma. The choice of value, supported by present experimental results, would be about 4% as in the NET, Demo and power reactor extrapolations. In recent American studies, on the other hand, values of 7% to 9% are common and in some cases much higher values have been postulated on the expectation of operation in the second stability regime – an expectation with some theoretical support but no experimental justification. Similarly, several designs invoke steady state operation assuming efficient current drive mechanisms which have not yet been demonstrated experimentally.

In the area of technology several extrapolations are often assumed. The reliable operation of large superconducting magnets at very high fields and stresses, or the routine containment of tritium with very low loss rates still have to be demonstrated. A key technology is the development of structural materials with resistance to radiation damage after high doses of 14 MeV neutrons, where neither suitable materials nor neutron sources for testing yet exist.

Clearly some compromise in this area is necessary. Some extrapolation of present knowledge is reasonable, but such extrapolations in reactor designs should be clearly identified, and if their effect is critical they should be fully justified.

2.8. Conclusions – the need for further reactor studies

Power reactor studies serve two important functions. Firstly, they guide and focus the technology development programme. Secondly, they are the basis of assessments of the long term prospects for fusion power.

The majority of the fusion technology programmes are oriented towards the needs of Next Step devices. Nevertheless, long term reactor studies should have a significant influence. In view of the wide choice of technical solutions to most problems, it is important to select those solutions which also have a good long term potential for power reactors. The development of technologies which only have applications in Next Step devices is wasteful of resources and may extend the timescale for the eventual construction of a demonstration reactor. Examples include the development of low temperature blanket concepts which could not be operated at the power densities or temperatures required for a power reactor; such blankets may appear attractive in terms of shorter development times, but delay the establishment of an adequate database and the operating experience required for the design of a reliable demonstration reactor. Similarly, design studies provide guidance for materials selection since power reactor conditions will be very different to those in a Next Step device.

The second reason for requiring up to date reactor studies is that they are the only present basis for assessing the potential of fusion power. There is a growing interest in the economic and environmental implications of fusion, and these can only be judged through detailed design studies. As knowledge of plasma physics and fusion technology progresses, it is desirable that the implications of the progress be fully assessed. In some countries the funding of fusion programmes is dependent on the present perception of the prospects for economic and safe fusion power, and it is therefore essential that such judgements be founded on the best available design studies.

It is usually true that power reactor design studies utilize less than 1% of the financial resources of national fusion programmes, whereas their impact on both the programmes and the public acceptance of fusion is much greater.

REFERENCES

[1] TONE, T., et al., Fusion Technol. 8 (1–2A) (July 1985) 731.

[2] ABDOU, M., et al., A Demonstration Tokamak Power Plant Study, Argonne National Laboratory. Rep. ANL/FPP-82-1 (September 1982).

[3] COOKE, P.I.H., REYNOLDS, P., A Demo Tokamak Reactor – Aspects of the Conceptual Design, Culham Laboratory Rep. CLM-R 254 (August 1985).

[4] VOLOSHCHENKO, G.N., et al., A Fusion Installation for Hydrogen Production Based on a Tokamak Reactor and High Temperature Electrolyser, Questions on Atomic Science and Technology: Thermonuclear Fusion 2 (1985) 44.

[5] PENG, Y-K.M., et al., Fusion Technol. 8(1–2A) (July 1985) 338; MILLER, R.L., et al., LANL Rep. LA-10740-MS (1986).

[6] SMITH, D.L., et al., Blanket Comparison and Selection Study – Final Report, Argonne National Laboratory Rep. ANL/FPP-84-1 (September 1984).

[7] CARRÉ, F., et al., Fusion Reactor Blanket Comparative Study, Rep. EMT/SEMA/BP/584 – CEA (1984).

3. STELLARATOR WORKSHOP
(I.N. Sviatoslavsky, F. Rau)

3.1. Reactor prospects

3.1.1. Introduction

The stellarator class of magnetic confinement fusion reactor designs continues to present a high level of interest to researchers around the world. Such reactors are characterized by distinct features which are superior to other magnetic fusion devices. They include ignited, steady state, disruption-free operation with reduced recirculating power fractions, an optional natural magnetic divertor and startup on existing magnetic surfaces. Furthermore, in principle, the magnetic configuration with sufficient β potential can be generated by a single coil system. International progress continues on a number of experiments, with their plasma parameters similar or even superior to those obtained on comparable tokamaks.

3.1.2. New developments since 1981

There have been several new developments with applicability to stellarator power reactors. Many of these developments have been aimed at modularizing the magnet system in order to make it more maintainable and to provide better access to reactor components such as blanket and shield modules. Refinements of modular non-planar coil systems continue at a high level at Garching, FRG, both in optimization with respect to various plasma performance parameters as well as geometric and structural considerations. Modular torsatron coil systems with multipolar compensation coils are being developed at the Kharkov Physico-Technical Institute, USSR, with emphasis on reproducing as closely as possible the magnetic topology of a continuous coil torsatron. Development of modular torsatrons is also taking place at ORNL, USA, under the name of symmotrons (symmetric modular torsatron). These have identical coils, each representing a field period, and can have windbacks on the inboard or the outboard side.

Development of continuous helical coil torsatrons (heliotron) is also proceeding in Japan, the USSR and the USA. The Japanese Heliotron H reactor design is contemplating demountable helical windings. Such demountable joints are at present being developed by Japanese industrial firms. At ORNL, USA, a low aspect ratio ($A = 4-5$) continuous coil torsatron reactor, ATF′, is presently being scoped out. It is patterned after the ATF experiment but has a lower aspect ratio and has the potential of operating in the second stability regime.

Two other developments aimed at improving the torsatron magnetic system have been studied by the Kharkov, USSR, group. The first is a torsatron with an additional toroidal field for reducing the helical ripple which when taken in association with the electric field permits a reduction of the reactor aspect ratio. The second is a simpler helical winding $\ell = 2$, $m = 2$ system with two nearly planar conductors.

Advanced stellarators developed at Garching, FRG, differ from standard stellarators by a certain reduction of the secondary plasma currents associated with reduced particle and energy losses as well as reduced Shafranov shift at finite β. Reasonable values of the aspect ratio, rotational transform and a magnetic well are maintained.

Very thin blankets utilizing beryllium metal as a multiplier/moderator, LiPb breeding material and He gas cooling are being proposed for the Garching, FRG, ASRA6C Advanced Stellarator power reactor and the ORNL ATF reactor. These blankets range in thickness from 16 to 35 cm and can give local breeding ratios up to 2.0. Since they are very thin, they can be designed with a uniform cross-section instead of tracking the plasma shape in the toroidal direction. Such a blanket has been adopted by ASRA6C and represents a vast improvement from the construction standpoint.

Alternatively, breeding under the coils can be sacrificed in exchange for a smaller coil radius, possibly leading to a lower aspect ratio and a reduced field at the conductor, as in the case of ATF′.

3.1.3. System studies

Studies of stellarator reactor systems at various levels are ongoing at all major laboratories working in the field. Short

TABLE 3-I. KEY FEATURES OF STELLARATOR REACTORS

Magnetic topology produced by currents outside the plasma, therefore: Startup from existing magnetic surfaces and continuous burn, no net plasma current required, no dangerous disruptions.

1. ATF′, Oak Ridge, USA	$\ell = 2$, $m = 9$ continuous coil low aspect ratio torsatron reactor, relatively high current density in helical winding, two additional VF coils, operating at average $\beta \cong 4.9\%$ in second stability regime with thermal power of 2500 MW at a first wall neutron load of 3 $MW \cdot m^{-2}$, modern thin gas cooled LiPb/Be blanket between helix, W/HT-9 shield.
2. Heliotron H, Kyoto, Japan	$\ell = 2$, $m = 15$ continuous coil moderate aspect ratio heliotron reactor, large rotational transform and shear, thermal power 3.4 GW, $\beta = 6\%$, first wall neutron load 1.3 $MW \cdot m^{-2}$, modular blanket; prototypes of demountable helix joints under development.
3. U-2MR, Kharkov, USSR	$\ell = 2$, $m = 4$ (and 2) torsatron reactors with additional toroidal field, moderate aspect ratio and low helical ripple ($\epsilon_h = 5\%$). Power 3.8 GW (and 1.9 GW) at $\beta = 8\%$ (2.7%) calculated with neoclassical transport and electrical field. At first wall 4 (and 2) $MW \cdot m^{-2}$.
4. TNPP, USSR	$\ell = 3$, $m = 21$ modular torsatron; 21 modules containing six compensation windings and three helical parts; helical diverter. Operation in extended plateau regime at 9.3 GW at $\beta = 10\%$ and 2 $MW \cdot m^{-2}$ neutron load on first wall.
5. MSR IIB, Los Alamos, USA	Modular $\ell = 2$ stellarator reactor with single coil system for magnetic topology, modular blanket and shield, pumped limiters for edge control, at $R_0 = 23$ m, $a = 0.8$ m, $\beta = 8\%$. Fusion power 3713 MW.
6. ASR25T7, Garching, FRG	Advanced stellarator reactor with improved confinement properties and equilibrium β, using 10 NbTi coils in each of the five field periods, low current density and peak field at coils, relatively large distance between plasma and coils.
7. ASRA6C, Garching, FRG:	Smaller $R = 20$ m by use of modern thin blanket and shield, six elliptical bore Nb_3Sn coils per field period, at $B_0 = 5.3$ T on axis and $B_m = 10.4$ T at coils, improved maintenance scheme with pumped limiters. Operation at 3.9 GW, and $\beta = 5\%$ with 300 MW additional radiative layer at edge under improved neoclassical heat conduction.

TABLE 3-II. TYPICAL STELLARATOR REACTOR PARAMETERS

Reactor	ATF'[d]	HELIOTRON H	U-2MR	TNPP	MSR II B	ASR25T7	ASRA6C
Country	ORNL (USA)	Kyoto (Japan)	Kharkov (USSR)	Kharkov (USSR)	LANL (USA)	Garching (FRG)	Garching (FRG)
System	Continuous helix and vertical field		Continuous helix w/addit. tor. field	Modular torsatron	Modular stellarator	Modular adv. stellarator	Modular adv. stellarator
Thermal power, P_{th}(GW)	2.5	3.4	3.8	9.3	4.0	3.6	3.9
Neutron wall load, P_n ($MW \cdot m^{-2}$)	3.3	1.3	4.0	2.0	2.0	1.1	1.5
Average β (%)	4.9	6	8	10	8	5.3	5.0
Magnetic topology							
Aspect ratio, $A = R_0/a$	4.6	12.4	9.4	17.5	28	14.6	12.5
Major radius, R_0(m)	10.8	21	15	37	23	25.5	20
Average minor radius, a(m)	2.3	1.7	1.6	2.1	0.81	1.75	1.60
Multipolarity, m/ℓ	9/2	15/2	4/2	21/3	4/2	5/(1,2,3)	5/(1,2,3)
Rotational transform, axis/edge ι_0/ι_a	0.25/0.9	/2.2	0.57/0.73	/2.4	0.63/0.80	0.58/0.61	0.47/0.46
Magnetic well, $-\delta v'/v'$ (%)							
Magnetic hill, $+\delta v'/v'$ (%)	+6				−0.05	−0.10	−1.0
Field on axis, B_0 (T)	5	4	5	4.4	6.47	5.3	5.3
Magnetic ripple, $\delta B/B$ axis/edge (%)			/5		2.9/11.32	1.8/12.6	2/6−13
Magnet system							
Conductor	Nb_3Sn	NbTi	Nb_3Sn	Nb_3Sn	Nb_3Sn	NbTl	Nb_3Sn
Coil radius (m)	3.3	3.2	3.8/8.2/6.1[a]	4.2/8.0	3.3	5.24	4.57
Winding pack dimensions, t/w (m^2)	0.55 × 1.09	0.6 × 2.1	0.9 × 5[b]	1.14 × 2.28	1.05 × 1.05	1.4 × 1.0	1.2 × 1.0
Current density, J ($MA \cdot m^{-2}$)	50	22	10	16	19	9.8	15
Maximum field at cond., B_m (T)	8.6	9			11.6	8.7	10.4
Stored energy (GJ)		120			108	170	117
Maximum force on coil, f_m ($MN \cdot m^{-3}$)					71	60	155
Maximum equiv. stress, σ_{VM} (MPa)			110	130		80	170
Maximum shear stress, σ_{ST} (MPa)							50
Maximum strain, ϵ (%)							0.2
Blanket							
Breeding material/coolant	LiPb/He	Li_2O/H_2O		C/LiFe/He	Li_2O/H_2O		LiPb/He
Multiplier/moderator	Be	Pb			Be		Be
Blanket thickness (m)	0.25	0.4		0.5	0.2		0.2
Reflector thickness (m)	0.49						0.46
Shield thickness (m)	0.57/0.77[c]	1.06		0.7	0.8		0.3
Distance from FW to winding pack (m)	0.92	1.26	0.7/2.0[c]		1.2	>1.8	1.2
Average breeding ratio	1.05	1.76		1.05	1.1		1.10
Average blanket energy mult.	1.26	1.1		1.2	1.1		1.38

[a] Helical winding/compensation winding/TF coil
[b] Helical winding
[c] Inboard/outboard
[d] Preliminary parameters in an evolving design

descriptions of these devices are presented in Table 3-I. With the exception of MSR IIB and Heliotron H, all studies have been initiated since 1981. The MSR IIB study was completed in 1983; developments for Heliotron H continue, with special emphasis on demountable joints in the superconducting helical winding. A detailed parameter list is provided in Table 3-II.

3.1.4. Magnets

The magnets in stellarator devices provide the toroidal and poloidal fields simultaneously. They operate in a steady state mode which avoids rather sophisticated conductors. Classical stellarators and tokamaks need at least two separate coil systems.

Non-planar modular twisted coils would have a moderate construction complexity similar to yin-yang coils. It is likely that several such coils would be housed in common cryostats. Intercoil support structure between these coils inside the cryostat would be relatively easy to realize. Providing intercoil supports at locations between cryostats is difficult, but preliminary designs show this is not an insurmountable problem.

A Nb_3Sn cable-in-conduit conductor seems to be a viable solution for modular stellarator coils where the conductor is strain limited in two planes, because of the compound bending. A conductor which is insulated from its casing would allow the buildup of a weld together winding pack providing an effective mechanical support system.

Continuous coil configurations can react forces easier but will be more difficult to construct. Further, if they are to be maintainable, they must have demountable joints which have to be capable of transmitting forces without loss of performance. Such joints would make the continuous coil configuration very attractive. Providing breeding blankets with high breeding ratio only between the helical coils appears possible. Vertical maintenance of such blanket modules can be provided with a minimal impact on the remaining reactor systems.

3.1.5. Physics critical issues

Transport (heat conduction and particle losses) appears to be tolerable for reactor regime plasmas if scaled from present experiments and theory. As in the case of other steady state magnetic fusion systems with good confinement, impurities may be a critical issue.

Equilibrium β-values as obtained by modern 3D-codes are adequate for reactor operation but this needs experimental verification. Stability β may be a critical issue; however, innovative new configurations are being developed which provide a solution.

3.1.6. Technology critical issues

No fatigue problems exist for materials which operate steady state. However, first wall, blanket and shield requirements for stellarators are similar to other magnetic fusion systems. The non-axisymmetric stellarator plasmas may lead to more complex structures, but this is somewhat alleviated at moderate to large aspect ratios. High heat load and particle load components such as divertor plates and pumped limiters, as in other magnetic systems, pose a critical issue.

As in other magnetic fusion reactors with superconducting magnets, stellarator magnets pose critical issues associated with mechanical loads. Stellarators with continuous helical windings (and possibly lower forces) either need a very high reliability or demountable joints. Coils with demountable joints pose a critical issue.

Stellarators with non-planar modular coils and modular torsatrons pose critical issues regarding intercoil supports at separation areas. Preliminary designs show that this is not an insurmountable problem but experimental verification is needed.

System integration and maintenance for stellarators may have unique issues that need more detailed development.

3.1.7. General critical issue

Although a considerable fraction of the database can be taken from the tokamak development programme, more integrated system studies (point designs) are needed to guide the development of stellarators towards favourable and economically competitive power reactors.

3.2. Next generation experiments

In the stellarator field, the next generation of experiments aims at providing the database needed for demonstrating the reactor properties of the concept. In doing so, one can concentrate on those key issues where the stellarator concept differs from the tokamak and assume certain features demonstrated on tokamaks to be valid for stellarators as well. This might be particularly true for the interaction between fusion generated alpha particles and the underlying bulk plasma. If this hypothesis holds it would be possible to design the next generation stellarator experiments to confine reactor-grade plasmas but without using tritium. Some aspects of alpha particle physics (e.g. stability, plasma–wall interaction, helium pumping) could be simulated by high energy helium injection. Examples along this line are Heliotron F, an upgrade version of ATF, and the presently developed Advanced Stellarator Wendelstein VII-X. All of them are under continuous re-evaluation to include the latest achievements of configuration optimization towards high stability beta. Superconducting coil systems are envisaged for most of them.

If the above hypothesis does not turn out to be valid, i.e. if there were serious doubts that tokamak alpha particle physics will be applicable to stellarators, then a stellarator fusion burner experiment will have to be built. Nevertheless, even in this case, the developed technologies, e.g. blankets, materials, and tritium technology, etc. can still be adopted from the tokamak technology programme. It is thus possible to reduce the total operating time of such stellarator burner experiments to that needed for studying the reactor physics, and to dispense with the need for fuel breeding. An early example of a scoping study along this line of thought is the Wendelstein Advanced Stellarator Burner ASB, the major parameters of which are given in Table 3-III.

3.3. Experimental situation

Considerable progress has been made in the experimental field with the major contributions coming from Heliotron E (Kyoto, Japan), L-2 (Moscow, USSR), Uragan-II (Kharkov, USSR) and Wendelstein VII-A (Garching, FRG). The principle of the modular divertor was demonstrated in the small scale experiment IMS (University of Wisconsin, USA).

3.3.1. New modes of operation

Earlier stellarator experiments used to start up from plasmas generated by Ohmic heating and were thus not free of tokamak effects. The availability of powerful ECF gyrotrons opened the possibility to create high temperature plasmas directly from the gas phase without any externally applied loop voltage. With high enough gyrotron frequency (50–70 GHz) also the plasma

TABLE 3-III. EXAMPLE OF A STELLARATOR BURNER EXPERIMENT (ASB)

	R (m)	a (m)	B_0 (T)	B_m (T)	β (%)	ι_0	P_{fus} (MW)	P_{heat} (MW)
ASB	15.2	0.9	7.0	12.6	2.5	0.53	400	30

TABLE 3-IV. MAJOR STELLARATOR DEVICES AVAILABLE NOW OR SOON

	HELIOTRON E	URAGAN 3	ATF	L-2M	W-VII AS	URAGAN 2 M
Laboratory, country	Kyoto, Japan	Kharkov, USSR	ORNL, USA	Lebedev, USSR	Garching, FRG	Kharkov, USSR
Type	Heliotron	Torsatron	Torsatron	Torsatron	Advanced stellarator	Torsatron with tor. field
Status of operation	Operating	Operating	1987	≅1988	1987	≅1989
Major radius, R (m)	2.2	1.0	2.1	1.12	2.06	1.7
Minor radius, ⟨a⟩ (m)	0.2	≤0.1	0.3	0.2	0.2	0.2
Multipolarity/field periods, ℓ/m	2/19	3/9	2/12	2/8	(1, 2, 3)/5	2/4
Magnetic field, B (T)	2.0	≅1.0	1 (2)	1.5(2.5)	3.0	2.0
Rotational transform, ι_0/ι_a	0.6/2.5	0.2/0.25	0.34/0.9	0.25/0.8	0.38/0.39	0.57/0.73
Heating power (MW)	6	1.5	3–4	3.5	5–6	5
Magnetic well depth −V″ (%) Magnetic hill +V″ (%)	±30	−5	+6	−7	−2	−1

density was high enough for continuing plasma heating by NBI or ICRH. In this way pure stellarator operation could be established in plasmas with densities and temperatures exceeding 10^{20} m^{-3} and 1 keV, respectively. Average beta values were 2% at 1.0 T (Heliotron E) and 0.45% at 3.2 T (Wendelstein VII-A) with no indication of instabilities in the latter case and no degradation of confinement by the applied heating.

A very important effect was discovered in these experiments: The ambipolarity condition for the particle loss fluxes leads to the generation of radial electrical fields which drastically reduce the transport through the ion channel as compared to neoclassical models which have neglected this effect. Reduction factors of up to two orders of magnitude were found. If these electric fields are properly considered, both ions and electrons seem to follow neoclassical predictions in the plasma core. Only where plasma density and temperature are low is electron heat conduction measured to be higher.

With these electric fields, the ion loss channel is of low importance in stellarators. The electrons, on the other hand, have a poloidal component of their velocity much higher than the $\vec{E} \times \vec{B}$ rotation and are thus affected by the radial electric field only to a negligible extent. Therefore, the main power loss for stellarator fusion reactors is expected through the electron channel and will show the high temperature dependence predicted by neoclassical effects. The values expected will reach the values needed for the output of the fusion alpha power.

One of the critical issues on which the activity is being concentrated is to operate under steady state conditions at a tolerable level of impurities. This is a problem with all concepts showing good confinement properties. Thus, methods under development in the tokamak field might work in stellarators as well.

3.3.2. Major stellarator facilities available now or coming on-line soon

The experimental devices available now or very soon will address the above problems (preferably under steady state conditions). They are also designed to bring the plasma parameters and confinement properties closer to reactor requirements. These devices and their key parameters are listed in Table 3-IV.

None of these devices is equipped with hardware for a divertor, because it would be highly desirable to achieve the necessary plasma boundary conditions with pumped limiters as they are being developed by the tokamak programme. They would make most efficient use of the magnetic field volume and perhaps work even better under stellarator conditions than under tokamak conditions since the stellarator confining fields are produced entirely by an external coil current and are thus rather stiff in space. If pumped limiters do not turn out to be feasible, all stellarator concepts can be made to possess an internal separatrix and can then be equipped with an open divertor of the INTOR type. This is particularly easy for Uragan III. To go even further and to carry flux bundles through the coil system to outer divertor plates is a possibility proven by field line studies but has to be introduced as an additional condition when defining the magnetic field topology.

3.4. Theory and code development

In this field the situation is best characterized by the availability of much more powerful computers. Their high computational speed and memory size have increased the possibility for more accurate three-dimensional codes. Even forecasts on the stability limits of beta are becoming possible with such means and allow the configurations to be optimized. in this respect. An analysis of the results of these computations indicates that a helical shape of the magnetic axis tends to improve stability (heliacs). There is hope, however, that proper excursions of the magnetic axis only in the equatorial plane would lead to similar improvements and allow for a somewhat simpler magnetic system. Rapid progress is expected during the coming years. Similar progress has been made with respect to transport properties including electric field effects. This effect provides the potential to reduce reactor size and power output.

3.5. Conclusions

In recent years, progress in the field of stellarators in experiments, theory and reactor system studies has exhibited the potential for favourable reactor properties and produced further important elements of the basis needed for developing steady state stellarator power reactors.

In some fields, however, further improvements are necessary to arrive at plasmas close to reactor requirements. More integrated system studies are also necessary to guide the development towards favourable and economically competitive stellarator power stations.

A considerable fraction of the database can be taken from the tokamak development programme. This includes such items as first wall and blanket concepts, tritium fuel circuit and material development. Compared to the pulsed systems, the requirements on materials are eased in steady state stellarators.

4. MAGNET SYSTEMS
(A. Kostenko)

The magnet system of a fusion reactor is a crucial component and a main cost driver of the whole machine. In all previous conceptual reactor designs, the magnet system is considered to be a semi-permanent component, and therefore it must be very reliable and safe.

4.1. Status and progress since 1981

Since 1981 several magnet designs for a next step machine have been performed (INTOR, NET, FER, OTR). In all these studies, only superconducting coils are considered for the TF and PF coil systems. In all studies, Nb_3Sn is preferred for the TF coils and NBTi for the PF coils, but in some recent studies Nb_3Sn is proposed for the PF coils, too. Forced flow cooling at the 4 K temperature level is always first choice.

Two tokamaks with superconducting TF magnets are actually under construction: T-15 and TORE Supra. T-15 uses a forced flow cooled Nb_3Sn conductor at 4.5 K at a maximum field of about 9 T, and TORE Supra utilizes a NbTi He II bath cooled conductor at 9 T. The operational experience of these experiments will be of great importance for later fusion devices. All coils are fabricated and nearly all of them are already tested as pairs of solenoids. They performed very well in accordance with the predictions. Recently two full scale prototype coils for T-15 have been tested successfully in a two coil arrangement. Eight coils (out of 24) are delivered to the Kurchatov Institute for testing before being assembled into the machine.

Several programmes exist all over the world to support the development of toroidal field coils. The most important programme is the international LCT programme in the frame of IEA, where six coils are now being tested in a toroidal arrangement at the Oak Ridge National Laboratory. The coils have been developed by the project partners US/DOE (acting through ORNL), Euratom (acting through KfK), JAERI, and Switzerland (acting through SIN), together with individual industrial contractors. Several conductor and coil designs and cooling schemes are being tested at different operational conditions. Before delivery to Oak Ridge, the Japanese and the Euratom coils were successfully tested in domestic single coil tests. A partial array test was performed earlier at Oak Ridge. The completed array with all six coils was cooled down first in January 1986. In the first test period all coils were energized successfully in single coil mode, each to the rated current. In addition, stability tests have been performed.

The results of the tests so far are very encouraging for a successful six coil test at 8 T planned for the very near future. It can be concluded that the technology of toroidal field superconducting magnets is reaching maturity for tokamak reactors in time.

In addition, several conductor and magnet test devices have become operational since 1981.

Poloidal field coils development programmes are not so advanced as the TF coil programmes. Such programmes have existed for several years in Japan and the USA. Recently, a Euratom PF coil development programme was initiated with the goal of constructing a PF coil for TORE Supra. In the USSR, the work on PF coil development has started recently.

The results of the construction and test of the MFTF-B magnet system at the LLNL, USA, are encouraging for the development of large mirror fusion reactor coils. This system consists of several types of magnets: low field solenoids of about 7 m in diameter, C-shaped and/or yin yang-type coils of the characteristic dimension of about 8 m (the weight of one yin yang is about 370 t), and high field coils of 12 T, using Nb_3Sn conductors. The successful test of this magnet system demonstrates that large coupled superconducting magnet systems can be built and operated reliably.

Several designs of tokamaks using sophisticated normally conducting magnets have been performed since 1981. These are mostly for compact tokamaks, aiming at reaching ignition. Modular, non-planar copper coils for stellarator experiments (about 1.5 m diameter) have been built and tested recently. However, normally conducting magnets cannot, in general, be envisaged for commercial tokamak reactors, for economic reasons.

TABLE 4-I. MAIN MAGNETIC SYSTEM PARAMETERS

Parameters	NET-DN	FER	OTR[b]
TFC system			
Number of coils	16	12	12
Maximum field (T)	10.4	12	11.6
Conductor current (kA)	16	30	25
Current density[a] ($kA \cdot cm^{-2}$)	2.0	3.0	2.5
Stored energy (GJ)	25	~23	25
Conductor	Nb_3Sn	Nb_3Sn	Nb_3Sn
Cooling mode	forced flow	forced flow	forced flow
Bore size (m)	6 X 11	6.8 X 8.7	7.4 X 11.4
PFC system			
Current (kA)	40	50	50
Current density[a] ($kA \cdot cm^{-2}$)		2.5	2.0
Stored energy (GJ)	4.2	7	10
Conductor	NbTi	Nb_3Sn/NbTi	Nb_3Sn/NbTi

[a] Current density averaged over the winding pack.
[b] Preliminary data.

4.2. Comparison of magnets for next step machines

Three design studies for next step machines were presented during this meeting: NET, FER, OTR. Table 4-I contains the main parameters of the TF and PF coil systems. In all studies only superconducting magnets with forced flow cooling conductors were proposed. For the TF coils, binary and ternary Nb_3Sn conductors are specified. In earlier studies, e.g. INTOR, NbTi conductors were the only option for the PF coils. In the present studies FER and OTR, Nb_3Sn is also proposed for the use in PF coils. Two machines (NET, OTR) use a vault structure and FER uses a bucking cylinder to support the centring forces acting on the TF coils. The shield is designed to limit the radiation dose at the magnet winding pack to 5×10^6 Gy (1 Gy = 100 rad), a dose permissible for fibreglass reinforced epoxy resins.

4.3. Issues discussed during the workshop

4.3.1. General TF-coil design aspects

The toroidal field coil system has to be reliable, compact and safe. The worldwide studies since 1981 for the next step machines consider superconducting coils only with Nb_3Sn superconductors for field levels around 12 T. In most studies, forced flow, internally cooled superconductors are proposed because they allow a higher voltage versus ground during quench protection discharge and a more rigid structure which is better suited to withstand the electromagnetic loads.

These conductors also allow higher current densities and higher mechanical forces within the same radial thickness as copper coils. Copper coils in a fusion reactor are only comparable to superconducting ones if they are nearer to the plasma, leading to radiation doses which are higher by several orders of magnitude. There, they require advanced insulation materials, such as ceramics, leading to a much more difficult coil design (e.g. provision of necessary rigidity to transfer mechanical loads to the support structures). In addition, the power consumption in copper coils is appreciably higher than in superconducting coil systems.

There exist some studies on hybrid toroidal field magnet configurations for next step machines. Hybrid magnets permit the field to be increased on the plasma axis by inserting copper coils inside the superconducting coils, replacing a part of the shield. However, integrating the coils into the shield and supporting the electromagnetic forces seems to be a problem and should be studied in more detail. For commercial reactors, only superconducting coil systems are envisaged, for economic reasons.

Two solutions to support the centring forces of the TF coils were proposed, one with a bucking post and the other one wedging the coils together (vault structure). The bucking cylinder is space consuming. The wedging approach requires very small tolerances during manufacture and assembly. Further studies on advantages and disadvantages are needed.

The most suitable insulation system is the one that implies a rigid coil winding in order to provide good force transmission from the turns to the support structure. From this point of view, vacuum impregnation with epoxy resin is the appropriate fabrication technique. It is also necessary to develop a resin system capable of withstanding radiation doses up to 5×10^7 Gy, with good mechanical properties and suitable to monolithic coil fabrication.

4.3.2. Poloidal field coil design

The experimental poloidal field coil developments have been based on NbTi conductors, so far. But in recent studies (OTR, FER) it has been proposed to use Nb_3Sn for poloidal field coils, too. This is highly desirable, because the central solenoids would then provide a higher flux swing in the machine and the large ring coils would have a higher safety margin in terms of temperature. Also, longer cooling channels would be allowed because of the higher temperature margin, leading to simpler coil construction with fewer conductor joints.

The main problem for the Nb_3Sn coils is the strain sensitivity of the conductor, limiting the minimum bending radius. The small bending radius (of the order of 1 m) for the central solenoids is, therefore, a difficult constraint for high current Nb_3Sn conductors (about 50 kA).

4.3.3. Insulation systems

The insulation system in a fusion magnet seems to be the weakest component, according to today's knowledge of irradiation behaviour. It is agreed that better insulation systems are required in this respect. Polyimide insulators seem to be promising, but there are disadvantages due to the high thermal contraction and the processing during vacuum impregnation.

With respect to ceramic insulation, no stringent need for it has been seen so far in the field of superconducting magnets, but it is necessary for copper coils in highly irradiated areas of fusion devices (e.g. active vertical position control coils).

4.4. Future prospects for advanced designs

At the recent IAEA INTOR related meeting on tokamak concept innovations, also several aspects of advanced magnet designs were discussed; they are also worth being discussed in the frame of this meeting.

Major impact is expected to come from the areas of material development, both advanced superconductors and structural materials, and from advanced cooling schemes (temperatures below 4.2 K, even superfluid He).

The development of advanced superconducting materials such as Nb_3Al, NbN and ternary Nb_3Sn, has been very successful in the last five years. Basic strands of Nb_3Al and ternary Nb_3Sn are available on a laboratory scale. They are being used in test facilities to prove their performance in small insert magnets. For magnetic fields higher than 12 T, where ternary Nb_3Sn conductors are very promising, there is some concern on their radiation resistivity. Compared with binary Nb_3Sn they seem to be, at least, a factor of about five more radiation sensitive. This has to be confirmed experimentally in the near future.

As far as the use of advanced superconducting materials in large fusion magnets is concerned, it is necessary beforehand, to prove their performance in medium sized magnets. The possible major impacts of these prospects on the magnet design data would be higher field levels and/or higher current densities for the winding pack.

4.5. Conclusions and recommendations

The superconducting magnet technology has achieved remarkable progress since 1981. It can be expected that it reaches maturity for next generation devices and for reactors in time, without insurmountable problems.

The Nb_3Sn technology for poloidal field coils in tokamaks is not yet fully developed, but since it is a highly desirable feature, research efforts to establish this technology are strongly recommended.

The development of radiation resistant insulation systems is strongly recommended.

Ceramic insulation need not be taken into account for superconducting magnets, but should be applied to Cu magnets in highly irradiated areas of fusion devices.

The development of advanced superconductors, as for example Nb_3Al and ternary Nb_3Sn, should be pushed to industrial maturity.

Irradiation experiments on advanced superconductors are recommended, especially at low temperatures (4 K).

The work on advanced high strength structural steels should be continued.

Normally conducting magnets are, in general, not envisaged for commercial tokamak fusion reactors, for reasons of power consumption.

5. INERTIAL CONFINEMENT FUSION
(W. Hogan)

5.1. Introduction

Participation in the ICF workshop included representatives from the USSR, USA, and Poland. While everyone was aware of the significant ICF efforts in other countries (e.g. Japan, France, Federal Republic of Germany, etc.), we did not presume to represent their latest accomplishments. Thus, these discussions, conclusions, and recommendations represent only those of the participants. Tables of major ICF facilities were assembled separately by the IAEA and are being published in the 1986 Special Supplement of the Nuclear Fusion Journal entitled "World Survey of Activities in Controlled Fusion Research".

5.2. Progress in the physics of laser-target interactions, target design and implosion experiments

Since the last workshop in Tokyo in 1981, there has been much progress in understanding the physical phenomena of laser driven inertial fusion. Results from LLNL (Nova) and Lebedev Institute (Delphin) include:

(i) Increased absorption of laser light as wavelength decreases below 1 μm – 50% to 95%.
(ii) Decreased production of hot electrons as wavelength is decreased – 30% to <2%.

(iii) Increased conversion of absorbed light to X-rays for hohlraum targets – 50% to 90% (for intensities of $\sim 10^{14}$ $W \cdot cm^{-2}$).

(iv) Acceleration of a cold shell before compression to a velocity of 200 $km \cdot s^{-1}$ with a hydrodynamic efficiency of 10%. Possibility of stable compression of high aspect ratio ($\sim$100) shells if one has good spherical illumination and very good target fabrication.

Achievement of high ion temperature ($\sim$10 keV) with some fuel compression in direct drive gas targets giving a neutron yield of 10^{13}.

Achievement of high compression ($\rho \sim 10-13$ $g \cdot cm^{-3}$) with moderate ion temperature (1.7 keV) in hohlraum targets, leading to a Lawson value of $n\tau > 10^{14}$ $s \cdot cm^{-3}$ at this temperature.

The associated neutron yield (10^{10}) was within a factor of two of the predicted value.

Construction of reactor size liquid DT targets in which the DT shell is held in place with a low density foam.

Mathematical codes for the physical processes of light absorption and conversion and of target compression and burn have shown (see Fig. 5.1) that it is possible to achieve $G \cong 300$ for optimistic assumptions with simple targets. More advanced targets will give even higher gain (shown in Fig. 5.1). It is reasonable to take these values as input data for the design of ICF reactors. Hybrid reactors (see below) will need gains of 50 to 100 to be economical, while pure fusion reactors will need gains of 100 to 500 (depending on reactor design) to be economical.

The most important problems which remain unsolved by experiments are the compression of the fuel up to a density of 200 $g \cdot cm^{-3}$, the achievement of moderate gain ($G \approx 1$), which can be achieved with several implosion schemes, and the achievement of high gain ($G \approx 100$), which will probably require wavelengths between 0.26 μm and 0.53 μm and temporally shaped pulses. Some laser-matter interaction physics must also be solved to achieve this goal (e.g. stability and fuel mixing during compression, proper selection of light intensity, illumination geometry, etc.). Finally, solid state laser technology is available to scale up to the 1 to 10 MJ required for a high gain test facility, but the costs of such a facility must be lowered.

In general, it seems now that the concept of laser driven inertial confinement fusion has a very good physical background for the next step: technical and engineering design of a high gain test facility with reactor target performance optimization, investigation of the effects of target output on materials, and performance of various physical experiments which achieve extreme plasma parameters with high (>100 MJ) thermonuclear yields.

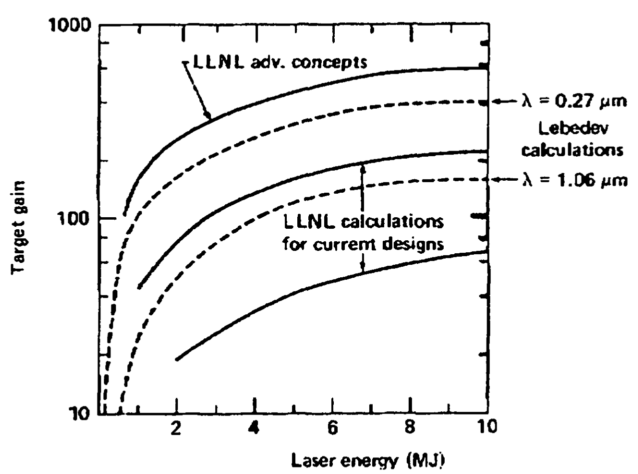

FIG. 5.1. Target gain versus laser energy as calculated by the Lebedev Institute on the DIANA code and by LLNL on the LASNEX code.

5.3. New ICF reactor concepts

Since the last workshop some progress has been achieved in already established concepts, and three entirely new concepts were proposed. Emphasis in the USSR has been on hybrid reactors for reasons of economy and relaxed physics requirements. The USA has worked on both pure fusion and hybrid types.

The LLNL has introduced a concept called Cascade. This concept employs a rotating biconical ceramic vessel in which centrifugal forces move a thick ($\sim$1 m) ceramic granule blanket through the reactor cavity. This C, BeO, $LiAlO_2$ blanket combines fusion energy conversion to thermal energy, tritium breeding, thermal energy transport, and protection of reactor structure from neutrons, X-rays and debris. The concept includes a high temperature He Brayton cycle for high thermodynamic cycle efficiency ($\sim$55%) and a low activation SiC/Al composite structure. Fire hazards, induced activity, BHP, tritium inventory,

TABLE 5-I. CASCADE REACTOR CHARACTERISTICS

Driver (laser or heavy ion driver)	
Efficiency	– 10% or 20%
Pulse energy	– 1.5 to 3 MJ
Pulse rate	– 5 Hz
Target yield – 300 MJ	
Chamber characteristics	
Max. radius	– 5 m
Cone angle	– 35°
Rotation speed	– 50 rpm
Material	– SiC tiles with SiC/Al composite tendons
Blanket (inside reactor wall)	
Pyrolytic C	– 10 to 30 mm
BeO	– 90 mm
$LiAlO_2$	– 900 mm
Power (MW)	
Fusion	– 1500
Thermal	– 1670
Net electric	– 815

TABLE 5-II. CHARACTERISTICS OF THE TIGER HYBRID ICF REACTOR

Laser energy – 2 MJ	
Target gain G = 60	
Reactor chamber – R = 3 m cylinder	
Liquid shield (thin Li wetted wall)	
Blanket – $^{238}U + 0.35\%\ ^{235}U$	
M = 10 (fusion energy multiplier)	
Pulse rate ~0.8 Hz	
Fusion hybrid station	
Thermal power	– 1 GW_t
Electric power	– 0.26 GW(e)
Pu production	– 700 kg per year
Associated fission stations' electric power – 1.7 GW(e)	

TABLE 5-III. CHARACTERISTICS OF MATERIALS PRODUCTION REACTOR (MPR)

Driver	
Efficiency	– 3%
Pulse energy	– 3 MJ
Pulse rate	– 2 Hz
Target yield	– 400 MJ
Reaction chamber	
Radius (cylinder)	– 3.3 m
Height	– 8.0 m
Graphite reflector	
Blanket (inside reactor wall)	
20% Li curtain (R = 1.9 m, ΔR = 10 cm)	
48% Be columns w/Li coolant channels	
30% Th pins in some channels or Th wire continuously fed through channels 2% Fe	
Fuel production – 3300 kg ^{233}U per year	
Power (MW)	
Fusion reactor	
Fusion	– 800
Thermal	– 2000
Associated fission reactors	– 26 000

and afterheat are low, because of extensive use of low activation, ceramic materials. Economics studies indicate that the cost of electricity from such a plant can be competitive with or lower than future coal and fission plants. Table 5-I contains the major parameters of this reactor.

Progress in LANL's wetted wall cavity concept has been primarily in the area of detailed static and dynamic structural integrity analysis and wall lifetime (radiation damage) projections. The integrity analysis included structural design as well as verification of Li adherence to the first wall. Also, the incorporation of a lithium boiler blanket concept has been proposed in order to achieve higher thermodynamic efficiencies.

Further analysis of the magnetically protected dry wall has more firmly established the ion clearing of the cavity.

Generic studies of cavity clearing in Li and other vapour atmospheres have established the importance of obtaining experimental data to establish the condensation times and therefore the repetition rates achievable.

Recent work has also been done in the USSR and the USA on hybrid ICF reactors. From the point of view of economy and simplicity of technical realization, there are advantages to the ICF reactor that simultaneously produces electrical power and fuel for conventional nuclear power stations. The USSR work emphasizes fast fission blanket designs while the USA work has emphasized suppressed fission designs, although both have done some studies of the other type. Characteristics of the TIGER concept proposed by the Lebedev Institute are shown in Table 5-II. The suppressed fission Materials Production Reactor (MPR), proposed by LLNL, is shown in Table 5-III. The material production characteristics of both reactors are discussed in more depth in the Hybrid Workshop discussion.

5.4. Status of ICF drivers

The principal ICF drivers under consideration can be divided into two categories – those primarily intended for a single pulse, high gain test facility and those intended for high pulse rate reactor use. In the former category there are, at present, Nd:glass lasers and light ion diodes while in the latter are the heavy ion accelerators, KrF gas laser, free electron laser (FEL), and high average power solid state lasers. Since the last meeting two driver technologies have been abandoned (electron beams and CO_2 laser), and a new one has been added (compact torus accelerator).

The Nd:glass laser has become the standard laboratory instrument worldwide for doing target design research because of its high state of development, high peak power, high energy, efficient frequency conversion, pulse shaping, and general flexibility. At present, the Nova facility at LLNL is the largest such facility, producing 100 to 150 MJ at 1.06 μm, 50 to 80 MJ at 0.53 μm and 40 to 70 MJ at 0.35 μm. The technology is scalable to the 5 to 10 MJ needed for high gain. The only remaining question is cost and research is focused on lowering this to a reasonable value.

The light ion effort in the USA has chosen to emphasize the Li ion. It has achieved adequate focusing (~20 mrad) at low voltage and current and is now concentrating on raising the diode voltage to 30 MV and the issues of focusing the necessary high currents onto a sufficiently small spot so that the ICF targets can be tested at 1 to 2 MJ of incident energy – perhaps by 1988.

Work is proceeding on KrF lasers as a driver development programme. Small scale experiments have shown ratios of laser light to electron beam energy deposited of over 12%. A 10 KJ amplifier has been built in the USA and work is proceeding on the optical stacking necessary to compress the pulse to a few nanoseconds.

A recent US study of heavy ion induction accelerators as drivers for ICF reactors has concluded that the drivers can be much less expensive than previously assumed. This has come about through many improvements, including the realization that beams of charge state +3 can be used. Suitable sources are under development. As a consequence, reactors using this driver may be economically competitive with future coal and fission plants in sizes of 1-2 GW(e) and competitive with magnetic fusion at even smaller sizes.

New work on advanced solid state lasers suggests they should be considered as possible reactor drivers as well as for a single pulse facility. The concept involves use of crystal lasing media co-doped with Cr and Nd in a gas cooled disk arrangement to extract the heat necessary for high pulse rate and efficiency. Experiments are underway to measure optical quality of gas cooled disks and on rapid growth of large crystals to cut costs. In addressing the efficiency question, a 30 cm rectangular disk amplifier has been built with stored energy of over 7% and small scale experiments show that Cr co-doping can increase efficiency by a factor of two.

A large free electron laser (FEL) programme has begun at LLNL for the Defense Program. This will provide an opportunity to consider the FEL as an ICF driver. Recent measurements in the microwave region (~1 cm) have shown an electron beam to laser light efficiency of 35% with a tapered wiggler. In the next few years, experiments will proceed toward wavelengths shorter than one micron.

Focusing and target interaction difficulties with electron drivers and CO_2 lasers have caused the abandonment of these efforts in the USA and the USSR. Finally, in the USA, at LLNL, a small new programme has begun to see if a compact, heavy ion plasma torus about 30 cm in diameter, generated with a Marshal gun, can be accelerated and compressed to the energy and diameter necessary to drive an ICF target.

5.5. Next development steps

5.5.1. High gain test facility

It appears that the understanding of target physics and driver technology is sufficient to define, with small uncertainty, the conditions necessary to achieve high gain. For a laser driver, these requirements are:

Pulse energy	– 3 to 10 MJ
Peak power	– 500 to 1500 TW
Pulse duration	– 0.1–100 ns
Pulse shape	– flexible w/dynamic range of 100:1
Pulse rate	– few per day
Wavelength	– flexible, 0.25 to 0.53 μm
Focusability and pointability	– ~1 mm spot at 5 to 30 m distance, flexible illumination geometry
Durability	– 10^3–10^4 shot life
Efficiency	– little importance (except as affects cost)
Cost	– US \$25 to US \$50/J

Considering the state of driver development, it is most likely that the solid state laser will be the only driver that will be able to satisfy these desirable characteristics within the next three years.

Once such a high gain facility is built, ICF progress will be more rapid. Of course, the first experiments on such a facility would be directed toward achieving high gain, understanding how target gain varies with driver energy for various target types, and trying to produce designs which will achieve high gain at the lowest driver energies. However, in addition to those fusion capsule physics experiments, many other experiments can be done with this facility. Target yields of up to 1000 MJ can be expected with associated neutron yields of 10^{20} and accompanying X-ray and ion energy. As an example of physics experiments that can be done on such a facility, Lebedev has proposed neutrino studies using such reactions as $^{11}B(n, \gamma)\ ^{12}B\bar{\nu}$, $^{12}C(p, n)\ ^{12}N\nu$, and $^{27}Al(n, 2n)$, $^{26}Al\nu$. Also, studies of the interaction of the output of an ICF target with possible reactor materials, optical protection schemes and walls can be performed. The workshop members consider it of great importance to move forward with proposals for the design of such a single pulse, high gain experimental facility.

5.5.2. High pulse rate technology test reactor

Between the single shot, high gain physics facility and the fully integrated demonstration power plants there is a need for a large volume, high neutron flux and high neutron fluence device. There has been little ICF activity in this area over the past decade in spite of the design of over 30 magnetic fusion facilities.

Two possibilities for such a test facility should be considered: (A) laser or heavy ion beam, and (B) light ion drivers. Each would have essentially the same neutron temporal and spectral characteristics, but there would be important differences in technology needs for such a device. One such study, Sirius-M, has been described at this conference and it relies on symmetrically illuminated, low-gain targets. The light ion approach can be represented by designs similar to Libra none of which were represented here.

Any ICF facility that would be used for these purposes must generate at least 1 to 2 $MW \cdot m^{-2}$ neutron wall loadings and have at least 50 to 100 liters of high flux volume for materials testing. There should be at least that much volume for blanket testing modules. If the facility can be built as a low power device (less than 100 MW), then it should be possible to purchase the tritium necessary for operation with up to 50% availability. Neutron fluences of at least 5 to 10 $MW \cdot a \cdot m^{-2}$ will be necessary for materials screening and qualification. Lower fluences (i.e. 1 to 3 $MW \cdot a \cdot m^{-2}$) could be useful for blanket testing and qualification.

The major issues that need to be resolved before such an ICF facility could be constructed include: (1) the operation of a high pulse rate (~10 Hz), high energy (~1–2 MJ), short wavelength (<1 micron) laser, (2) target design, testing and mass fabrication schemes, (3) injection of target and aiming of mirrors on a repetitive, reliable basis, and (4) testing of cavity protection and clearing schemes. In the case of a light ion facility, there are the additional issues of development of a high pulse rate light ion diode, protection of the diode from the fusion pulse effects, and propagation of multiple ion beams through plasma channels to the target. All the above issues could be the subject of individual test facilities in the 1990s, and an integrated ICF technology test facility could be built shortly after the turn of the century.

5.5.3. Reactor technology needs

Independent of the reactor type, it is possible to define many problems that can and should be solved now. Several examples are given here.

The survivability and protection of large optics under the severe loading of laser beams, X-ray radiation, debris and neutron flux.

For ICF reactors with liquid wall (thin or thick), it is necessary to investigate experimentally the shielding layer under radiation, shock and neutron loading and to achieve acceptable conditions for the laser beam focusing onto the target, taking into account the possibility of laser induced breakdown in the liquid shield vapour environment.

It is important to obtain experimental data on the problem of vaporization and condensation of materials between fusion pulses (e.g. Li, C, Ta). Models are being developed and improved but the most important need at the present time is for data to test the models.

The various drivers have different technological development needs:

Solid state lasers

lower cost
high average power with good optical quality (high pulse rate)
higher efficiency (>10%)

KrF

lower cost
obtaining high peak power (pulse compression)
higher system efficiency (>10%)

Light ions

focusing
pulse shaping
multibeam channel transport
high pulse rate diode

Heavy ions

stable transport in cavity atmosphere
beam neutralization
lower cost

FEL

focusability and pointing
efficiency at $<1.0\ \mu$m

Compact torus

basic feasibility
transport to target

Target technology developments are also required, independent of reactor and driver type:

(1) Development of inexpensive means of mass target production (US \$0.1–1/target)
(2) Fast target insertion with high accuracy tracking (~1 mm) in focusing region.

5.6. Conclusions and recommendations

The workshop concluded that much progress has been made in the last five years, and that it is now time to develop designs for a low pulse rate, high gain test facility (HGTF). We conclude that the driver requirements are well defined, and it is important in the next three years to define the experiments that would be done on such a facility and to develop flexible designs that would be capable of carrying out those experiments. The strongest recommendation we have is that in three or four years there should be an international conference at which the designs for such a facility and for proposed experiments on it could be presented and discussed. Perhaps the IAEA would agree to support such an international meeting.

The second conclusion is that there are a number of ICF reactor technology issues that are common to many reactor designs and drivers. Experiments should proceed on these issues first. These include:

(1) Vaporization and condensation phenomena for a number of materials (Li, C, BeO, SiC, Ta) associated with chamber clearing time
(2) Survivability and protection of final optics (damage mechanisms and levels, etc.)
(3) Isochoric heating of liquid metals
(4) Damaging effects of pulsed radiation on materials
(5) Target manufacture and injection at high rate.

We feel that it is time to emphasize reactor technology programmes that address these issues. Perhaps experimental results could be presented at the above proposed meeting.

6. ALTERNATIVE FUSION CONCEPTS (AFCs)
(R. Miller)

6.1. Introduction and status

The 1981 Tokyo workshop considered a relatively large and diverse group of AFCs, representing a prudent programme of backup and competition for the mainline tokamak approach. Certain AFCs may still offer distinct technical and/or economic advantages as eventual commercial reactors or even as nearer term ignition test devices or engineering test reactors although the latter applications have received only little attention. Several trends in AFC research since 1981 should be mentioned. The renewed interest in, and activity related to, the stellarator/torsatron/heliotron (S/T/H) class of devices (characterized by no net internal plasma currents) has grown to the point that a separate workshop is appropriate. On the other hand, a large number of AFC approaches has fallen into inactivity since 1981. The remaining AFC category now emphasizes a related class of poloidal field dominated (PFD) systems including the compact toroids (CTs) as well as reversed field pinches (RFPs). CT plasmoids without internal toroidal field are classified as field reversed configurations (FRCs), whereas plasmoids containing internal toroidal fields are termed spheromaks. CTs are distinguished from RFPs by the absence of linked coils or

TABLE 6-I. SUMMARY OF KEY REACTOR PARAMETERS FOR ALTERNATIVE FUSION CONCEPTS (AFCs)

Concept	FRC		CSR	CRFPR	
Group	Efremov (USSR)		Los Alamos (USA)	Los Alamos (USA)	
Parameter					
Fuel	D–T	D–^{3}He	D–T	D–T	
Minor radius (m)	~2.5	~2.5		0.71	1.42
Length/major radius (m)			1.9	3.9	7.6
Plasma volume (m^3)	350	350	105	38	303
Density (10^{20} m^{-3})	3	5	2.3	6.6	2.3
Temperature (keV)	10	50	20	10	10
Average beta	0.8–0.9	0.8–0.9	0.1	0.13	0.13
Magnetic field at plasma (T)	1.6	5.0	5.0	5.2	3.0
Magnetic field at coil (T)	TBD[a]	TBD	2.6	~3.5	~2.7
Plasma current (MA)	NA[b]	NA	47	18.4	21.6
Stored field energy (GJ)	TBD	TBD	1.5	1.7	~5.0
Neutron current at first wall ($MW\cdot m^{-2}$)	6	5 (X-rays) $\leqslant$1 (neutrons)	19.8	19.0	5.0
Thermal power (MW(th))	1.200	1.500	3.416	3.472	3.609
Net power (MW(e))	TBD	TBD	1.000	1.000	1.000
Recirculating power fraction	TBD	TBD	0.16	0.20	0.22
Net plant efficiency	TBD	TBD	0.30	0.29	0.28
Mass power density (kW(e)/tonne)	TBD	TBD	1.200	900	500

[a] To be determined.
[b] Not applicable.

other constructive elements in the centre of the torus. It may be mentioned that the low aspect ratio spherical torus (ST) variant of the tokamak is another PFD system. This class of systems is characterized by relatively high beta values (0.15), making compact, high power density (resistive coil) applications or advanced fuel (D-^{3}He) operation particularly interesting topics. Reduced cost as well as fabrication/maintenance advantages are also projected. A rather broad range of experiments, including significant new constructions, in a number of countries exists to support this class of PFD systems.

6.2. Summary of concepts

Table 6-I summarizes key AFC commercial reactor parameters for three PFD systems presented at this workshop, including the quasi-stationary FRC reactor (in contrast to the translating FRC, CTOR, presented at the 1981 workshop) in both D–T and D–^{3}He versions, the Compact Spheromak Reactor (CSR), and the Compact Reversed Field Pinch Reactor (CRFPR, introduced in a preliminary form at the 1981 workshop) for two neutron first wall loadings. The following sections describe each of these approaches.

6.2.1. Field reversed configurations (FRCs)

The field reversed configuration is characterized by no internal toroidal field, a prolate (cigar-like) plasmoid shape, and beta values in excess of 0.8. Experiments in the USA and the USSR have explored transport and translation issues, with transport being understood as due to anomalous lower hybrid drift (LHD) processes. Such transport, although in a more pessimistic vein than in 1981, is considered adequate for reactor conditions. Experiments proposed or under construction aim at exploring the regime of higher $S = x_s r_w/\sqrt{2}\rho_{i0}$, noting that $\tau_E \propto S^2$ and $\tau_N \propto S^4$. The parameter $x_s = r_s/r_w$ is the ratio of the plasmoid separatrix radius to the wall radius and ρ_{i0} is the ion Larmor radius calculated in the field at the separatrix. A Los Alamos translating plasmoid reactor embodiment, CTOR, was introduced at the 1981 workshop, including passive wall stabilization in a long (40 m) burn chamber and requiring pulsed thermal loading of the first wall. The present Efremov approach allows a quasi-stationary operating mode to be considered, with

injection/merging of new plasmoids as a fuelling and flux injection mechanism;
reduced fluctuations in first wall thermal and neutron loadings;
efficient heating of new, low temperature plasmoids by fusion products and adiabatic compression;
phased introduction of plasmoids containing increasing fractions of ^{3}He (replacing T) to achieve advanced fuel operation after D–T startup.

Stability of the quasi-stationary FRC could be provided by feedback coil systems, which have yet to be worked out in greater detail. Impurity control is provided by the natural divertor intrinsic to the configuration. The quasi-stationary mode offers significant reductions in the mass power density of the fusion power core (FPC) relative to the translating mode. Generally, the technology features of the FRC burn chamber are similar to those of the tandem mirror reactor central cell. Consideration has been limited to modest first wall loadings.

6.2.2. Compact spheromak reactor (CSR)

The spheromak is a low aspect ratio (oblate) CT with nearly equal poloidal and toroidal fields, which has been formed experimentally by flux core (PPPL) or magnetized gun (Los Alamos) techniques. Fundamentally, this approach promises high beta, Ohmic heating to ignition, convenient (spherical) reactor geometry, and either translating or stationary plasmoid operation. A particular stationary plasmoid reactor embodiment has been developed by Los Alamos in which the magnetized gun source is also used to provide electrodes for helicity injection (i.e. steady state current drive) and magnetic divertor impurity control, in a thin blanket, resistive coil (equilibrium coils only) reactor. Key issues for the spheromak approach include:

scaling of beta and transport with plasma current;
edge-plasma/electrode interactions;
generic high power density, high heat flux technology issues.

6.2.3. Compact reversed field pinch reactor (CRFPR)

Although conventional, superconducting versions of the RFP reactor have been designed at both Los Alamos and Culham Laboratories, the resistive coil, compact, high power density CRFPR was introduced in a preliminary form at the 1981 workshop to explore the high mass power density regime of operation. Since that time, the CRFPR Fusion Power Core (FPC) has undergone considerable evolution at Los Alamos, [1, 2], and a larger, multi-institutional study of this class of systems (TITAN) has recently been initiated in the USA. Experiments at reactor relevant beta values, including major new constructions, exist in Europe, the USA, and Japan. Generally, the RFP configuration promises high beta operation, Ohmic heating to ignition, steady state oscillating field current drive (OFCD), impurity control by either (toroidal field) magnetic divertors or pumped limiters, and efficient low field (resistive) coil operation. Key issues for the RFP include:

scaling of confinement with increased plasma current or current density;
stability role/requirement of the conducting shell;
demonstration of DC current drive exploiting the non-linear internal flux conversion mechanism (dynamo);
perturbation of the confining edge plasma by the impurity control scheme;
generic high power density, high heat flux technology issues.

6.3. Conclusions

The AFC area has narrowed to a class of poloidal field dominated systems, supported by a significant worldwide experimental programme that anticipates attractive reactor operation, owing to high beta and reduced auxiliary heating requirements. Both superconducting and resistive coil embodiments are credible. In the latter case, high values of mass power density are achieved, resulting in cost savings and indirect fabrication and maintenance advantages. Operation of AFCs at high power density requires only modest technology development and extrapolation. The AFC programme offers both a backup and a competitive stimulus to the mainline tokamak programmes with distinct technical and economic advantages in commercial applications. Operation of AFCs at high beta values promises feasible exploitation of the D–^{3}He fuel cycle. Given that the long term source of ^{3}He may well be extraterrestrial [3] and that a worldwide programme of PFD system research exists, advanced fuel reactor design based on AFCs is a suitable topic for international collaboration.

REFERENCES

[1] HAGENSON, R.L., et al., Compact Reversed-Field Pinch Reactors (CRFPR): Preliminary Engineering Considerations, Los Alamos National Laboratory Rep. LA-10200-MS (August 1984).
[2] COPENHAVER, C., et al., Compact Reversed-Field Pinch Reactors (CRFPR): Fusion Power Core Integration Study, Los Alamos National Laboratory, Rep. PPG-969 (April 1986).
[3] WITTENBERG, L.J., SANTARIUS, J.F., KULCINSKI, G.L., Fusion Technol. **10** (1986) 167.

7. FUSION NUCLEAR TECHNOLOGY AND MATERIALS
(M.A. Abdou)

During the past few years there has been greatly increased interest and effort in fusion nuclear technology, which is reflected in the fact that this Fourth IAEA Technical

TABLE 7-I. DISTRIBUTION OF INTERNATIONAL EFFORT AMONG THE VARIOUS BLANKET CONCEPTS
(H = Strong Interest, L = Lesser Interest)

	Breeder/coolant/structure/multiplier	USSR[a]	USA[d]	Japan[d]	EC[d]
Self-cooled liquid breeders	Li/Li/V		H	L	
	Li/Li/FS		L[b]		
	Li/Li/AS				
	Li/Li/AS/U-Be	c			
	LiPb/LiPb/V		L[b]/H		
	LiPb/LiPb/AS				
	LiPb/LiPb/FS				L
	Flibe/Flibe/V		H		
Quasi-stagnant liquid breeders	Li/He/FS		L/H		
	Li/He/Mo			H	
	Li/He/FS/U-Pb	L			
	LiPb/H_2O/AS-FS				H
	LiPb/H_2O/AS/U-Pb	H			
	LiPb/He/FS/Be		H[b]		
	LiPb/PbBi/AS/U-Pb	H			
	Flibe/He/FS/Be		H		
Li_2O blankets	Li_2O/He/FS		L/H		
	Li_2O/He/FS-V/Be		L		
	Li_2O/He/Mo/Be			H	
	Li_2O/He/FS/U-Pb	L			
	Li_2O/H_2O/AS/Be			L	
	Li_2O/H_2O/AS/U-Pb	L[b]			
Ternary ceramics	$LiAlO_2$/He/FS/Be		L/H		H
	$LiAlO_2$/He/Mo/Be			L	
	$LiAlO_2$/He/FS/U-Pb	L			
	$LiAlO_2$/H_2O/FS/Be		H		
	$LiAlO_2$/H_2O/U-Pb	L[b]			
	$LiAlO_2$/NS/FS/Be		L		

a Hybrid blankets include ^{238}U as neutron multiplier.
b Considered for tandem mirror reactor.
c Considered for inertial confinement reactor.
d Does not include hybrid designs.

Committee Meeting and Workshop is the first to include fusion nuclear technology as a specific and separate issue for discussion. This increased interest and effort has come about because of the recognition that technology issues are crucial to the realization of the great economic and safety potential of fusion energy.

Fusion nuclear technology (FNT) encompasses those components that are concerned with fuel production and processing, energy extraction and use, and radiation protection of components and personnel. These components include the blanket, the first wall, the radiation shield, the tritium processing subsystem, and plasma interactive components. Since 1981, progress in FNT and materials has been made in developing basic property data, in the understanding of phenomena, and in identifying solutions to problems.

This summary is divided into three sections. Section 7.1 summarizes progress since 1981. Progress is reported in three areas: (1) design studies and analysis, (2) experimental data and modelling, and (3) safety and environmental considerations. Section 7.2 summarizes the key technical issues and identifies

TABLE 7-II. MAIN DESIGN FEATURES OF BLANKET CONCEPTS DEVELOPED FOR LONG TERM AND NEXT STEP REACTORS

	United States of America													
	Tokamak only										Mirror[f]			RFP[f]
	BCSS							TPSS			BCSS		MINIMARS	TITAN
Breeder	Li	Li	Flibe	Li_2O	$LiAlO_2$	$LiAlO_2$	$LiAlO_2$	Li_2O + Be	Li	Flibe	Li	LiPb	LiPb	LiPb
Coolant	Li	He	He	He	He	H_2O	NS	He	Li	Flibe	Li	LiPb	He	LiPb/H_2O
Structure	V	FS	FS	FS	FS	FS	FS	V	V	V	FS	V	FS/FS	FS
Neutron multiplier	–	–	Be	–	Be	Be	Be	Be	(Be)	–	–	–	Be	Be
Geometry	Rect. channels	Tubes pods	Tubes pods	Plates pods	Plates pods	Sphere pac pods	Sphere pac pods	Tubes	–		Rect. channels	Rect. channels	Tubes	Tubes
Flow direction	Poloidal	Toroidal	Toroidal	Toroidal	Toroidal	Toroidal	Toroidal		Poloidal		Poloidal	Poloidal		
Tritium carrier	Li	Li	Flibe	Purge	Purge	Purge	Purge	Coolant	Li	Flibe	Li	LiPb	LiPb	LiPb
Interest	H	L	L	L	L	L	–	L	H	(g)	L	H	H	H

	Soviet Union							Japan						European Community		
	ICFR[d]	OTR-3	OTR-1	OTR-2	OTR-4	TROL[b]	OTR-5[e]	JAERI						NET and national laboratories		
Breeder	Li	Li	LiPb	LiPb	TC[d], Li_2O	TC[d], Li_2O	TC[d], Li_2O	Li	Li	Li_2O	Li_2O	$LiAlO_2$	LiPb	LiPb	LiPb	$LiAlO_2$ Li_2SiO_3–Li_4SiO_4
Coolant	Li	He	H_2O	PbBi	He	H_2O	H_2O	Li	He	H_2O	He	He	H_2O	LiPb	He	He
Structure	AS	FS	AS	AS	FS	AS	Al	V	Mo	AS	Mo	Mo	AS/FS	FS	FS	AS/FS
Neutron multiplier	Hybrid[a]	Hybrid[a]	Hybrid[a]	Hybrid[a]	Hybrid[a]	Hybrid[a]	Pb	–	–	Be	Be	Be	–	–	Be	Be
Geometry	Tubes in sectors	Box	Tubes in box	Channels	Tubes	Tubes	Tubes	Rect. channels	Rect. channels	Sphere pac out of tube	Sphere pac out of tube	Sphere pac in tube	Tubes	Rect. channels	Canisters	Pins in tubes Sphere pac in pods Sphere pac in tubes
Flow direction	Radial	Poloidal	Poloidal	Poloidal	Toroidal	Poloidal	Poloidal	Poloidal	Poloidal	Poloidal	Poloidal	Poloidal	Poloidal	Poloidal	Radial	Radial toroidal (poloidal)
Tritium carrier	Li	Li	LiPb	LiPb	He	He	He	Li	Li	Purge coolant	Purge	Coolant	LiPb coolant	LiPb	Coolant	Purge coolant
Interest	–	L	H	H	L	L	H	L	H	L	H	L	H	L		H

[a] Hybrid blankets include ^{238}U as neutron multiplier.
[b] Considered for tandem mirror reactors.
[c] Considered for inertial confinement reactor.
[d] Ternary ceramics: $LiAlO_2$, Li_2SiO_3, Li_4SiO_4.
[e] Low power blanket.
[f] Other than tokamak copies.
[g] Concept receiving recent consideration.

the needs for new experiments and facilities. Section 7.3 presents the key recommendations. Efforts related to hybrid concepts are not discussed in this section because they are covered elsewhere. Reference to hybrid work is mentioned only when there is a strong overlap with the topics treated here.

7.1. Progress since 1981

The field of Fusion Nuclear Technology (FNT) has advanced considerably since 1981. Overall, the available database and analysis have more than doubled in the past five years. However, this progress has been uneven, with some areas making excellent progress while other areas remain somewhat behind. This indicates a continuing need for comprehensive technical planning. This section is divided into three subsections: (7.1.1) Blanket design studies and analysis, (7.1.2) Experimental data and modelling, (7.1.3) Safety and environmental considerations.

7.1.1. Blanket design studies and analysis

7.1.1.1. Major blanket design studies since 1981

The blanket design activity reported at this workshop reflects a double interest in commercial fusion reactors and in next step devices, with both pure fusion (NET in the EC) and hybrid (OTR in USSR) reactors being considered. The long term work has been dominated since 1981 by major blanket evaluation and comparison studies, updated by data from experimental work in progress.

The effort invested in near term blanket development has varied among the major fusion programmes depending on the ambitions of the various countries about nuclear technology testing in their respective next fusion device. Table 7-I indicates the present distribution of international effort among the various classes of blanket concepts. Table 7-II summarizes the main features of the blankets developed by each country, as well as an indication of the level of interest for each specific concept. The objectives aimed at by the various national studies are briefly presented in the following subsections.

7.1.1.1.1. Studies in the Soviet Union

Over the last few years, a number of different conceptual designs for experimental and commercial tokamak reactors have been developed in the Soviet Union. Almost all of these designs are devoted to fusion–fission hybrid reactors.

Accordingly, the major goal of these design analyses has been to maximize fission fuel production and energy multiplication. Because of the devotion to fuel production rather than pure fusion energy production, the major parameters, such as neutron wall loading, are less ambitious than in other commercial scale design studies.

Another important objective of the design studies was to assure adequate safety and reliability characteristics of blankets over the desired lifetime of $<5\ \mathrm{MW\cdot a\cdot m^{-2}}$. Several combinations of breeding and structural materials, as well as coolants, have been considered.

7.1.1.1.2. Studies in the United States

Blanket design activity in the United States, over the past five years, has been dominated by the Blanket Comparison and Selection Study (BCSS) for tokamak and tandem mirror reactors, with the primary objectives of (1) defining a limited number of blanket concepts to provide the focus of the blanket R&D programme, and (2) identifying and prioritizing critical issues for the leading concepts. The methodology involved a wide range of blanket concepts which were developed in sufficient detail to permit a relative ranking in the respective areas of engineering feasibility, economics, safety and R and D requirements.

The current design studies in the USA are the Tokamak Power System Studies (TPSS), MINIMARS, and TITAN. TPSS is a multi-laboratory effort focusing on making substantial advances and improvements in tokamak commercial designs. MINIMARS is a similar study of tandem mirror reactors and represents an update of the MARS results. TITAN is a new study devoted to the analysis of possible conceptual designs based on the reverse field pinch (RFP) confinement concept.

7.1.1.1.3. Studies in Japan

The recent blanket design activity in Japan has been mainly directed towards the performance analysis of candidate blanket concepts developed for a reference tokamak power reactor. This work aims at comparing the potential of various blanket designs to meet the requirements of self-sufficient tritium breeding, in-situ tritium recovery, and electricity generation. Moreover, the technological problems and critical R and D issues specific to power blankets are addressed by the evaluation of the candidate blanket concepts on the basis of the present database and in terms of engineering feasibility, economic performance and prospects for future improvements.

7.1.1.1.4. Studies in the European Community

The European Fusion Technology Programme, launched in 1982, set the framework for the co-ordination of national efforts on design studies and experimental programmes. The activity in blanket designs started with a comparison and evaluation study of concepts relevant to commercial tokamak reactors. The present effort in this field is increasingly directed towards NET, with major attention paid to the safety and reliability issues, and to the reactor relevance of the developed concepts. So far, the studies have concentrated on the design and analysis of $Li_{17}Pb_{83}$ blankets, helium cooled ceramic blankets, and first wall concepts.

In addition to these activities, the conceptual study of a DEMO reactor addressed many issues relevant to a tritium self-sufficient, electricity generating reactor. In particular, the impact of the first wall and divertor on the blanket design was strongly emphasized in this study.

7.1.1.2. Major trends and new ideas in blanket design in the last four years

A measure of the progress made in blanket design studies since the 1981 workshop may be made by briefly commenting on the current ideas at that time.

Among the blanket designs reported in the 1981 workshop:

(i) HYBALL, TOSCA and WITAMIR used the eutectic lithium–lead alloy, $Li_{17}Pb_{83}$, and NUWMAK used a lithium–lead alloy with a composition different from the eutectic ($Li_{62}Pb_{38}$).
(ii) None considered self-cooled lithium; only the FINTOR-D study proposed the option of a helium cooled blanket for a DEMO reactor.
(iii) All non-lithium blankets were water cooled (NUWMAK, STARFIRE, SPTR-P, and ANL-DEMO). All solid blanket concepts consequently utilized packed beds of ceramics cooled by pressure tubes.
(iv) All blanket concepts other than NUWMAK used stainless steel as structural material (STARFIRE, SPTR-P, FINTOR-D, and ANL-DEMO).

Recent work in blanket design reveals a strong evolution from former design options for both liquid and solid breeder blanket concepts, and an increasing convergence of international interest in specific areas such as self-cooled lithium blankets, static $Li_{17}Pb_{83}$ blankets, and helium cooled ceramic blanket concepts. Although not all progress in this field can be covered here, the more significant trends and ideas are briefly presented in the following paragraphs.

7.1.1.2.1. General trends

The differences in the distribution of the international effort among the various blanket concepts (Table 7-I) reflect the major interest of the USSR in hybrid blankets and the interest of the USA, USSR and Japan in a wide range of blanket solutions. In contrast, the European Community has concentrated its efforts on a small number of leading concepts. In particular, lithium coolant/breeder, Li_2O breeder and water cooled solid breeder blankets have not, at present, been thoroughly addressed in the European fusion technology programme.

Of importance is the increasing attention paid to safety issues since 1981, and the trend to implement early in the design adequate features for passive removal of afterheat and for safe control of pressure pipe rupture and the resulting loss of coolant accident. Furthermore, computational tools developed to simulate the tritium exchanges between the tritiated components, such as the blanket and the processing units, may be used to optimize tritium management with respect to safety.

The trend towards more compact blanket designs was illustrated by the evolution from UWMAK to STARFIRE and was reported as a major trend of the 1977 to 1981 period; no further progress in this direction was recommended afterwards since the STARFIRE design was adopted as reference boundary conditions for the candidate tokamak reactor blankets considered in BCSS.

Also noticeable is the replacement of pumped limiters, extensively considered around 1981, by single or double null divertors in many of the present DEMO or commercial reactor designs. In addition to consideration of the blanket integration and maintenance schemes, this trend results in a decrease in the blanket coverage ratio and in a strong incentive to enhance the breeding performances. On the other hand, the trend in the USA is to further simplify the design and to consider replacing actively pumped limiters with self-pumping passive limiters.

7.1.1.2.2. Liquid breeder blankets

The major trend in the area of liquid blankets is a strong renewal of interest in self-cooled lithium blankets (USA, Japan, USSR), expected to lead to simpler designs and, consequently, to improved reliability. This trend not only provides incentive to more experimental investigation of MHD issues, but also reactivates interest in advanced structural materials (vanadium or molybdenum alloys) and the investigation of alternatives to water as coolant in the divertor/limiter and shield.

Also noteworthy is the renewed interest in the lithium-lead eutectic alloy as breeding material, either water cooled (EC, USSR) or self-cooled (USA, EC), and contained in austenitic or ferritic steel structures or vanadium alloy. A new class of static $Li_{17}Pb_{83}$ blankets is emerging, with breeding properties enhanced by the use of beryllium as an additional neutron multiplier, and with the tritium directly recovered from the helium coolant stream (USA, EC).

Interest in helium cooled lithium blankets in ferritic steel or molybdenum alloys is widely shared among the USA, Japan and USSR. The activity in this area reflects more the tentative optimization of the module geometry, rather than a radical evolution from the FINTOR-D design.

The use of Flibe in place of lithium in self-cooled or static blankets is being considered in the USA with increasing interest. If chemistry control appears acceptable, the Flibe concept would retain the lithium advantages of simple design while eliminating the lithium disadvantages of high chemical reactivity and high MHD pressure drop.

The potential use of PbBi as coolant in ducts of ferritic steel with an oxidized inner surface is receiving attention in the USSR.

7.1.1.2.3. Solid breeder blankets

Recent solid blanket design studies reflect the effort invested in innovative helium cooled designs, in contrast to the limited improvement of water cooled blankets since STARFIRE (1980). A major trend, therefore, is the renewed interest in helium as an attractive coolant for the solid blankets (USA, Japan, EC, and USSR) and the recognition of the efficient breeder temperature control afforded by the radial and the radial/toroidal flow directions. As a result, most helium cooled blanket concepts are based on pods (USA, Japan, EC), radial canisters (USSR, EC), or toroidal tubes (USA, USSR, EC). This recent progress contrasts with the fact that no significant improvement in helium cooled solid blankets was produced between the UWMAK II design (1976) and 1981.

Also important is the recognition of beryllium as the most credible neutron multiplier (USA, Japan, EC) and the increasing trend to use it distributed among the breeder elements (e.g. sandwiched between adjacent ceramic layers), or mixed with ceramic pellets or packed beds (USA, Japan, EC), rather than as a slab at the front, as considered in the STARFIRE design. Among the advantages over other neutron multipliers is the enhancement of breeding performance and of the energy multiplication factor and the potential to improve the thermal conductivity of Be/ceramics breeder elements (USA, Japan, EC). Lead is still considered as a potential candidate for low power blankets (INTOR, USSR).

The renewal of interest in helium as a coolant also reactivated the trend of externally cooled canned breeder elements such as pin bundles (EC), tubes (USA, Japan, EC), or plates (USA) as alternatives to packed beds cooled by pressure tubes.

7.1.1.2.4. Tritium modelling

Of significant importance for the blanket design effort is the considerable attention given since 1981 to the area of tritium breeding and processing, which strongly interacts with blanket design requirements. Extensive modelling of tritium transport, trapping and permeation phenomena, updated by data generated by ongoing experimental activity, enabled the development of computational tools to simulate tritium exchanges between the various media and fluids of the blanket. Furthermore, the analytical modelling of some tritium recovery techniques, backed up by the experimental work in progress, permitted calculation of the exchanges of tritiated streams between the major components of a fusion reactor, and thus, simulation of the relevant tritium cycle.

These studies were applied to evaluate the minimum breeding requirements to achieve tritium self-sufficiency and demonstrated the incentive to minimize total tritium inventory in the blanket and its processing lines as a whole, rather than the blanket inventory only. Meeting this requirement strongly affects the design and operating conditions of solid and static liquid blankets. One particular point arising from such analyses is the possibility of recovering all the bred tritium from the primary coolant as an alternative to keeping special purge circuits which significantly complicate the design.

Also of great importance is the capability of such computational tools to accommodate the progress made in tritium technology, and to predict, for example, the consequences for the blanket operation of an easier tritium release from the breeder in a reducing atmosphere, as evidenced by the recent in-pile tritium release experiments.

7.1.2. Experimental data and modelling

Acquiring experimental data and the development of theory and computational models are critical to understanding and quantifying relevant phenomena for fusion nuclear technology. Remarkable progress has been made in a number of areas. A summary of progress in acquiring experimental data and development of models is given in this section for the following areas: (1) liquid metal MHD, (2) liquid metal corrosion and mass transfer, (3) solid breeder materials, (4) blanket tritium extraction, (5) neutronics and tritium breeding, (6) electromagnetics, (7) plasma interactive components, and (8) structural materials.

7.1.2.1. Liquid metal MHD

The main trend in liquid metal (LM) MHD experiments has been to obtain higher values of characteristic dimensionless parameters, such as Hartmann number, M, and MHD interaction parameter, N. Considerable progress has been achieved with putting into operation some new test facilities. The main features of these facilities are summarized in Table 7-III.

Certain steps have been made in testing several types of channels with complex geometry. Many more experiments with channels of increasing complexity are planned for the new facilities.

The first series of experiments with free surface MHD flows consistent with recently suggested LM plasma interactive components design concepts (LM limiters/divertor plates) have been initiated. The feasibility of several types of MHD film and droplet flows at moderate interaction parameters has been successfully demonstrated. These experiments provide a viable basis for starting complex testing of LM concepts in larger facilities with more advanced simulation of parameters and geometry.

7.1.2.2. Liquid metal corrosion and mass transfer

Corrosion and mass transfer are key concerns for liquid metal blankets. Since 1981 a significant effort has been devoted to the compatibility of steels in the presence of flowing lithium-lead eutectic alloy, $Li_{17}Pb_{83}$. Comparatively, the compatibility studies with lithium were appreciably decreasing, owing to relatively better knowledge of corrosion by this liquid metal.

(a) *Test facilities.* Thermal convection and forced circulation loops are now relatively numerous (Table 7-IV). Generally, these loops are small, but some of them are fitted with mechanical devices, allowing the study of stress-corrosion interaction and liquid metal embrittlement.

(b) *Corrosion by* $Li_{17}Pb_{83}$. The results obtained in thermal convection or forced circulation with low velocity (about 10 $cm \cdot s^{-1}$) are summarized in Table 7-V.

The majority of the tests were devoted to austenitic steels such as 316 SS, but some results concerning ferritic materials are already available. In both cases, corrosion is characterized by mass transfer between hot and cold legs. The hot leg specimens are covered by a porous corrosion layer consisting of ferrite and a network of channels filled with lithium-lead. Mass transfer appears to be five to ten times greater for austenitic steels than for ferritic steels.

At 450°C the results obtained in the different laboratories are in good agreement and reveal an unacceptable corrosion rate for austenitic steels. At 400°C the attack is less severe but remains non-negligible, since the ferrite thickness is about 15 μm after 3000 h, with the corrosion rate being practically linear.

Therefore, the temperature limit for austenitic steel utilization probably cannot go beyond 400°C and might be even lower. The temperature limit is certainly higher for ferritic steels, although its determination requires more experimental results.

(c) *Corrosion by lithium.* The results summarized in Table 7-VI are very scattered. That confirms the strong dependence of the corrosion rate on parameters other than temperature, such as velocity of lithium, presence of impurities, type of purification, and geometry of the loop. However, comparative results obtained at 450°C in the same laboratory with similar loops show that lithium is much less corrosive than $Li_{17}Pb_{83}$ alloy.

(d) *Future orientations.* Generally speaking, the future experimentations on liquid metal corrosion should tend to

TABLE 7-III. KEY PARAMETERS OF NEW MHD TEST FACILITIES

Facility, location	Start of operation	Maximum magnetic field (T)	Testing volume (m X m X m)	LM	Flow rate, ($L \cdot s^{-1}$)	Maximum MHD parameters, (M, N)
Sodium loop, Inst. Phys., Riga, USSR	1982	2	$0.25^a \times 0.25$	Na	15	10^4, 5×10^4
ALEX, USA	1985	2	$1.5 \times 0.8 \times 0.15$	NaK	19	10^4, 10^4
MALICE, Belgium	1985	2	$1 \times 0.1 \times 0.1$	Li		

[a] Diameter of a solenoid bore.

TABLE 7-IV. LITHIUM AND LITHIUM-LEAD CORROSION LOOPS

Laboratory	Li		$Li_{17}Pb_{83}$	
	Thermal convection	Forced circulation	Thermal convection	Forced circulation
MOL (Belgium)	X	X	X	
CEA (France)			X	
Kurchatov Efremov (USSR)		X		
ANL ORNL (USA)	X	X	X	X
KfK (FRG)		X		X

TABLE 7-V. CORROSION RATES IN $Li_{17}Pb_{83}$

Laboratory	Alloy	Temperature (°C)	Duration (h)	Ferrite thickness (μm)	Corrosion rate ($\mu m \cdot a^{-1}$)
ANL	316 SS	450	3000	32	90
CEA		450	3000	33	–
MOL		450	1000	35	83
CEA		400	3000	15	–
MOL			1000	8	13
ANL	Fe-9Cr	450	3000		9

TABLE 7-VI. CORROSION RATES IN LITHIUM

Laboratory	Alloy	Temperature (°C)	Corrosion rate ($\mu m \cdot a^{-1}$)
ANL	316 SS	450	2
USSR		420	250
MOL		400	12
USSR	20 Cr-45 Ni	420	1000
USSR	Fe-13 Cr	420	200
ANL	Fe-9 Cr	450	0.1
MOL		400	5

approach as near as possible the operating conditions of the blanket. In this respect, parameters such as long term exposure, stress, magnetic field, geometry of the blanket, and realistic impurity level depending on the possibility of purification may influence the corrosion rate and should be taken into account. In the particular case of the $Li_{17}Pb_{83}$ alloy, the main effort should be devoted to ferritic steels, and possibly to vanadium alloys if metallurgical and mechanical properties prove satisfactory. In the analytical field an accurate method of determination of the lithium-lead ratio has to be developed with a view towards keeping the breeder at the eutectic composition.

7.1.2.3. Solid breeder materials

(a) *Properties.* The properties database on lithium oxides and on ternary compounds, $LiAlO_2$, Li_2SiO_4, Li_4SiO_4 and Li_2ZrO_3 has been considerably extended. Preirradiation thermal and thermomechanical characteristics are being defined on a statistical basis and as a function of structure (texture, morphology, porosity, etc.). Thermomechanical properties have been stated, and the upper levels of the temperature operating windows have been tentatively defined for all the ceramic material considered. Li_2O was studied mainly in Japan and the USA. The EC countries and the USA evaluated the ternary oxides. The gamma-$LiAlO_2$ is being widely studied in Europe to relate its physicochemical properties to the defective status induced by the different fabrication procedures developed in France, the UK and Italy. Belgium and the Federal Republic of Germany are mainly considering the lithium silicates, with the UK involved in all the candidate breeder materials.

(b) *Compatibility with structural materials.* The reactivity of solid Li_2O, $LiAlO_2$ and Li_2SiO_3 with austenitic and ferritic alloys has been investigated at 600 and 700°C. Li_2O appears to be the more reactive material. The corrosion by $LiAlO_2$ and Li_2SiO_3 is very limited. However, the presence of impurities in ceramic and/or in helium may induce localized attacks of the steels. The behaviour in-pile has to be investigated.

(c) *Irradiation experiments in closed capsules.* Table 7-VII summarizes the irradiation experiments completed and active worldwide since 1984. The FUBR-IA data have been published and Li_2O showed considerable swelling under irradiation, while the ternary ceramics, especially $LiAlO_2$, gave non-remarkable dimensional and structural variations. Most of the post-irradiation examinations are presently ongoing in EC laboratories. Preliminary data confirm that $LiAlO_2$ and Li_2SiO_3 do not undergo considerable swelling.

Soon many and quite differently prepared materials will be irradiated under considerable fast neutron fluences as part of the BEATRIX programme. They will undergo radiation dose equivalent to up to about one year of an operating fusion reactor blanket, at temperatures ranging from 500°C to 900°C, in the EBR-II reactor.

Intercomparison experiments to determine the end of life of the various candidate breeder materials (burnup max) are also envisioned in the frame of the EC programme by using both fast neutron (KNK reactor) and thermal neutron (OSIRIS reactor) sources.

(d) *In-situ tritium recovery.* The TRIO experiment has been successfully performed in the USA, demonstrating the feasibility of the tritium recovery process in conditions also closely related to those expected in an operating solid breeder blanket. Both in TRIO and in the more recent European LISA and EXOTIC experiments, addition of hydrogen to the helium sweeping gas was found essential to obtain reasonable tritium inventories in $LiAlO_2$ and Li_2SiO_3 ceramics. This statement has important implications on the tritium reprocessing plant design. Minimizing the tritium inventory by reducing the particle sizes of the $LiAlO_2$ breeder compound has to be fully demonstrated. Grain size was, in fact, found to play an important role, but tritium extraction from the ceramic materials does not seem to be entirely controlled by its solid state diffusion coefficient. Other phenomena such as intragranular diffusion, permeation through the interconnected porosity, and adsorption processes, may greatly affect the tritiated species as a function of the chemical conditions of the environment other than that of the temperature. These aspects have to be clarified by the planned experiments in the SILOE and NRU (Canada) reactors, supported by appropriate physicochemical characteristics.

Minimizing the grain size to levels that give high specific surfaces, or surface activations, could adversely influence the tritium recovery process from ceramic bodies representative of realistic dimensions for its component in the blanket. Finally, a further goal of these experiments is the definition of the lower temperature operating levels allowed to obtain the required tritium inventory in the blanket.

(e) *Fabrication.* Many fabrication processes have been developed on a lab scale in order to get suitable 'ceramic grade' powders of the ternary oxides and Li_2O. Forming processes, such as hot pressing (US and UK) or cold pressing and sintering (EC countries) have been developed to prepare pellets with the required geometry and density, taking a near fully open porosity. Table 7-VII also summarizes the variety in the examined products.

Microspheres of gamma-$LiAlO_2$ have been and are being prepared for blanket concepts based on sphere-pac or pebble solid breeders. Pilot plant industrial oriented productions have also

TABLE 7-VII. COMPLETED AND ACTIVE SOLID BREEDER MATERIAL IRRADIATION EXPERIMENTS

Experiment	Ceramic	Grain size (μm)	Density (% TD)	Temperature (°C)	Li burnup (Max. at.%)	Reactor	Time frame
Closed capsule							
ORR (US)	Li_2O	<47	70	750, 850, 1000	0.05	ORR	–
TULIP (US)	Li_2O	50	87	600	3	EBR-II	84
FUBR-1A (US)	Li_2O	6	85	500, 700, 900	1.5	EBR-II	84/85
	$LiAlO_2$	<1	85, 95	500, 700, 900	3		84/85
	Li_4SiO_4	2	85	500, 700, 900	2		84/85
	Li_2ZrO_3	2	85	500, 700, 900	2		84/85
FUBR-1B (US)	Li_2O	<5	60, 80	500, 700, 900	5	EBR-II	85/89
	Li_2O	<5	80	500–700/1000			
	$LiAlO_2$	<5–10	80	500, 700, 900	9		85/89
	(sphere-pac)		80	500–700/1000			
	Li_4SiO_4	<5	80	400–500	9		85/89
	Li_8ZrO_6	<5	80	600–700	7		85/89
	Li_2ZrO_3	<5	85	520–620	7		85/89
ALICE (France)	$LiAlO_2$	0.35–13	71–84	400, 600	–	OSIRIS	85/86
DELICE (FRG)	Li_2SiO_3 (Li_4SiO_4)	–	65, 85, 95	400, 600, 700	<0.02	OSIRIS	85/86
ORDALIA (Italy)	$LiAlO_2$	0.4–2.0–10	80	500, 700	–	OSIRIS	86/87
EXOTIC (Neth./UK/ Belgium)	Li_2SiO_3	–	80	400, 600	–	HFR	85/86
	Li_2O	–	–		–		85/86
	$LiAlO_2$	30	80		–		85/86
	Li_2ZrO_3	–	–		–		85/86
CREATE (Canada)	$LiAlO_2$	<1	80, 90	100	–	NRU	85/86
In-situ tritium recovery							
TRIO (US)	$LiAlO_2$	0.2 (50 μm particles, 0.9 cm thick annular pellet)	65	400, ..., 700	0.2	ORR	84/85
VOM-15H (Japan)	Li_2O	<10	86	480, ..., 760	0.24		84
VOM 22/23 (Japan) (Japan)	Li_2O	– (1.1 cm pebbles)	–	400–900	0.04		–
	$LiAlO_2$	– (1.1 cm pebbles)	–	400–900	0.1		–
LILA (France)	$LiAlO_2$	1–30 (1 cm diameter pellet)	78	375–600	<0.02	SILOE	86
LISA (FRG, France)	$LiAlO_2$	0.4	78	450–730	–	SILOE	86
	Li_2SiO_3	30–80	86–93	450–730	–		
	Li_4SiO_4	26 (1 cm diameter pellet)	94	450–730	–		
EXOTIC (Neth./UK/ Belgium)	$LiAlO_2$	30 (1.4 cm diameter pellet)	80, 95	400, 600	<0.4	HFR	86
	Li_2SiO_3	– (1.4 cm diameter pellet)	50	400, 600	<0.4		86
TEQUILA (Italy)	$LiAlO_2$	0.4–2.0–10 (1 cm diameter pellet)	80	500, 700	–	SILOE	86/87
CRITIC (Canada)	Li_2O	60 (1 cm thick annular pellet)	90	400–900	0.15	NRU	86

TABLE 7-VIII. BLANKET TRITIUM EXTRACTION METHODS AND INTERNATIONAL INTEREST

Carrier fluid	Tritium form		Extraction method[a]	USSR	USA	Japan	Europe	Canada
	T_2/HT	T_2O/HTO						
LiPb	X		Vacuum degassing	X				
	X		*Extraction with countercurrent He flow*	X	X		X	
	X		Permeation combined with catalytic oxidation				X	
Li	X		Absorption with solid getters					
	X		*Extraction with molten salt*		X			
He (purge)	X	X[b]	*Absorption with solid getters*	X				
	X[c]	X	*Absorption with molecular sieves*		X			X
	X[c]	X	Freezing out in cold traps				X	
He (coolant)	X	X[b]	*Absorption with molecular sieves*	X	X	X	X	X
H_2O		X	Vapour phase catalytic exchange					
		X	Liquid phase catalytic exchange		X			X
		X	Electrolysis				X	

[a] Preferred method in italics.
[b] Additional process needed to decompose T_2O, HTO.
[c] Additional process needed to oxidize T_2, HT.

been tested both for Li_2O and $LiAlO_2$. It seems possible to get large productions of all the materials considered, once reasonable and well defined specifications are made.

7.1.2.4. Blanket tritium extraction

During the past five years tritium extraction research and development has been restricted mainly to defining concepts and performing process flow analysis, although in the USA some experimental work is in progress in connection with the Tritium Systems Test Assembly. Generally, tritium extraction methods have been considered in the context of overall blanket studies since the particular method chosen has a significant impact on the blanket design and performance.

The various methods of blanket tritium extraction can be summarized according to the fluid used to transport the tritium from the breeder. The potential carriers in different breeder systems are $Li_{17}Pb_{83}$, Li and He. Extraction of permeated tritium from water is also of interest. The various extraction processes and level of international interest in each one is shown in Table 7-VIII.

For solid breeder blanket concepts, tritium extraction via a separate helium purge circuit with either molecular sieve adsorption or cold trap tritiated water freeze out has received considerable attention. On the other hand, the realization that the amount of blanket structure must be minimized on the grounds of simplicity and neutron economy has led to an increase in interest in direct coolant extraction techniques, in which case molecular sieve adsorption columns are generally required. For both extraction methods there is a major interest in the use of molecular sieves.

Tritium extraction from liquid metal blankets has, in the case of LiPb breeder concepts, been focused largely on the use of a counter current helium flow scheme in which helium is bubbled up through the breeder material, although no experiments have yet been performed. In addition, increased attention has been focused on the difficulty of tritium extraction due to its low solubility in LiPb. Other methods such as vacuum degassing and in or out of reactor permeation are also under consideration. For Li breeder blanket concepts the preferred method of tritium extraction uses a molten salt process, the feasibility of which was established before 1981.

In general, a large number of tritium extraction concepts have been evaluated as part of detailed blanket and reactor designs. However, at present, many of these concepts remain to be validated by experimental programmes.

7.1.2.5. Neutronics and tritium breeding

Significant progress has been made in three areas: (1) basic nuclear data measurements and evaluation, (2) integral benchmark experiments, and (3) neutronics analysis. Of particular significance are new measurements on beryllium (n, 2n) and ^7Li (n, n'α)t cross-sections. A new facility dedicated to fusion integral neutronics experiments has been completed at JAERI, Japan. As part of a collaborative effort between the USA and Japan, a series of experiments simulating fusion blankets have been performed.

Results have shown general agreement between calculation and experiments but significant discrepancies were uncovered in several areas, particularly in blanket assemblies containing beryllium. New advances occurred in experimental measurement techniques, most notably in measurement of tritium production rates, to a high accuracy and in reliably measuring neutron energy spectra below 1 MeV. Integral experiments have also been carried out in other facilities such as OKTAVIAN in Japan and LOTUS in Switzerland.

Detailed analysis of DT fuel self-sufficiency has greatly improved our understanding of the key factors and critical issues for both the achievable and attainable tritium breeding ratios. Results show that meeting the fuel self-sufficiency conditions depends not only on the neutronics characteristics of the blanket, but also depends on many reactor system parameters such as tritium fractional burnup, tritium extraction efficiency, tritium inventory retained in the blanket, and size and geometry of penetrations in the blanket.

7.1.2.6. Electromagnetics

As reactor designs have become more detailed, there has been a growing awareness of the need to perform a more comprehensive analysis of the effects of electromagnetic transients on engineering designs. In the USA, Europe and Japan, considerable effort has been expended in the last five years in developing sophisticated computer codes which allow the effects of transient phenomena, such as disruptions, on complex reactor components to be predicted. Several of these codes have been useful design tools allowing complete stress calculations to be performed. In the USSR, considerable progress has been made in obtaining analytical models to describe both plasma disruptions and the stability of elongated plasmas in the presence of conducting components, the latter allowing the analysis of active stabilization to be made. On the experimental side, although the US FELIX experiment is now inactive, increasing use of existing fusion machines has been made, particularly in the area of plasma stability control. During the last five years, electromagnetics has become an important issue in fusion nuclear technology and one which has received increasing attention throughout the design process.

7.1.2.7. Plasma interactive components

The high fluxes of plasma particles and heat to components such as limiters, divertor plates, RF antennas and launchers, as well as to the first wall, place considerable demands on materials. The key issues in this area can be grouped into the categories of particle and impurity control, heat removal, and off-normal events. Activities in each of these depend strongly on materials characterization and development, and are also tied closely to experimentation in actual confinement devices.

During the past years, several international collaborations have been initiated in areas which include plasma–materials interaction and high heat flux (PMI/HHF) issues. These include investigations using plasma physics machines such as TEXTOR (KfK, Jülich), TORE SUPRA (CEA, Cadarache), ASDEX Upgrade (MPI, Garching), and JET (CEC, Culham).

In the USA, there are three primary laboratory facilities which concentrate on PMI/HHF issues. These are the Plasma Materials Test Facility (PMTF), at Sandia National Laboratories, Albuquerque (SNLA); the Plasma–Surface Interactions Experimental Facility (PISCES), at the University of California, Los Angeles (UCLA); and the Tritium Plasma Experiment (TPX), at Sandia National Laboratories, Livermore (SNLL). In addition, the US programme has a significant effort in determining basic PMI property data using accelerator facilities. These facilities have been used to perform experiments on erosion/redeposition, tritium permeation/retention and simulation of the effects of disruptions on plasma side materials.

In the USSR, plasma–wall interaction studies have been carried out on tokamaks (T-7, T-19 and others) and also in experiments with charged particle accelerators. Some synergistic effects were investigated by the simultaneous irradiation of structural materials with different particle fluxes as well as the mass transfer and disruptions impact on the selective sputtering, radiation erosion, and surface chemical composition.

In addition to the above experimental work, conceptual studies of divertor/limiter designs have been undertaken. Where low plasma edge temperatures have been assumed, the studies have been concerned with solid limiter and divertor target designs. On the other hand, where more pessimistic physics assumptions and high plasma edge temperature have been used, designs have focused on the use of liquid metal covered divertor target concepts.

A new issue is now receiving some attention. The use of reactive liquid metals in the blanket precludes the use of water in the plasma interactive components (PIC). Therefore, the viability of cooling limiters/divertors with liquid metals is now being assessed.

7.1.2.8. Structural materials

During the period from 1981 to 1986, great progress has been achieved in the understanding of materials for the fusion reactor first wall and blanket. At present, fission irradiation data have been obtained on austenitic SS at up to 100 dpa ($8-10\ \mathrm{MW \cdot a \cdot m^{-2}}$). It has been shown that cold work and new alloying composition reduces various types of steel swelling, though at doses higher than 100 dpa swelling is still unacceptable.

It has also been shown that ferritic steels are highly resistant to swelling (at up to 100 dpa) and do not exhibit signs of swelling in simulation tests at high doses. Initial results on the compatibility of ferritic steels with liquid metal breeders (Li, $Li_{17}Pb_{83}$) are promising.

Efforts should be made on low activation alloys such as vanadium. Samples of austenitic and ferritic materials have been developed which resulted in lower activities in the neutron field. New results indicate that it would be possible to increase the radiation resistance of both superconducting materials and polymid insulators.

The growth of international collaboration in solving fusion technology and material problems for fusion reactors is noted. In some cases, the experimental database is too small to fully estimate the materials performance in future fusion reactors.

7.1.3. Safety and environmental considerations

The objectives of fusion safety and environmental studies are to (a) understand safety-related phenomena, (b) integrate

TABLE 7-IX. SAFETY RELATED PROGRESS IN DESIGN STUDIES

Issue	Progress	Current level of understanding	Comments
Low long term activation materials (waste management)	H	H	Some key uncertainties remain. More work needed on cost/benefit of this approach to waste management
Low short term activation materials (accident safety)	M	L	Can greatly impact accident hazards and decay heat. Possible conflict w/high MPD and low long activation
Impact of varying blanket concepts	H	M	BCSS included detailed examination
Impact of different physics confinement schemes	L	L	Physics determines effects and partially determines frequency of disruptions and hazardous plasma terminations
Impact of different magnet coils	L	L	Copper versus superconducting magnets have different failure modes and effects
Design of inherent/passive safety	M	M	Safety issues being defined. More attention needed on potential costs and/or economic benefits

H – high progress or understanding
M – medium
L – low

this understanding into designs of experimental facilities and conceptual power plants, and (c) help fulfil fusion's high potential in achieving excellent safety and environmental attractiveness.

7.1.3.1. Trends and overall progress

Since 1981, considerable progress has been made in considering safety and environmental concerns in fusion designs and research. Fusion safety programmes are now active in the USA, Europe, Japan and the USSR. Three major trends can be observed:

(1) More sophisticated data and computer models are becoming available. These make possible meaningful safety analyses which were previously impossible and make possible reduction of some overconservatisms.
(2) The spectrum of issues being considered has expanded from a few, narrow viewpoints to include virtually all significant concerns. This involves examination of interrelationships, tradeoffs, and optimization. This process has helped lead to renewal of interest in liquid lithium.
(3) Although more progress is needed, conceptual designers are increasingly considering safety and environmental factors early in the design process, rather than as add-on features later. Experience in the US Blanket Comparison and Selection Study (BCSS) suggests that this approach helps lead toward the required improvement in both economics and safety. It is notable that the best four blankets in the overall study were also the best four blankets in the BCSS safety evaluation.

Associated with these trends are four major new safety ideas which deserve special note:

(1) Use of low activation materials. Low activation appears to be a unique fusion opportunity. Complications arise from the difference between long term activation (waste management) and short term activation (accidents, decay heat).
(2) Design for inherent/passive safety. Such designs would attempt to prohibit very serious public consequences from *any* accident.
(3) Use of probabilistic risk assessment (PRA) tools. These tools help maximize safety and reliability with minimum cost.
(4) Recognition of tradeoffs involving safety. One class of tradeoffs is among safety and environmental concerns. Another class of tradeoffs is among safety, economics, and reliability and maintenance.

The progress and current level of understanding in safety related designs and system integration are summarized in Table 7-IX.

Finally, the first DT operation of plasma test devices is only a few years away. Safe operation of this first generation of DT burning test facilities must be ensured, recognizing that any significant accident could adversely influence public opinion towards fusion.

7.1.3.2. Experiments

Very valuable experimental data have become available since 1981. In fact, data produced in the last five years have more than doubled the fusion safety database. The overall

TABLE 7-X. SAFETY RELATED EXPERIMENTAL PROGRESS AND STATUS

Issue	Progress	Current level of understanding	Comments
Reaction severity of lithium with:			
– air	H	H	Remaining major issue is how to safely use lithium in designs, which appears difficult
– water	H	H	
– concrete	H	H	
Reaction severity of $Li_{17}Pb_{83}$ with:			
– air	H	M	Air reaction benign except for some volatiles. Water and concrete reactions produce hydrogen and radioactive volatiles
– water	M	L	
– concrete	M	M	
Tritium control of:			
– inventory	M	M	Long-recognized issues; substantial database being acquired
– routine effluents	M	M	
– accidental releases	L	L	
Activation product control:			
– structural radioactivity	M	L	Pathways include fluid spills and processing, structural oxidation, and combustion
– fluid radioactivity	M	L	
Thermal-hydraulic transients:			
– water coolant	L	H	Non-fusion data generally adequate for water and helium. MHD complicates liquid metal analysis
– helium coolant	L	H	
– liquid metal coolant	L	L	
Plasma transients:			
– effects on FW/blanket/ limiter, etc.	L, L	L, L	First value for tokamaks, second for others. Should consider plasma transients caused by external events, e.g. coolant leakage
– prevention	M, L	L, L	
Magnet transients:			
– arcs	M	M	Magnets are potential cause for structural damage
– quench	M	M	
– loss of coolant	L	L	

H – high progress or understanding
M – medium
L – low

progress and current level of understanding are summarized in Table 7-X. Space does not allow detailed discussion; however, three achievements deserve special attention:

(1) Substantially improved understanding of tritium control. The TSTA (Tritium Systems Test Assembly) began operation in the USA in 1985. TSTA is providing required data on typical tritium components and systems. Canadian experience in CANDU fission reactors provides evidence of improved prospects in reducing tritiated water leakage from water coolant systems and evidence of improved prospects for reducing the cost of removing tritium from water coolant.

(2) Substantially improved understanding of liquid metal (LM) chemical reaction severity. Experiments at the Hanford Engineering Development Laboratory (HEDL), as part of the US Fusion Safety Program, have examined lithium and $Li_{17}Pb_{83}$ reactions with air, water and concrete. Experiments at the Joint Research Centre (JRC) in Ispra, Italy, have been conducted with $Li_{17}Pb_{83}$ and air and water. The reactions with lithium are now generally well understood; the main issue is how to safely use lithium in designs. The $Li_{17}Pb_{83}$ reactions are now known to be far less severe than lithium reactions in terms of elevated temperatures and pressures. However,

TABLE 7-XI. SAFETY RELATED COMPUTER MODELLING PROGRESS AND STATUS

Issue	Progress	Current level of modelling	Comments
Reaction severity of lithium with:			
– air	H	H	Geometry effects of lithium-water reaction needed.
– water	L	L	Li concrete modelling can be based on Na concrete
– concrete	L	L	
Reaction severity of $Li_{17}Pb_{83}$ with:			
– air	L	L	Air–LiPb case may not be needed. Water–LiPb
– water	M	L	geometry effects must be modelled
– concrete	L	L	
Tritium control of:			
– inventory	H	M	Further modelling often limited by lack of data in
– routine effluents	H	M	relevant regimes
– accidental releases	H	M	
Activation product control:			
– structural radioactivity	L	L	Extremely complex phenomena and range of
– fluid radioactivity	M	L	conditions, with limited data, make modelling difficult
Thermal hydraulic transients:			
– water coolant	M	H	Non-fusion codes generally good basis for water,
– helium coolant	M	H	helium. MHD complicates liquid metal case
– liquid metal coolant	M	M	
Plasma transients:			
– effects	H, L	M, L	First value for tokamaks, second for others. Both
– prevention	L, L	L, L	thermal and EM effects are important. Both plasma induced and externally induced transients are important
Magnet transients:			
– arcs	M	M	Major issue is copper versus superconducting coils
– quench	M	M	
– loss of coolant	L	L	

H – high progress or understanding
M – medium
L – low

it is known that in some geometries, $Li_{17}Pb_{83}$-water reactions can cause significant problems, e.g. hydrogen generation and some mobilization of radioactive species from the $Li_{17}Pb_{83}$.

(b) Initiation of study of activation product behaviour. Historically, tritium has received the most attention in fusion safety and environmental studies. The other major hazard (except for hybrid reactors, which have fission products) is fusion neutron activation products. The hazard potential caused by activation products depends on the fluence, selected materials, accident energy sources, design response to transient fault conditions, and the potential for the activation products to be volatilized. Some data are now available on the volatilization of structural metals, PCA, HT-9, and V15Cr5Ti, and of liquid metal constituents and corrosion products. Structural volatilization can result from accidental high temperature oxidation. Fluid species volatilization can result from combustion or simple spill. Routine processing of fluids to control corrosion is a possible source of effluents. Considerably more data are required in the area of activation products to be able to confidently predict routine and accident behaviour of activation products.

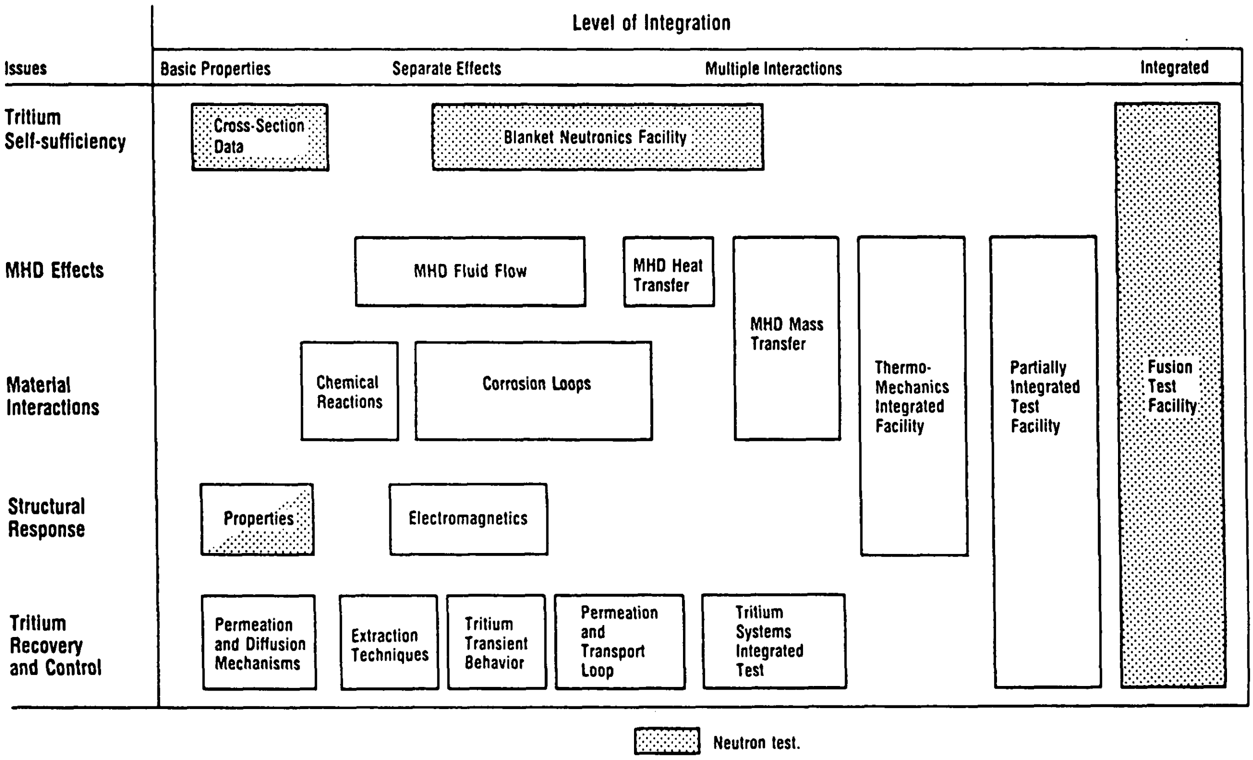

FIG. 7.1. Types of experiments and facilities for liquid metal blankets. Details in Ref. [1]. *Some experiments or facilities already exist.*

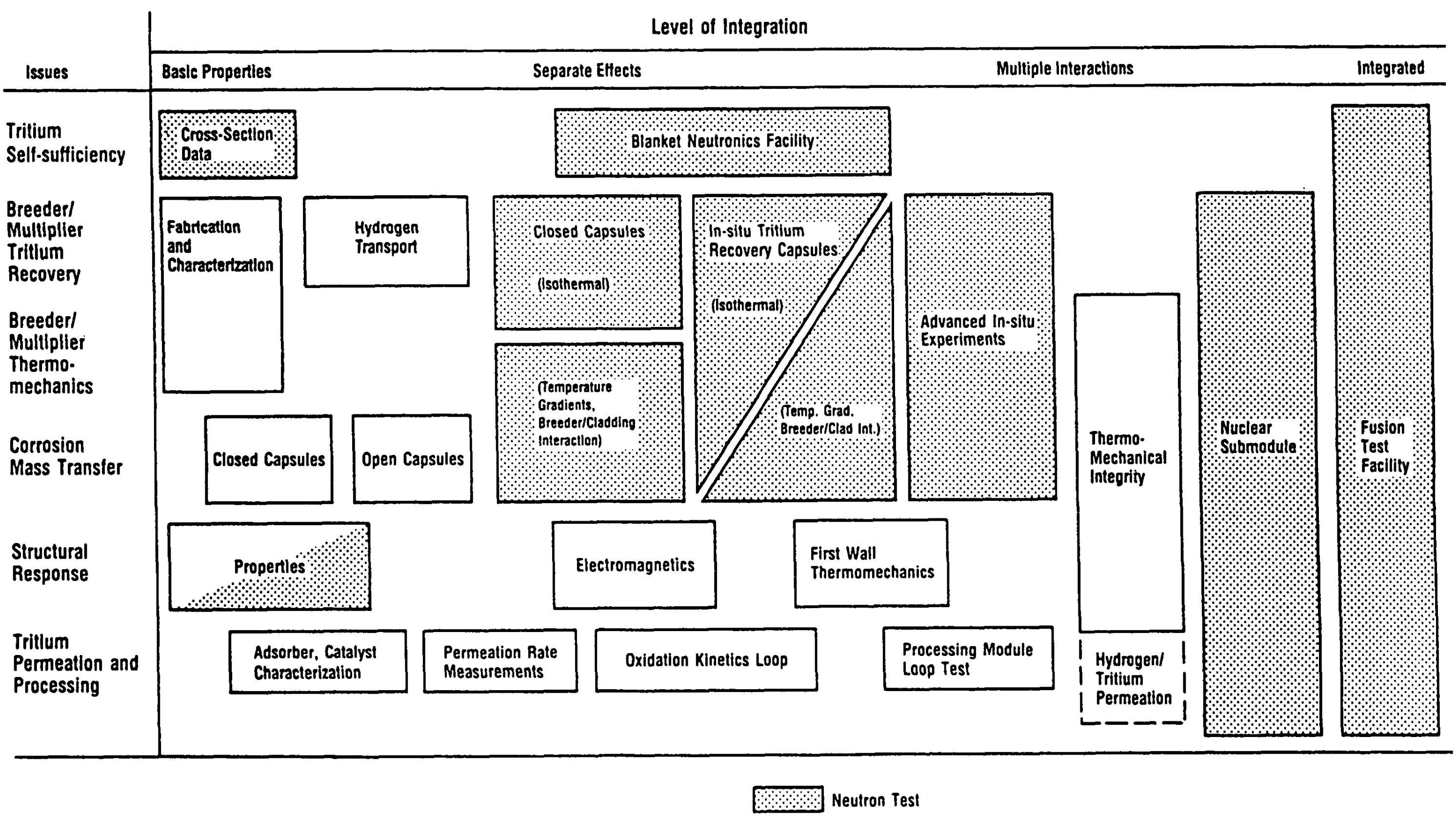

FIG. 7.2. Types of experiments and facilities for solid breeder blankets. Details in Ref. [1]. *Some experiments or facilities already exist.*

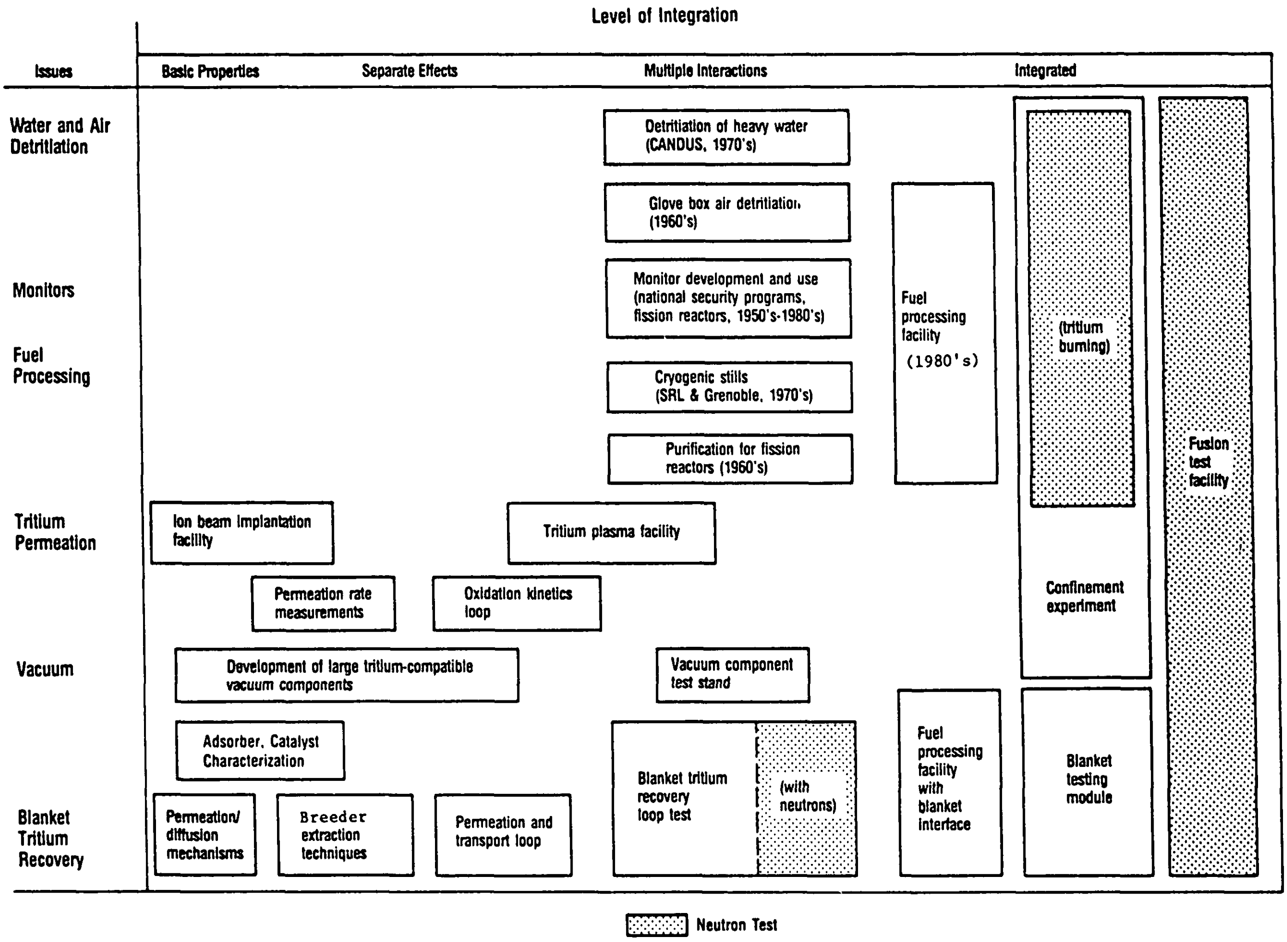

FIG. 7.3. Types of experiments and facilities for tritium processing and vacuum systems. Details in Ref. [1]. *Some experiments or facilities already exist.*

7.1.3.3. Modelling

Experiments can never cover all possible ranges of relevant parameters. Modelling is required to interpolate between and extrapolate beyond available experimental data. This is especially true in the field of safety because many conceivable transient conditions would be extremely and prohibitively expensive to examine experimentally. Rather, the approach is to use data and theory to develop models of phenomena of interest. Ultimately, resulting models should be validated against an independent data set, hopefully with new conditions.

Since 1981, the progress in safety related modelling has been quite impressive, dwarfing pre-1981 fusion safety modelling. The overall progress and current level of modelling related to safety are summarized in Table 7-XI.

A comparison of data and modelling is interesting. Early fission research was data-rich and modelling-poor because experiments were less expensive than today and computer capabilities were poor. Now, the situation is often reversed: experiments of complex transient phenomena are often expensive while the computing power is much greater. Thus, some areas are data-poor and model-rich. In these cases, the model complexity has significantly exceeded the database and more experiments are badly needed. Also, although such models are extremely important and useful, there is some tendency to place too much faith in unvalidated models.

Comparison of Tables 7-X and 7-XI indicates that some areas are particularly data-poor and model-rich, e.g. tritium inventory, permeation, and accidental release phenomena. A key problem in tritium behaviour is extrapolation to the very low tritium partial pressure (P_{T_2}) thought to be required in some systems to limit tritium effluents to acceptable levels. For example, permeation dependence is thought to favourably change from $P_{T_2}^{1/2}$ to P_{T_2} at some partial pressure for given material, temperature, etc. However, tests at relevant conditions are very difficult to perform. The result is at least an order of magnitude uncertainty in some systems in the calculated tritium permeation. One is generally forced to be conservative, i.e. assume $P_{T_2}^{1/2}$ dependence, even though this is likely to be overconservative in many cases.

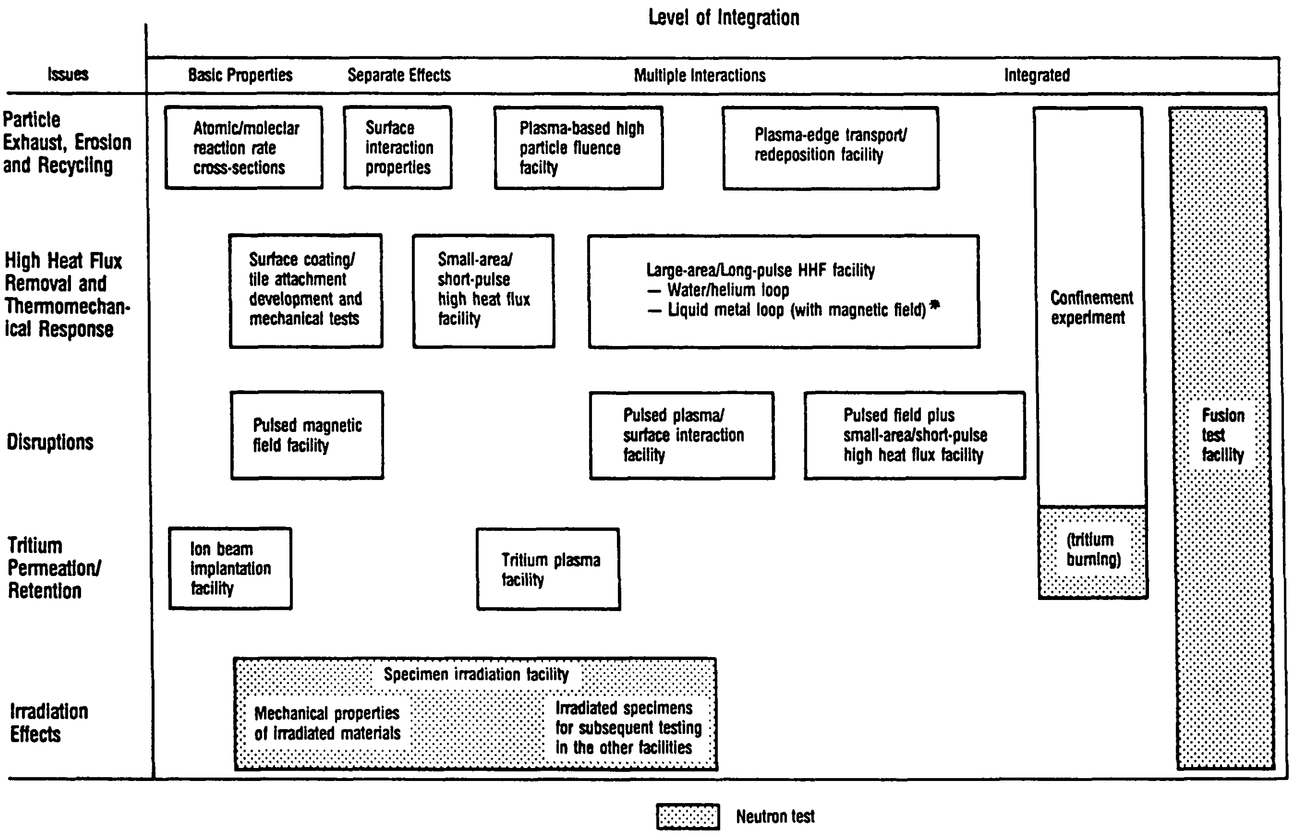

FIG. 7.4. Types of experiments and facilities for plasma interactive components. Details in Ref. [1]. *Some experiments or facilities already exist.* *Details of MHD experiments are given in Fig. 7.1.*

7.2. Issues and R and D needs

There has been considerable progress over the past several years in understanding the key technical issues for fusion nuclear technology and materials. This understanding has evolved from design and system studies summarized in previous sections. The FINESSE study [1], carried out in the USA with participation of specialists from Canada, Europe and Japan, has analysed and quantified the important issues and defined the major experiments and facilities required to resolve these issues. These results were discussed in the workshop. Similar effort has been reported to be presently underway in the USSR [2]. There is general agreement in the fusion community on the generic critical issues and on the types of experiments and facilities required for fusion nuclear technology. The workshop participants adopted the FINESSE summary figures [1], which are discussed below for the (a) blanket/first wall, (b) tritium processing, and (c) plasma interactive components.

Blankets of interest to the international fusion community can be broadly classified into liquid breeder and solid breeder blankets. The generic critical issues and the types of required experiments and facilities are summarized in Fig. 7.1 for liquid metal and in Fig. 7.2 for solid breeder blankets. The critical issues are given in the left hand columns of both figures. The types of experiments and facilities are shown as a function of the level of integration, which refers to the degree of simulation provided in the experiments for the environmental conditions (e.g. bulk heating, magnetic field, neutrons), and the physical elements in the component simulated (e.g. breeder, clad, coolant, tritium purge). Separate and multiple effect experiments provide basic property data and exploration of phenomena. These experiments can be performed in non-fusion facilities. Fully integrated tests can be performed only in fusion facilities. It is noted that the liquid metal issues are dominated by MHD and material interaction effects which require non-neutron test stands. In contrast, many of the key solid breeder issues, such as tritium recovery and thermomechanics, depend greatly on the thermal conditions and thus required simulation of bulk heating in the experiments. Fission reactors are the only practical means presently available to provide bulk heating and tritium simulation.

The key issues and the types of experiments and facilities required for the tritium processing and vacuum systems are given in Fig. 7.3. Experiments and facilities for extraction of tritium bred in the blanket from a fluid carrier are also indicated. It should be noted that a fuel processing facility is now operational in the USA, and others are being constructed or planned in other countries. These facilities have a high level of integration and are sufficient to provide the database on tritium processing required for construction of a tritium burning fusion device.

The key issues and types of experiments and facilities required for plasma interactive components (PIC) are summarized in Fig. 7.4. This area has strong coupling to plasma physics and nuclear technology. The appropriate facilities include plasma confinement-type devices and non-confinement test stands. Many of the high heat flux and thermomechanical issues relate to a number of issues in the blanket. It is most effective to consider facilities where these common issues for PIC and blanket can be simultaneously addressed. Clear examples are the MHD effects for liquid metal coolants for PIC and for blankets.

Effect of irradiation on materials is a key issue that has been delineated in previous studies and workshops. The effect of irradiation on materials in the blanket and PIC have been covered in Figs 7.1 to 7.4. The effect of irradiation on structural materials requires special attention. Fission reactors and ion simulation facilities are now used to obtain data on the behaviour of materials under irradiation. In addition to the 'specimen-type' testing now being carried out, the workshop participants have agreed on the importance of larger volume tests in which the interactions among structural material and other elements of the blanket are simulated. Data on failure modes, particularly those that occur in the component early life, are of importance to component designers.

7.3. Recommendations

The recommendations of the Fusion Nuclear Technology Workshop are as follows:

(1) An enhanced effort is required in the fusion nuclear technology (FNT) experimental programme. Key reasons for this recommendation are:
   (a) Current conceptual design studies are severely restrained by inadequacies in the present database.
   (b) The economic and safety attractiveness of fusion can only be demonstrated by improved concepts that will depend on new data.
   (c) Experimental programmes have long lead times.
   In addition, experimental programmes should be supported by concerted theory/modelling efforts.
(2) The initiation and planning of experiments and conceptual design studies should be undertaken in such a way that a balanced and broadly based programme is achieved. This is necessary in order to ensure that fusion nuclear technology development is not impeded by a lack of data in one or a small number of key areas and that resources are fully used in expanding the technology across a broad front.
(3) Close co-operation and coupling between experimental programmes and design studies and between physics and technology related areas should be maintained. This has the dual advantage of encouraging feedback between experiment and design and allowing the maximum use of data generated in current physics experiments in addressing technology issues.
(4) A group for co-ordinating international activities on fusion nuclear technology should be formed. There are particularly strong incentives for pursuing international co-operation on fusion nuclear technology. Among these reasons are:
   (a) There are many areas of key R and D needs for FNT which are of common interest to all countries, and which constitute opportunities for international co-operation.
   (b) Substantial resources in terms of manpower and facilities are required to resolve key FNT issues. International co-operation is thus desirable as a cost-effective, and in some cases necessary, means for conducting the R and D required in many areas of FNT.
   (c) International co-operation provides an excellent mechanism to accelerate progress and enhance the prospects for success in the development of credible and attractive fusion nuclear components. Effective co-ordination of intellectual and hardware resources in the world programmes will permit more complete and faster exploration of promising options, identification of critical problems and development of attractive solutions.

Forming an international co-ordinating group for fusion nuclear technology is believed to be an important step in implementing international co-operation on FNT. The function of such a group and examples of its activities are:

(a) initiate job planning of FNT R and D, identify issues and experiments of common interest;
(b) suggest mechanisms for implementing co-operation, including plans for shared facilities, joint experiments and exchange of information;
(c) periodical review of technical status and progress, recommend action;
(d) develop common technical requirements for performance of materials and components;
(e) co-ordinate technical meetings to address specific topics and/or to facilitate information exchange among researchers in the field;
(f) investigate the need and develop format and mechanisms for an international databank.

REFERENCES

[1] ABDOU, M., et al., Technical Issues and Requirements of Experiments and Facilities for Fusion Nuclear Technology: FINESSE Phase I Report, University of California, Los Angeles, Rep. PPG-909, UCLA-ENG-85-39 (December 1985).

[2] GLUKHIKH, V.A., "The USSR Fusion Technology Programme", IAEA Technical Committee Meeting on Fusion Design and Technology, Yalta, USSR, (26 May–6 June, 1986).

8. HYBRID FUSION–FISSION REACTORS
(V.V. Orlov)

8.1. Status

Since the last Technical Committee (Tokyo, 1981), hybrid reactor studies have been carried out mainly in the USA and the USSR.

The major effort in the USA has been directed towards comprehensive design studies of suppressed fission blankets for various types of fusion reactors.

In the USSR, along with the investigation of hybrid reactor concepts based on tokamaks, mirror machines and laser systems, the engineering design of the experimental power reactor (OTR) has been carried out. Major attention in these studies was paid to the fast ^{238}U blanket options, including those with the direct fuel enrichment for fission reactors. In 1985, a Soviet-American fission–fusion seminar took place in Moscow.

A discussion on the role of hybrids in the fusion programme is being continued both within and outside the fusion community. There are different viewpoints on this issue depending on the diverse judgements on nuclear power development in various countries, on the one hand, and on the ways towards the final goals of fusion and their achievement in the nearest future, on the other hand.

The ultimate goal of fusion development is to provide an alternative to fission reactors while incorporating new and cheap energy sources into the fuel balance. It should, however, be stressed that these goals have to be pursued with regard to economics because the power production scale inevitably makes the economic factor (with proper regard to safety) the decisive one for the selection of power production methods.

Economy puts a high barrier in front of any new power technology. To overcome it, a large 'strength margin' and long time are needed. For fusion reactors, this barrier is particularly high since they depend on new technologies much more strongly than fission reactors.

Historical experience on power engineering, including nuclear power, shows the extreme inertia of the energy system as far as new technologies are concerned. Now, 30 years after the startup of the first nuclear power plant, the nuclear fuel share in the world fuel balance amounts only to about 3% and in the world electricity production to some 12%. On the other hand, this experience has shown that power engineering is moving towards its final goal always step by step.

As long ago as in the 1960s some people called for nuclear power engineering development on the basis of fast breeders that could immediately have solved the fuel problem for many years to come. However, the first and rather long stage of nuclear power development was based on the simpler and cheaper LWRs.

The present physical concepts of fusion reactors as well as the material available rely only on the feasibility of the D-T reaction, with tritium and neutron involved in this reaction. This reaction can be realized in tokamaks, mirror machines, with an $n\tau_e$ value that is insufficient for self-sustained burning, or laser inertial confinement reactors with low driver efficiency. The neutron load to the first wall in such reactors is $P_n \sim 1\ MW \cdot m^{-2}$. High input power and structural materials activation by neutrons are to be dealt with, as well. According to many estimates, such engineering and physical parameters are not sufficient for the realization of an economically competitive fusion reactor with sufficiently low radioactivity. However, there are ideas that hold out hopes for much better designs in the future.

The hybrid reactor with high fissile fuel production rate and energy multiplication in the blanket can reach the level of economic competitiveness with physical and engineering parameters attainable in the nearest future. This is why the hybrid may be considered to be the first natural step to practical fusion employment and, at the same time, an intermediate stage on the way towards pure fusion reactors.

Recent nuclear power engineering economy and the projected situation on the uranium market have resulted in a lower rate of nuclear power plant construction and, consequently, in a reduction of the uranium price. This situation is not favourable for activities connected with fuel breeding and, in particular, not for hybrids. It is, however, out of place to predict the fusion programme on the basis of the present situation only. The goal of fusion is to provide energy for the next century. One must take into account the high activity in the development of nuclear power engineering in France, USSR, FRG, USA, Japan, and some other countries. Also, there is a probability that this activity will grow in the USA at the end of this century and that the interest in nuclear power will be intensified in many developing countries. According to some conservative evaluations, the total installed electric capacity of nuclear power plants in the world near the year 2000 will be about 700 GW(e), and then will rise at a rate of about 50 GW(e) per year. In this case, the price of uranium will even rise at the beginning of the next century. The estimated world resources of relatively cheap uranium (5 million tonnes and up to 20 million tonnes including expected but so far undiscovered resources) will be used up by 2020 to 2040. A reduction of the uranium consumption by a factor of two can be achieved by using a closed fuel cycle in LWRs and with their improvement (the conversion ratio being increased up to 0.8). However, a uranium deficit is predicted to be inevitable in the first half of the next century, in some countries earlier, in others later. It is, therefore, reasonable to consider once again the logic of fusion–fission hybrids from this point of view.

8.2. Progress in the development of hybrids

8.2.1. Major studies

The OTR reference option is under development with superconducting magnets (Nb_3Sn) and a single null magnetic divertor. As a coolant, gas (He), water and liquid metal were considered.

The fusion power of the reactor is 520 MW, the magnetic field 5.8 T, and the major radius is 6.2 m.

Simultaneously, the conceptual design of a commercial hybrid reactor tokamak with $P = 1.3 - 1.8$ GW(e) is under consideration.

Other activities on the development of fusion–fission hybrids include:

(a) TROL-2. This design was developed at the High Temperature Institute, Moscow. (Reference: A.V. Nedospasov, V.L. Lokshin, N.N. Vasiliev, A Hybrid Tandem Mirror Reactor, TROL-2 – Report on the Soviet-American Fusion-Fission Reactor Workshop, Moscow, April, 1985.) The depleted uranium blanket is cooled by boiling water in tubes with one bank of the tubes used for superheating the water. The energy multiplication is of the order of ten in the fast fission blanket. An efficiency of about 25% is obtained even with a plasma Q value of only two. This design is based on so well established a technology in nuclear power engineering that its feasibility is unquestioned.

(b) Pebble Bed Fast Fission Tokamak Hybrid – This design established by a multilaboratory American team is based on metallic uranium pebbles cooled by helium. [Reference: D.L. Jassby, et al., "Fast fission tokamak breeder reactor", J. Fusion Energy (1986) (in press)]. The use of pebbles allows removing the uranium. Feasibility issues to be addressed are:

(i) adequate heat transfer
(ii) self welding of uranium pebbles
(iii) possible need for coatings
(iv) safety

(c) Molten Salt, Fission Suppressed Hybrid. [Reference: R.W. Moir, et al., Design of a Helium-Cooled Molten-Salt Fission Breeder, Fusion Technol. 8 (1945) 465.] This design uses a bed of beryllium pebbles cooled by helium with molten salt flowing slowly through 316 type stainless steel tubes in the bed. Tritium, in the form of T_2, is continuously removed as a gas from the molten salt.

The technical issues are:

(i) development of a tritium permeation barrier and demonstration of tritium removal from helium;
(ii) experimental verification of predicted neutron multiplication in 20 cm full density equivalent of beryllium to reduce uncertainty in breeding ratio;
(iii) experimental data are needed on beryllium radiation damage;
(iv) experimental tests to see if beryllium self-welding and beryllium welding to steel will prevent pebbles from flowing. If the bed does not flow swelling could cause heavy blanket failure.

(d) Li-Cooled Fission Suppressed Tandem Mirror Hybrid. This design was carried out over several years by a multilaboratory team. [Reference: D.H. Berwold, et al., "Updated Reference Design of a Liquid Metal Cooled Tandem Mirror Fusion Breeder", Fusion Technol. (1986) (in press).]

The fusion power is 2600 MW, the average power is 5075 MW. The neutron wall loading is 1.7 $MW \cdot m^{-2}$. The cost is estimated to be 2.3 times that of an LWR and produces ^{233}U at a breakeven cost of U_3O_8 of 100 US$ per pound. The technical issues are:

(i) experimental verification of MHD pressure drop across the bed due to the flow of liquid lithium;
(ii) experiments to show if the beryllium pebbles will self-weld, preventing flow needed for removal of thorium. Welding of beryllium to thorium and to steel in the lithium environment are also an important issue;
(iii) experimental studies of radiation damage to determine pebble lifetime;
(iv) experimental verification of neutron multiplication;
(v) development and demonstration of ^{233}U removal from thorium metal by magnesium dissolution process.

(e) ICF Fission Suppressed Hybrid. Lawrence Livermore National Laboratory and GA Technologies designed a ^{233}U producing hybrid using beryllium columns to multiply neutrons (see reports by W. Meiser et al.). Falling lithium is used to absorb the microexplosion blast. Lithium flows over the beryllium columns and thorium fuel elements. The breeding ratio or gain is somewhat more than for magnetic fusion designs, and the cost appears less dependent on the driver cost.

(f) The feasibility of applying the hybrid fusion–fission reactor blanket to direct enrichment of fuel elements made of depleted uranium or thorium was studied in the Soviet Union. It was shown that fuel elements enriched up to the necessary concentration of fissile isotopes within the hybrid blanket may be transferred into the core of a thermal reactor without additional processing.

For heavy water or graphite thermal reactors with good neutron balance, ready for operation with natural uranium, it is sufficient to accumulate 3 to 4 kg of Pu per tonne of uranium to have fuel applicable for such reactors with a burnup of up to 6 $MW\ day \cdot kg^{-1}$ and higher. The number of supported reactors for such a system can be about n = 10 or even higher. It gives some grounds to expect acceptable economic parameters of the hybrid fission system at rather high cost of the hybrid reactor.

(g) Direct Enrichment in Fission Suppressed Blanket. It has been shown by E.T. Cheng in the paper by R.W. Moir et al., "Fusion–Fission Hybrid Studies in the United States" at this meeting in Yalta that direct enrichment is possible in a fission suppressed blanket design using beryllium as neutron multiplier. Thorium oxide or uranium oxide microspheres (~1 mm dia.) are suspended and circulated in $Li_{17}Pb_{83}$ flowing in tubes through the beryllium blanket.

8.2.2. Highlights during last five years of hybrid work

(i) Significant design progress on fast fission tokamak hybrid OTR.
(ii) The importance of resonant self-shielding for suppressed fission blanket was widely recognized.
(iii) The use of pyrochemical reprocessing was appreciated even for fast fission because it lowers the reprocessing costs thus allowing low discharge enrichment and low power swing.
(iv) Fission suppressed designs were developed in detail, based on beryllium as neutron multiplier.
(v) Fast fission designs were developed in great detail based on water cooling. Clear feasibility of this design is established (TROL-2).
(vi) Improvements based on uranium in pebble form were identified which enhance the safety of fast fission blankets. Feasibility of direct enrichment of fissile fuel for fission reactors was studied.
(vii) A new concept of a blanket cooled by eutectic material was developed.

8.3. Problems

In addition to the problems which are common to all fusion reactors and to those mentioned above in connection with particular designs, the following problems are most important for further efforts in hybrid development:

(1) LOCA-type accident analysis.
(2) Technological and MHD issues of the blanket cooled by liquid metal.
(3) Fuel element performance under cycling loads (for tokamaks and inertial confinement systems).
(4) Rod-type fuel elements reloading.
(5) Exposed fuel isotopic composition analysis with respect to non-proliferation.

9. CONCLUDING REMARKS
(J. Kupitz)

The participants of the Fourth Technical Committee Meeting and Workshop on Fusion Reactor Design and Technology in Yalta, USSR, May 26 – June 6, 1986, considered the technical information presented and discussed at the meeting to be very important and useful for all the countries involved in the development and design of fusion reactors.

Since the last meeting in Tokyo in 1981, there has been substantial progress in the construction and operation of new large fusion machines and outstanding design studies have been performed for the next step in tokamak technology and for the development of the technology in non-fusion facilities. However, there has been a noticeable reduction in the number of reactor designs and long term fusion applications.

While the time between the previous meetings has been progressively increasing up to five years, it was recommended that the next meeting of this kind should take place within three years, when further key results are expected from the large fusion devices which were put into operation recently, and when the definition processes of the planned near term reactors are expected to approach their final stage. Specific recommendations for further implementing international collaboration in various areas are included in the pertinent sections. It would, therefore, seem appropriate to bring together fusion reactor designers and those who are involved in experimental work on the large machines.

The effort and resources committed to fusion have reached a level where prudent use of all resources is necessary and international co-operation highly desirable. The participants felt that regular international information exchange on the results of fusion reactor design and on the status of fusion technology would provide an essential contribution to the successful development of fusion reactors. The IAEA, as the only global international organization dealing with the development of nuclear power, has always been the international forum for information exchange and for the promotion of international co-operation. The participants recommend the IAEA to continue organizing specialists meetings and technical committee meetings on aspects of fusion reactor design and technology, including fusion safety which is of increasing worldwide interest.

The participants expressed the opinion that an international tokamak fusion reactor project, the proposal for which is presently being discussed at the governmental level, would be highly desirable as an important means to demonstrate the feasibility of fusion reactor technology. The fact that all four major fusion programmes in the world (USA, USSR, CEC and Japan) project a Tokamak Reactor (NTTR) as the next step, will greatly facilitate international co-operation. The INTOR workshop series, which is hosted by the Agency, has repeatedly proved the usefulness of such co-operation. At the same time, a joint fusion reactor project would be an important contribution to the improvement of international relations and peace in the world.