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TIBER II/ETR Final Design Report

Volume 3

**5.0 Radiation Safety and Environment
6.0 Physics and Technology R&D Needs**

Technical Editor: J. D. Lee

Lawrence Livermore National Laboratory (LLNL)
Argonne National Laboratory (ANL)
Canadian Fusion Fuels Technology Project (CFFTP)
Fusion Engineering Design Center (FEDC)
Fusion Technology Institute - University of Wisconsin - Madison
GA Technologies (GA)
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University of California - Los Angeles
University of Illinois

September 1987

University of California

**Lawrence Livermore
National Laboratory**

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Volume 3

5.0 Radiation Safety and Environment 6.0 Physics and Technology R&D Needs

Technical Editor: J. D. Lee

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5.0 RADIATION SAFETY AND ENVIRONMENT

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5.0 RADIATION SAFETY AND ENVIRONMENT

The safety and environmental (S&E) objectives for TIBER II are:

1. Improve the design to increase the likelihood that operation of a TIBER-like machine would (a) have negligible health and environmental impact and (b) fulfill its research mission--with minimum cost impact.
2. Make TIBER an appropriate step toward fulfilling fusion's highest ultimate S&E potential.
3. Set precedents toward a rational system of S&E licensing approvals for fusion energy.

From highest to lowest priority, safety considerations aim to protect (a) the public, (b) plant workers, and (c) the facility. This is generally accomplished by focusing on the reverse order: i.e., proper design first protects the machine from potential off-normal events. Such machines are "fault tolerant" and may achieve passive safety. Active emergency safety systems are added only as a last resort.

The TIBER S&E approach has been to:

1. Derive appropriate TIBER design targets from possible requirements,[1]
2. Participate in the design effort to help meet the targets, and
3. Assess the final design, comparing design parameters with the initial target.

The design must satisfy requirements, which are either appropriate legal limits, e.g. U.S. Environmental Protection Agency (EPA) regulatory requirements, or possible future regulations appropriate for fusion. The design should satisfy targets; however, targets are changeable. The TIBER II design does not meet some of the initial design targets, as explained below. Future design efforts should improve the design in these areas--unless future cost/benefit examination of the design target shows a need to change it.

TIBER II resources for S&E analyses were quite limited. Accordingly, efforts were focused on high-leverage design features:

1. Analysis of innovative TIBER features to identify major safety impacts (good or bad).
2. Ideas that could significantly improve the S&E characteristics of ETR-class machines.
3. Areas where new data or modeling may significantly change results from past design studies.

These items are listed in Table 5.0-1, along with the level of safety analyses in TIBER II.

It was possible to explore potential issues only far enough to determine if the concern was serious enough to impact design. Analyses were terminated when (a) a concern was eliminated by simple calculation, (b) a possible solution was identified that would eliminate the concern, or (c) it became clear that more detailed analyses were necessary for resolution. It must be recognized that the possible solutions may not be optimal; they are offered in the spirit that their existence provides a way around a potential concern, either by use of that solution or some other one. Of course, if the solution is easy to implement, such implementation is cost effective compared with searching for other solutions or performing complex, difficult-to-defend calculations that might show that a solution was not actually needed.

TIBER parameters were compared with the requirements and targets. No S&E problem has been uncovered in the design that would prohibit meeting probable S&E requirements. However, serious concerns do remain that could impact the viability of various TIBER design concepts and choices. The reader primarily interested in conclusions should refer to Sec. 5.7.

Analyses are sufficient to conclude that TIBER has some potential for "passive safety," i.e. it may be possible that the public could be protected against all severe accidents (offsite dose < 2 Sv (200 rem)) without depending on active safety systems. See Refs. [2,3] for discussion of "passive safety." More design details and analyses are required to firmly determine whether existing passive features are adequate. Three particular items could be problems--(a) demonstrating a rapid, passive, benign way to shut off the plasma when needed, (b) removal of decay heat from the tungsten inboard shield

Table 5.0-1 High-leverage safety and environmental design features.

High-leverage design feature	Associated issues	Level of S&E analyses
Steady-state operation	Increased reliability Increased neutron fluence	Nil Included in neutronics
Design for multiple physics operating points	Increased design margins for most operating points	Nil
Aggressive superconducting magnet damage limits	Increased chance of magnet failure	Nil, coils must stay in compression
Passive safety design target	Seek passive solutions rather than active safety systems	Moderate
Minimum major radius (Reduced total power)	Use of tungsten in shield increases decay heat	Moderate
Testing of lithium and LiPb modules in a water environment	Lithium-water reactions LiPb-water reactions Lithium-air reactions Hydrogen gas control	Moderate Moderate Moderate Low
Aqueous salt breeding in shield coolant water	Large tritium inventory in water C-14 production in water	Moderate Low
Tritium self-sufficiency	Large beryllium inventory	Low
Filtered, vented confinement building	Adequacy of radioactivity control Building overpressure limits	Moderate Low

if coolant is lost, and (c) possible chemical interactions between liquid-metal test modules and the water coolant in the shield and divertors.

Accident pathways do appear to exist that violate the design target to limit offsite doses from any accident below 10 mSv (1 rem).

TIBER II should be able to meet all effluent and maintenance requirements and targets. However, recovery from off-normal events depends on fully-remote maintenance. Therefore, designers must be able to anticipate all possible failure states of components to allow repair. Adequate control of tritium effluents will require that the shield coolant system be nuclear grade with high leak tightness.

TIBER II should be able to meet waste management and decommissioning requirements. However, the design target to produce no high-level waste is not met. Adding lithium nitrate to the shield water to make tritium produces C-14 sufficient to cause the wastes from coolant processing to be high-level wastes when compared with 10CFR61 [4] (10CFR61 defines limitations on low-level waste disposal in the U.S.). The basic TIBER II structure may qualify as low-level waste, depending on interpretation of 10CFR61 and how the components are processed for disposal.

The following sections detail:

- 5.1 Safety and environmental requirements and design targets,
- 5.2 Accident analyses,
- 5.3 Personnel safety and maintenance exposure,
- 5.4 Effluent control,
- 5.5 Waste management and decommissioning,
- 5.6 Safety considerations in building design, and
- 5.7 Safety and environmental conclusions and recommendations.

5.1 SAFETY AND ENVIRONMENTAL REQUIREMENTS AND DESIGN TARGETS

Identifying S&E requirements for commercial fusion plants helps set (a) precedents for evolution of a rational regulatory system and (b) TIBER design targets so that TIBER is an appropriate step toward attractive, competitive fusion energy. Possible requirements for commercial power plants and for ETR-class machines are both relevant in determining TIBER design targets.[1]

Design targets are derived from requirements by incorporating a design margin against violating a requirement and by consideration of the As Low As Reasonably Achievable (ALARA) principle. Where meeting a target appears especially difficult, more in-depth analysis of cost versus benefit, as implied in the ALARA principle, is warranted in future studies to determine if a better design target exists.

5.1.1 POSSIBLE TIBER REQUIREMENTS

One cannot know what regulatory requirements will be imposed for an ETR. These requirements may be country-dependent. Certain requirements would likely be imposed, e.g., EPA effluent limits. Others are less clear. TIBER requirements are therefore divided into "probable" and "possible."

Probable legal requirements:

1. The public will be protected against the consequences of any credible accident such that the maximum offsite 2-hour dose is 250 mSv (25 rem). (Could take credit for active systems and confinement building.)
2. The maximum radiological exposure to plant workers must be below 50 mSv/yr (5 rem/yr).
3. Public individual exposure from routine atmospheric effluents must be below 250 μ Sv/yr (25 mrem/yr).
4. Public individual exposure from routine surface water effluents must be below 40 μ Sv/yr (4 mrem/yr).
5. There must be a plan for handling waste and decommissioning the facility.
6. Radiation exposures must be As Low As Reasonably Achievable (ALARA).

Possible requirements (very difficult to quantify):

7. Individual members of the public will be provided a level of protection against the consequences of fusion facility operation such that no individual bears significant additional risk (0.1%) to life or health.
8. Plant workers will be provided a level of protection while at work such that workers will be safer at work than away from work. (Worker risk is below 10% of background.)
9. The facility will be protected against off-normal events such that there is only a small chance (1%) that off-normal events require abandoning the facility and its research mission.

5.1.2 REFERENCE SITE DEFINITION

Translating requirements into useful design targets is complicated by issues such as (a) appropriate design margins to assure meeting the requirement and (b) site characteristics. The latter relates curies-released to public doses. Key reference site parameters include:[1]

- 1-km site boundary,
- Effluents released from 50-m stack,
- Cooling tower for waste heat, and
- Seismic zone II, average U.S. case.

For routine effluents, the maximally-exposed individual among the public will therefore receive about $0.5 \mu\text{Sv/yr}$ ($50 \mu\text{rem/yr}$) per $\text{Ci-H}^3/\text{day}$. Other reasonable site parameters could raise or lower this by a factor of 3.[1] The societal dose commitment is assumed to be $100 \text{ person-mSv/yr}$ per Ci/day (10 man-rem/yr per $\text{Ci-H}^3/\text{day}$).

For accidents, the worst weather and release characteristics are taken to be 1-km site boundary, ground level release, 1-m/s wind speed, and Pasquill stability class F. Taking some credit for building wake effects suggests that the maximally-exposed member of the public will receive a dose of about $50 \text{ nSv/Ci-H}^3\text{-released}$ ($0.05 \text{ rem/g-H}^3\text{-released}$). Again, the value is uncertain by a factor of 3, depending on weather and site parameters.[1]

No detailed calculations were performed for external events (seismic, aircraft impact, winds, floods) and so no design-basis return frequencies or magnitudes were defined for them.

5.1.3 TIBER DESIGN TARGETS

TIBER II resources were insufficient to adequately explore the impact of the targets on all design aspects. Simple, conservative estimates were often necessary to provide initial quantitative guidance to the design process. The TIBER II design will guide future ETR designs as to the difficulty of meeting S&E targets, the optimum partition of effluents and risk among systems, and the success of some design approaches--just as TIBER builds on past studies.

An important issue is the maximum offsite dose from credible accidents. Recent fusion conceptual power plant studies have adopted 2-hour dose targets of 10 mSv (MINIMARS [5]) and 10-50 mSv (TITAN [6]). The Compact Ignition Tokamak (CIT) maximum dose target is 50 mSv.[7] The TIBER design target is 10 mSv (1 rem), depending only on passive safety features and not depending on a containment/confinement building to limit the maximum release. This leads to a maximum release target for tritium oxide of 20 g H^3 (200 kCi) or for elemental tritium of 200 g (2 MCi). Comparison with existing regulations [8] indicates that meeting this 10-mSv design target would eliminate any need to have a public evacuation plan.

Future accident requirements are likely to include some limit on probabilistic risk to the public. In lieu of attempting probabilistic risk assessments (PRA) in the current TIBER effort, off-normal events are classified by estimated frequency of occurrence, see Table 5.1-1. For each frequency regime, there are design targets for limiting public dose, worker dose, and facility damage. This is the same basic approach as for CIT.[7] Although one cannot currently quantify all failure frequencies, one can conservatively classify initiators and transients into frequency regimes. Detailed design and accident analyses would determine which regime various transients actually fall into.

Figure 5.1-1 shows the design targets for each accident frequency regime for limiting the maximum offsite individual dose. For all credible events, the offsite dose should be below 10 mSv. For higher frequency events, the offsite doses should be lower. The design targets should be reached without

Table 5.1-1 Accident frequency regimes and TIBER examples.

Frequency regime	Approximate ^a frequency range	Initiator examples
Anticipated	$1-10^{-2}$	Vacuum boundary failure Plasma disruptions Offsite power loss Inside of test module fails Minor LOCA and LOFA transients ^b Neutral beam firing when plasma not present ^c Pellet injector firing when plasma not present ^c
Low probability	$10^{-2}-10^{-4}$	Massive failure of plasma control system Major LOCA and LOFA transients Shell of test module fails (note d, below)
Very low probability	$10^{-4}-10^{-6}$	Worst case external events (seismic, winds, aircraft impact) ^e
Incredible	$< 10^{-6}$	Meteor strikes

a. See Ref. [10] for discussion of some failure frequencies.

b. LOCA = loss of coolant accident, LOFA = loss of flow accident.

c. Use of interlock system reduces frequency, perhaps enough to move event into "low probability" regime.

d. Assumes double-walled test modules.

e. Actual magnitude of external events dependent on site characteristics and national regulations. Seismic events and aircraft impact are more worrisome since they occur without warning.

resort to either active isolation of the confinement building or active emergency safety systems.

Figure 5.1-1 shows the 0.1% background risk curves for cancer fatality and acute accidents as well as the passive safety consequence limit of 2 Sv. Meeting this passive safety consequence limit insures no prompt public fatalities without depending on active safety systems.[2,3] Note, the line for 0.1% background risk represents risk from all contributors; the design target line should be applied to each transient. Accidents should be considered from both the upper-bound consequence and PRA standpoints.

Figure 5.1-1 TIBER design targets for limiting public risk from accidents and routine effluents.

Table 5.1-2 Summary of key safety and environmental design targets.

Condition	Maximum public dose (mSv) ^a		Maximum worker dose (mSv) ^a		Maximum facility damage
	Probably required	Design target	Required	Design target	
Routine operation and maintenance	< 0.25	< 0.05	< 50	< 10	None, continue operation
Accidents ^b :					
Anticipated (10^{-2} /yr)		< 0.1		< 10	Continue operation; downtime < 1 month
Low probability (10^{-2} - 10^{-4} /yr)		< 1		< 100	Limit damage to 1 major system; downtime < 6 month
Very low probability facility; (10^{-4} - 10^{-6} /yr)	< 250	< 10 (c)		< 1000	Can repair downtime < 2 years
Incredible ($< 10^{-6}$ /yr)		> 10		> 1000	

a. 1 mSv = 100 mrem.

b. The accident design targets should be met without depending on active systems or taking credit for isolation, via active means, of the confinement building.

c. 10 mSv = 1 rem.

The figure also shows routine effluent limits: the point representing EPA regulations [9] for risk from atmospheric effluents and the point representing the TIBER design target for effluents. The design targets are summarized in Tables 5.1-2 and 5.1-3.

Table 5.1-3 Additional environmental design targets.

Maintenance:

Maximum worker dose < 10 mSv/yr (1 rem/yr)

Radiation zone for planned maintenance < 25 μ Sv/h (2.5 mrem/h) at 24 hours after shutdown. There is no planned maintenance inside reactor hall.

Radiation zone for contingency maintenance 0.1-1 mSv/h (10-100 mrem/h) at 4 weeks after shutdown. Currently, no areas for contingency maintenance have been identified.

Radiation zone outside reactor hall < 5 μ Sv/h (0.5 mrem/h) during tokamak operation.

Effluent control:

Total atmospheric dose^a < 50 μ Sv/yr (5 mrem/yr) partitioned:

Total tritium effluents < 20 μ Sv/yr

Activated air < 10 μ Sv/yr

Direct irradiation < 10 μ Sv/yr

Fluid processing systems effluents < 10 μ Sv/yr

Tritium releases to air < 40 Ci/day

Average tritium concentration in shield cooling water < 10 Ci/liter-water^b

Waste management:

All waste should classify as either

- recyclable, i.e., contact dose < 25 μ Sv/h,
- low-level waste, i.e., meet 10CFR61 and relevant fusion extensions thereof.

-
- a. The partition among contributing pathways is somewhat arbitrary, future efforts should optimize the partition.
- b. Not a true design target; the 20 μ Sv/yr tritium effluent target leads to an average 10 Ci/liter-water value for the shield coolant.

5.2 ACCIDENT ANALYSES

TIBER accident analyses were divided into four areas:

1. Estimating the potentially releasable tritium inventory
2. Estimating the potentially releasable activation product inventory
3. Examining tritium and activation products for specific accidents involving TIBER systems, e.g., decay heat or chemical reactions
4. Examining possible consequences from accidents involving the test modules

Tritium and activation products were separated because of greatly differing mobilization characteristics; Secs. 5.2.1 and 5.2.2 summarize how much inventory (tritium, activation products) might be available for accidental release. Sections 5.2.3 and 5.2.4 focus on a limited set of accidents involving the TIBER machine and the test modules.

5.2.1 TRITIUM-DOMINATED ACCIDENTS

Table 5.2-1 summarizes the major system tritium inventories and potential for release. The TIBER design is not sufficiently detailed to provide accurate inventory estimates; many estimates are scaled from NET.[11]

The largest inventory is in the fuel storage system. At 5% burnup and 300 MW of fusion energy, 2-3 days of operation require about 2-kg-H³ throughput. Thus, this much tritium storage capacity is needed to allow operation if the fuel processing systems fail. The tritium would be stored as a metal tritide (MT) with low vulnerability in metals like titanium and uncertain vulnerability in uranium. Titanium getter storage is appropriate as backup with infrequent use. Uranium bed storage may be needed for frequent loading and unloading associated with routine operation. Active U-bed storage may be vulnerable in the event of air ingress; released tritium would likely be in the HTO form if sufficient oxygen were available.

The breeding shield has the second largest inventory. The tritium recovery system is sized to allow up to 30 Ci H³/liter-water. This peak level would only be reached during high availability (> 50%) operation extending

Table 5.2-1 TIBER tritium inventories and release potential.^a

System	Process elements	Form	Inven- tory (g)	Release potential
Torus	Gas fill	DT	0.2	High
Vacuum pumps	Mechanical pumps and piping	DT	1-5	High
Impurity removal	Palladium diffuser	DT	60	Moderate
Impurity processing	Oxidizer/trap/electrolysis	DTO	1-10	Moderate
Isotope separation	Cryogenic distillation	T2	80-100	High
Fuel delivery	Pellet injector	T2	10-50	High
Fuel storage	Metal getters	MT	1000- 2000	Very low- moderate
Water coolant:				
no salt	Primary coolant loop	HTO	0.1-10	Moderate
with salt	Primary coolant loop	HTO	100-600	Moderate
Impurity removal	Side stream ion exchange	HTO	30	Low
Shield cleanup	Distill/exchange/electrolysis	HTO	300	Moderate
Multiplier	Beryllium balls	MT	80-200	Low
Blanket test modules	Module and processing systems	HT/ HTO	10- 100	Low- high
Low-level water cleanup	Electrolysis	HTO	5	Moderate
Solid wastes	Degassing/storage	HTO/MT	0.1-10	Low
Atmosphere cleanup	Molecular sieve dryers	HTO	0.1	Moderate
Glovebox cleanup	Getters	HT	0.1	Moderate
First wall/armor	Graphite and steel	T	20	Moderate
Divertor	Tungsten	T	1-100	Moderate
Piping/walls	Steel and concrete	HT/HTO	0.1	Moderate

a. Release potential is based on engineering judgment; see text.

over weeks. The average tritium level is 10 Ci/liter at 30% availability. The peak level is about 600 g tritium in 200 m³ of primary coolant at 80°C and 0.2-0.3 MPa and 300 g in the water detritiation facility, mostly in water distillation columns at 50°C and sub-atmospheric pressure. This tritium is moderately vulnerable, given its wide distribution around the reactor as a primary coolant, but generally has low potential for evaporation because of its low temperature. The high-inventory water detritiation components can be double-contained.

Tritium is also produced in the beryllium multiplier balls at 0.3-1% of overall the tritium breeding ratio. For example, the BCSS [12] estimates tritium production as 7×10^{-6} g H^3 /g Be per MW-yr/m² for beryllium in the first 100 mm of the blanket. For TIBER, this is about 37 g H^3 /m³ Be or 220 g H^3 in the first 100 mm of beryllium balls. The total beryllium zone is 410-mm thick, so a reasonable estimate of tritium production in the beryllium is about 300 g H^3 . The inventory is limited by tritium decay and diffusion into the water shield coolant. Diffusion data is very limited; diffusion may decrease inventory if the ball diameter were decreased from the nominal 30 mm to perhaps 10 mm. Also, aqueous salt corrosion may necessitate coating the beryllium balls, which would seal in the tritium. The tritium inventory in beryllium is estimated at 80-200 g, which might be released if transient temperatures became high enough. The behavior of tritium in beryllium is not adequately known to predict any contribution to effluents or accidents.[12]

Estimating tritium in first wall materials has considerable uncertainties. Values from NET [11] are likely typical: 5-100 g in bare stainless steel (perhaps 10 g average estimate) and 1-100 g in graphite (average value of 5 g). The tritium in the divertor plate was not estimated.

The fuel processing system contains 200-300 g H^3 , distributed over a number of components around the torus. The tritium is primarily in the DT form at atmospheric pressure with temperatures ranging from 20 to 600 K. This tritium is moderately vulnerable; however, most components can be placed in glove boxes or double containment.

There will be 4-8 blanket test modules producing tritium at any one time. Assuming about 10 wppm tritium in the modules, and a comparable amount in the tritium extraction hardware, the inventory is about 100 g. The vulnerability is a function of breeder material,[12] with LiPb the most vulnerable and lithium aluminate the least.

Finally, there may be 10-200 g tritium dispersed throughout the facility in bulk or surface contamination and in waste treatment systems. The bulk tritium will be fairly invulnerable after initial out-gassing. The surface contamination will be moderately vulnerable.

Given the TIBER availability of 30%, the component inventories will vary substantially with time, although the total inventory may remain fairly

constant. During shutdown, the tritium would generally be transferred from a more dispersed and vulnerable form to less vulnerable storage at the time constant of the tritium processing system--36 hours for cryogenic distillation columns and 2 weeks for the primary coolant water.

For simple conservative accident estimates, 10% of an elemental tritium release is assumed to be converted before release to the environment. Thus, in terms of accidental offsite dose, 10 Ci of HT is assumed to have the same offsite hazard as 1 Ci of HTO. Using this approximation and reference site characteristics, one can estimate maximum offsite dose as a function of release potential, as follows:

1. High release potential - 8 mSv (0.8 rem).
2. Moderate release potential - 520 mSv (52 rem).
3. Low release potential - 120 mSv (12 rem).
4. Very low release potential - 1000 mSv (100 rem).

The reader is cautioned that these values assume 100% mobilization and release of the tritium in each release potential category, and are therefore very conservative. No credit is taken for the confinement building. The pleasing result is that 100% release of the most vulnerable tritium leads to doses less than 10 mSv (1 rem), which is the design target for very low-probability accidents. The major concern is the moderately vulnerable tritium, essentially all in the more hazardous HTO form. Except for the tritium stored as metal hydride (very low release potential if correct material chosen), the largest hazards are the 600 g H^3 in the shield coolant and 300 g H^3 in shield water cleanup system; 100% release of this 0.9 kg would cause 450 mSv (45 rem) maximum offsite dose. This must be made incredible by design. Section 5.2.3 addresses the success of the design in this regard.

5.2.2 ACTIVATION PRODUCT DOMINATED ACCIDENTS

The largest radioactivity inventory in any fusion machine is neutron activation products, rather than tritium.

5.2.2.1 Activation Product Inventory.

Major activation product inventories include the inboard shield, outboard shield, divertor plate, divertor shield, shield coolant water, and the test modules. The divertor shield and divertor plate inventories were not calculated, but are expected to fall in-between the inboard and outboard shields. Table 5.2-2 summarizes the inventory in the outboard and inboard shields. The coolant inventory is summarized in Table 5.2-3.

Resources were inadequate to provide dose calculations corresponding to these inventories. However, by scaling by Biological Hazard Potential (BHP-air) and site characteristics, one can approximate maximum offsite doses. Note, BHP is defined as the offsite Maximum Permissible Concentration (MPC) of an isotope ($\text{Ci}/\text{m}^3\text{-air}$ or $\text{Ci}/\text{m}^3\text{-water}$) divided by the inventory; BHP refers to BHP-air unless otherwise stated. For the inboard shield PCA first wall, the maximum public dose is about 25 mSv/kg-released (2.5 rem/kg). The outboard value is 50 mSv/kg-released (5 rem/kg). Per mass, the outboard first wall is twice the hazard of the inboard because the outboard average flux is 1.73 times the inboard flux ($1.52 \text{ MW}/\text{m}^2$ versus $0.88 \text{ MW}/\text{m}^2$), and the steel inboard first wall is covered by 8 mm of beryllium, which reduces steel activation. (Actually, the final design uses carbon tiles instead of beryllium, which was assumed in the activation analyses. The change from beryllium to carbon will have only a minor impact on activation levels.) Thus, release of about 80 kg (inboard) or 40 kg (outboard) could cause a 2-Sv (200-rem) offsite dose. To meet the design target of 10 mSv (1 rem) offsite dose, the released steel must be kept below about 400 g (inboard) or 200 g (outboard).

The TIBER BHP (not including test modules) is divided as follows:

Inboard first wall	$4.4 \times 10^6 \text{ km}^3\text{-air}$
Inboard total	$1.2 \times 10^8 \text{ km}^3\text{-air}$, 90% of total BHP
Outboard first wall	$3.4 \times 10^6 \text{ km}^3\text{-air}$
Outboard total	$1.3 \times 10^7 \text{ km}^3\text{-air}$, 10% of total BHP
Shield coolant (no tritium)	$4.9 \times 10^3 \text{ km}^3\text{-air}$, 0.0036% of total BHP
<u>Peak tritium in shield water</u>	<u>$3.0 \times 10^4 \text{ km}^3\text{-air}$, 0.022% of total BHP</u>
Total	$1.4 \times 10^8 \text{ km}^3\text{-air}$

Thus, the total radiological hazard is dominated by the inboard shield, which is dominated by the tungsten (W-2Ni-1Fe) blocks.

The corrosion product inventory is based on end-of-life fluence and a corrosion rate of $1 \mu\text{m}/\text{full-power-yr}$ for the $< 100^\circ\text{C}$ salt water, with no credit for removing corrosion products during operation. This equates to $0.3 \mu\text{m}/\text{calendar-year}$. A more detailed calculation for STARFIRE,[15] with a corrosion rate of $0.45 \mu\text{m}/\text{calendar-yr}$ (only 1/3 released to water) at 300°C water and on-line removal of corrosion products gave about the same total corrosion inventory. The assumed TIBER release rate into the water is thus twice the that used in STARFIRE. TIBER's higher release rate comes from the more corrosive water chemistry and the lack of steady-state water conditions (only 30% availability). However, because of the higher power level in STARFIRE, its heat transport system is 2-3 times larger in volume than TIBER. Overall, the effects cancel and the TIBER and STARFIRE corrosion product inventories are quite similar. For STARFIRE, the corrosion product inventory is only 0.004% of the structure, compared with 0.01% for TIBER.

Above Sec. 3.3.2 includes details on the corrosion product estimate (the values used here are for continuous irradiation); Sec. 3.3.2 also discusses the impact (about 25%) of step irradiation, e.g., 4 months on and 4 months off, versus the conservative assumption of continuous irradiation. The uncertainties in corrosion rates and corrosion product clean-up systems outweigh the impact of step, versus continuous, irradiation.

The air activation inventory has been estimated (Sec. 3.3.2) as under 2 mCi of C-14 and 0.1 Ci of N-16. The resulting concentration in the building is below MPC-air values for uncontrolled areas. Given further dilution in an accident, the offsite doses from air activation products would be insignificant.

5.2.2.2 Activation Product Release Potential

The possible mobilization mechanisms are (a) vaporization of material by plasma disruptions or misfiring neutral beams, (b) accumulation of erosion/sputtering products in the torus, so-called tokamak dust, (c) spill of corrosion products, and (d) volatilization of oxides produced by high temperature oxidizing conditions, e.g., decay heat or chemical fires coupled with ingress of air or water.

Table 5.2-2 Activation product inventory (Ci) in TIBER shields.^a

Isotope	First wall inboard ^b	Total inboard	First wall outboard ^b	Total outboard	Total shields	Total ^c BHP (km ³)
Sc46	5.1e3 ^d	8.6e3	4.0e3	1.6e4	2.4e4	3.5e4
V52	7.1e5	1.2e6	5.6e6	2.2e6	3.4e6	5.6e3
Cr51	2.5e6	4.1e6	2.0e6	7.6e6	1.2e7	2.0e5
Mn53	1.9e-1	3.2e-1	1.5e-1	5.0e-1	8.1e-1	4.1e-2
Mn54	1.4e6	2.3e6	1.1e6	4.1e6	6.4e6	6.4e6
Mn56	5.4e6	9.0e6	3.7e6	1.8e7	2.7e7	1.4e66
FE55	6.9e6	1.2e7	5.7e6	1.9e7	3.0e7	1.5e6
FE59	2.1e4	2.9e4	1.3e4	9.2e4	1.2e5	6.0e4
Co57	7.6e5	1.4e6	6.3e5	2.0e6	3.4e6	8.6e5
Co58	2.6e6	4.8e6	2.0e6	8.3e6	1.3e7	6.6e6
Co60	2.1e5	3.8e5	1.7e5	5.7e5	9.6e5	3.2e6
Co60m	4.4e5	8.0e5	3.8e5	1.1e6	1.9e6	3.8e2
Ni57	1.5e5	2.7e5	1.2e5	3.7e5	6.4e5	8.0e4
Ni59	6.9	9.5	3.9	3.2e1	4.1e1	6.9
Ni63	8.7e3	1.1e4	4.6e3	4.2e4	5.3e4	1.8e4
Zr89	4.0e3	6.7e3	3.3e3	1.1e4	1.8e4	1.8e3
Nb92	4.9e-4	8.3e-4	3.9e-4	1.4e-3	2.2e-3	5.5e-3
Nb93m	3.0e1	1.3e2	1.2e1	8.7e1	2.2e2	1.1e2
Nb94	4.5e-1	1.0	2.8e-1	1.6	2.6	8.6
Nb96	1.9e3	3.2e3	1.6e3	5.2e3	8.5e3	1.4e3
Mo93	2.2e3	7.8e2	1.2e2	6.2e2	1.4e3	1.6e2
Mo99	1.6e5	3.7e5	9.7e4	5.0e5	8.7e5	1.2e5
Tc99	1.3	3.0	7.9e-1	4.1	7.0	3.5
Tc99m	1.6e5	3.7e5	9.7e4	5.0e5	8.7e5	2.9e3
Hf181	6.9	5.9e4	5.5	1.8e1	5.9e4	5.9e4
Ta182	6.1e4	2.0e6	2.7e4	3.1e5	2.3e6	3.9e6
Ta183	8.3	1.6e5	7.2	2.2e1	1.6e5	2.0e4
Ta184	6.3	1.3e5	5.4	1.6e1	1.3e5	1.3e4
W181	3.9e3	9.6e7	3.0e3	1.1e4	9.6e7	4.8e6
W185	7.9e3	2.6e8	4.4e3	2.6e4	2.6e8	8.6e7
W187	7.1e4	2.1e8	3.1e4	3.6e5	2.1e8	2.1e7
TOTAL	2.2e7	6.1e8	1.7e7	2.4e8	8.5e8	1.4e8

Table 5.2-2 Activation product inventory (Ci) in TIBER shields.^a(NOTES)

- a. Inventory given in curies of each isotope. Activation calculation (Sec. 3.3.2) were per vertical height. Values here use inboard height = 5 m; outboard = 4 m. These heights should also partially account for the divertor shield region.
- b. Values shown and used in safety analyses are based on 4-mm inboard first wall (covered by 8-mm Be) and bare 1-mm outboard first wall. The final design used C rather than Be tiles, with minor activation impact. However, an erosion margin (about 5 mm) was added to the outboard first wall, which would increase the inventory beyond that given here. The structure is PCA with low Nb content, with W-2Ni-1Fe in the inboard shield.
- c. BHP-air values (inventory divided by Maximum Permissible Concentration) use MPC values calculated by Fetter, [13] because this procedure avoids over-penalizing isotopes not currently listed in 10CFR20. [14] For each isotope, the worst chemical form (water soluble or insoluble) is assumed.
- d. Powers of 10 are indicated by the letter "e": e.g., "e3" represents " $\times 10^3$."

The 10-mSv (1-rem) maximum offsite dose target leads to a target to keep the vaporization of first wall steel below 400 g (inboard) or 200 g (outboard). The same is true for mobilization of tokamak erosion dust. Release of all the corrosion products would produce a maximum offsite dose of about 50 mSv, so a 20% release of corrosion products would match the offsite dose target. In fact, for the detailed corrosion modeling in STARFIRE, the maximum possible corrosion product release was 50%; the corrosion inventory is dominated by material on the pipe walls and only the outer one-half of the oxide layer appeared releasable. Thus, one expects that less than 50% of the TIBER corrosion product inventory can be mobilized, for a maximum dose of 25 mSv. This is still quite conservative because no credit was taken for (a) on-line removal of corrosion products, (b) the fact that most of the corrosion products will not be in gaseous or aerosol form, (c) no credit is taken for plate-out in the building, or (d) successful confinement building function. Overall, the maximum possible corrosion product release should be under the 10 mSv design target.

The final mobilization pathway is the production, then volatilization, of oxide species during high temperature oxidizing conditions. This requires the presence of air or steam while the material is quite hot. No volatilization data exist for tungsten or beryllium, hence the study of these materials is a research need. However, some data exist [16,17] for PCA in air from 600-1300°C

Table 5.2-3 Summary of shield coolant inventory.^a

Isotope	Isotope concentration Ci/m ³	Inventory Ci	Inventory BHP-air (km ³ -air)	Inventory BHP-water (km ³ -water)
Corrosion products: ^b				
Cr51	152	3.0e4 ^c	7.6e1	1.5e-2
Mn54	38	7.6e3	7.6e2	7.6e-2
Mn56	126	2.5e4	8.4e2	2.5e-1
Fe55	267	5.3e4	1.8e3	6.7e-2
Fe59	2.6	5.2e2	1.1e2	8.7e-3
Co57	66	1.3e4	1.3e2	2.7e-2
Co58	106	2.1e4	7.1e2	2.1e-1
Co60	3.6	7.2e2	7.2e1	1.5e-2
Ni63	1.0	2.0e2	9.8e1	6.5e-3
Mo99	16	3.2e3	1.1e2	1.6e-2
Subtotal	779	1.6e5	4.7e3	6.9e-1
Tritium: ^d				
H3 average	1.0e4	2.0e6	1.0e4	6.7e-1
H3 peak	3.0e4	6.0e6	3.0e4	2.0
Activation products from lithium nitrate (160 g/liter-water): (e)				
Ci14	94.	1.9e4	1.9e2	2.4e-2
Total	1.1e4	2.2e6	1.5e4	1.4

a. BHP-air and BHP-water based on MPC values assuming the isotope in soluble form.

b. Corrosion products estimate (Sec. 3.3.2) normalized to 200-m³ shield water volume, at end-of-life.

c. Powers of 10 are indicated by the letter "e": e.g., "e3" represents " $\times 10^3$."

d. Tritium inventory discussed in Sec. 3.3.3.

e. Activation product inventory of lithium nitrate (Sec. 3.3.2) normalized to 200-m³ shield water, at end-of-life.

and HT-9 in air and steam from 600 to 1500°C. These data show [17] that exposure to steam increases volatility by about a factor of 2 versus air for HT-9. The PCA data do not include tungsten volatility from the steel, but the HT-9 do. Using existing data, the mobilization of PCA structure in TIBER can be estimated as a function of time and temperature, assuming ingress of air. The resulting mobilization fraction (amount-mobilized/inventory) for oxidation of the inboard steel first wall is shown in Fig. 5.2-1 for 24-hour exposure for two cases. First, only the PCA data are used. Second, the PCA data are augmented by using tungsten mobilization data from HT-9 and using HT-9 data for 1300-1500°C. Also, technetium is assumed to have the same volatility ($\text{g-mobilized/m}^2\text{-h}$) as molybdenum.

Figure 5.2-1 indicates that the 10-mSv offsite dose target could be exceeded by first wall shield steel exposure to 1100°C for 24 hours. The 2-Sv (200 rem) threshold dose that could cause acute public fatalities would be exceeded by first wall exposure to about 1500°C for 24 hours, assuming all mobilized material were released. For outboard, the corresponding limiting temperatures are somewhat lower, about 1050°C and 1400°C, respectively. To meet the 10-mSv target, the mobilization fraction must be kept below 2.0×10^{-4} (inboard) or 2.6×10^{-4} (outboard).

Of course, actual exposure of steel to these temperatures for extended times would likely lead to considerable structural damage. This would also make defendable consequences extremely difficult to perform.

Unlike the vaporization and dust mobilization pathways, the contribution of isotopes from oxidation/volatilization is not equal to their contribution to total inventory. Some elements are preferentially volatilized. Using augmented data, the major contributors to mobilized BHP from 24-hours at 1000°C (inboard) are technetium (37%), tungsten (34%), manganese (25%), iron (2%), and molybdenum (1%). At 24-hours at 1500°C, the breakdown is manganese (44%), cobalt (43%), tungsten (7%), and iron (5%). See Ref. [18] for more information on oxidation/volatilization calculations and uncertainties.

5.2.3 TRANSIENT ANALYSES

Analyses were adequate to estimate the potential offsite consequences from various transients. In all cases, no credit is taken for the confinement building; all material mobilized is very conservatively assumed released to

Figure 5.2-1. Fraction of radioactivity (BHP) mobilized from inboard first wall by 24-hour exposure to air.

the environment. In most cases, more detailed analyses are required to resolve uncertainties.

5.2.3.1 Plasma Disruption Effects

The energy available to the first wall in the reference plasma disruption (Sec. 2.10) is 109 MJ from the thermal quench and 75 MJ from the current quench. The thermal quench is assumed to take about 0.1 ms and to deposit energy 30% uniformly on all walls and 70% uniformly on the divertor plates. The current quench is assumed to take 10 ms (1 ms/MA) and to deposit energy 33% uniformly and 67% on the inboard side. The calculation of how much material is vaporized depends on assumptions on how vaporized material diffuses back into the plasma.

Detailed calculations (Sec. 2.10) predict 250 g of divertor tile material (tungsten) could be vaporized in a disruption. Activation calculations of the divertor were not performed. However, tungsten BHP is not too dissimilar to PCA. If released, 250 g of PCA could lead to maximum offsite doses of 6 mSv (inboard side activation levels) to 13 mSv (outboard side); the divertor region has in-between activation levels. Thus, disruptions could only mobilize enough material to cause a maximum offsite dose of about 10 mSv, even if all the vaporized material were somehow released to the environment.

However, disruptions are anticipated events, about 102/yr. Thus, careful attention is needed to insure that the design survives disruptions and that mobilized material is confined to the vacuum vessel.

5.2.3.2 Off-Normal Plasma Effects

The preceding plasma disruption effects are based on "typical" severe plasma disruptions in tokamaks, i.e., disruptions caused by known phenomena. Other off-normal plasma states may be possible, e.g., disruptions caused by coil failures or movement, in-leakage of gases. Thus, it is interesting to examine the maximum theoretical vaporization of first wall material from perfectly focused plasma energy. Taking all the energy in the plasma into account, about 23 kg of steel could be vaporized. The resulting maximum offsite dose, 1.1 Sv (outboard) or 0.6 Sv (inboard), is below the 2-Sv acute fatality threshold, even without taking credit for all the mechanisms that will lower the doses (plate out, non-perfect focusing of plasma energy, etc.).

One way the plasma energy could be focused is if some component accidentally fell into the plasma. Something similar has happened at JET where a Langmuir probe was vaporized by runaway electrons, depositing on the vessel walls.[19] Probes, tiles, and perhaps structural pieces (in accidents) could fall into the TIBER plasma and be vaporized. Provision should be provided to clean or replace the first wall if needed.

The other unresolved issue is plasma stability. If plasma control systems fail, the plasma temperature might increase, which might lead to a plasma disruption. Fortunately, the current driven mode probably has inherent stability in that overpower excursion may be impossible, even if plasma control systems fail. Additional analysis of plasma stability, especially any potential for overpower transients, is needed and is on-going.

5.2.3.3 Plasma Erosion/Sputtering Dust

Routine operation and anticipated plasma disruptions will produce so-called tokamak dust. Analyses for NET [20] predict that 3 kg of dust could be mobilized by accidental ingress of air into the torus. This amount is less than 1% of the amount eroded or sputtered over the NET lifetime.[20] The amount eroded in TIBER is probably less because of lower surface area. Conservatively assuming TIBER might have the same amount of mobile dust as NET, half originating from the inboard and half from the outboard, the maximum offsite dose would be 75 mSv (7.5 rem). Note, under routine conditions no activated steel would be mobilized from the inboard because it is covered with carbon. Thus, only the 1.5 kg dust from the outboard side would contribute, the carbon hazard being small compared with steel.

This high a dose would be quite undesirable; routine cleanup and control of this tokamak dust would seem quite important. Other sources of dust, and their radiological hazards, should be examined, e.g., tungsten divertor plates. Overall, the generation and behavior of activation erosion dust is unknown and must be studied.

5.2.3.4 LOCA/LOFA Analyses with Plasma On

In an accident, the plasma will eventually terminate from one of several causes:

1. Operator terminates the burn in a controlled manner.

2. Cause of LOCA/LOFA also causes plasma disruption, e.g. seismic event moving coils.
3. Heatup of wall material increases impurity flux to plasma, e.g. STARFIRE [15].
4. Heatup of walls leads to loss of vacuum boundary; in-leaking fluids terminate plasma.
5. Heatup of walls initiates some type of passive injection of impurities to plasma.

One desires that the plasma be terminated in a benign manner (no worse than "typical" plasma disruption) by passive, versus active, methods, quick enough to avoid damaging the machine. In fact, licensing of TIBER would likely require assurance that such a shutoff method existed.

It is instructive to examine the possible time scale required to shutoff the plasma. For outboard LOFA, the surface heat load during the first 10 seconds will be dissipated by coolant boiling, and the first wall temperature will reach 120°C. For outboard LOCA, 10 seconds of plasma burn raises the maximum first wall temperature to 450°C. Lower temperature rises are expected for the inboard first wall, because the wall is thicker and the heat flux lower. From the shield perspective, the plasma must terminate within about 10 seconds.

However, the high surface heat flux on the divertor makes the divertor the limiting component. The peak surface flux is 6.6 MW/m^2 . One key temperature to avoid is the liquidus temperature of the braze between the tungsten tile and the copper substrate, about 880°C. The initial tile temperature ranges from 198°C directly over the copper tubing to 241°C at the tile corner. Two simple calculations were performed to estimate how long the divertor could tolerate a complete loss of coolant without braze failure.

Treating the braze as an adiabatic boundary, only 0.5 s of plasma operation during full LOCA could be tolerated. Including the copper raises the allowable time to 3.5 s. The worse divertor for LOCA purposes is the lower one because a LOCA could completely and rapidly drain water from the divertor tubing. Realistically, the plasma must be terminated within 1-5 s of a LOCA to avoid divertor damage. A complex two-dimensional analysis should be performed to narrow this range. Even after plasma termination, the heatup of

the divertor would be significant because both tungsten and copper are high decay heat materials.

If significant divertor damage occurred, it could probably be repaired since the divertor is replaceable. However, broken tiles or melted braze or copper could lodge somewhere in the machine, making repair/recovery difficult. Furthermore, defensible analyses of transient consequences above 880°C would be extremely difficult because of changing geometry. Furthermore, the complex chemical behavior of molten materials, especially with air intrusion, would be most difficult to analyze.

Available analyses are insufficient to determine if TIBER has a rapid (1-5 s), passive, benign plasma shutoff method. The 1/e energy confinement time is about 1 s. With pellet refueling occurring at 0.9 Hz, stopping fueling (active but benign) may not be quick enough if the required time scale is 1 s, rather than 5 s. A rapid, active, benign shutoff method should be possible to design. However, finding a passive mechanism requires more thought and analysis.

5.2.3.5 LOCA/LOFA Analyses with Plasma Off

Assuming the plasma is terminated following loss-of-coolant accident (LOCA) or loss-of-coolant flow accident (LOFA), the question is whether removal of decay heat is adequate. No LOCA/LOFA analyses were conducted for the divertor plate and divertor shield, but their response should fall in-between that of the inboard and outboard shields.

Including the inboard and outboard shields (and first walls), the decay heat at shutdown is only 1.3% of the operating power. This falls to 0.35% at 1 minute, 0.31% at 1 hour, and 0.16% at 1 day. Thus, less than 1% of the operating coolant flow is required to keep shield temperatures at or below design temperatures. Thus, natural circulation of the coolant, as might occur in a LOFA, should be adequate to prevent shield heatup. Placing a passive air heat exchanger in series with the main heat exchanger (Sec. 3.3.2) would be beneficial. Sizing the air exchanger to reject about 1% of the normal heat load would provide for decay heat rejection during LOFA or loss of the secondary system. However, a detailed study of whether natural circulation would occur in all parts of the shield has not been performed. Some areas of the complex shield probably do not have flow paths sufficiently conducive to cause natural circulation; more analyses of the final design are required.

In any case, the more severe condition is LOCA, where natural circulation of the water is not possible. Simple calculations were performed to examine the severity of LOCA. These adiabatic calculations assumed no heat removal from the shield, but perfect conduction within the shield. The average outboard temperature rises only 30°C over 1 day. Thus, even though the first wall temperature rise will be higher than this average, LOCA's do not appear to be a significant concern for the outboard shield. This was expected since the decay heat density for the outboard shield is slightly lower than that of a helium-cooled lithium oxide HT-9 design [18] for a commercial fusion power plant that was found to be passively safe, with minor temperature increases under any LOCA/LOFA condition.

However, the inboard side decay heat density is an order of magnitude higher than the outboard side (or the passively safe power plant design) for the first 24 hours after shutdown. The tungsten blocks contribute about 90% of the inboard decay heat over the first day. That is, without the tungsten, the inboard side would be similar to the outboard side and would not be a major concern. Thus, with tungsten, TIBER has a worse decay heat problem than most past commercial conceptual design studies or NET.

An adiabatic calculation shows that the inboard shield average temperature rises 1200°C in 24 hours, from 80 GJ of decay heat. The inboard first wall would be expected to heat up somewhat faster, and would melt before 24 hours (PCA melting point 1430°C). If representative, the adiabatic calculation indicates that large-scale melting of the inboard shield and significant mobilization of radioactivity (doses above 2 Sv) would result from loss of coolant to the inboard shield, followed by release to the environment.

More realistic calculations should show that the inboard response is more benign because (a) some decay heat will be "absorbed" by the inboard magnets, (b) some heat will escape vertically past the divertor shield regions, and (c) some heat will escape via thermal radiation to the outboard shield. Some possible solutions to the decay heat were considered:

1. Reduce the amount of tungsten. More detailed LOCA calculations are needed to show how much tungsten decay heat could be tolerated.
2. Design for natural circulation of building cover gas (nitrogen) through the inboard coolant manifold. If this overheats the building, passive building coolers (Sec. 5.6.5.2) would be considered.

3. Design for natural circulation of air through the inboard coolant manifold, bring air into the building and allowing it to exit directly up a 30-40 m stack. Calculations (Sec. 3.3.2) do show that this solution would work. However, it is very doubtful that direct venting of gas to the environment, without filtration, would be permitted during an accident. Furthermore, deliberate air ingress to the building would exacerbate the chemical reaction concerns with lithium, lithium-lead, and graphite (Sec. 5.2.4.2). Passive initiation of such a system for all relevant transients could be difficult.
4. Design for gravity drain of water from an atmospheric-pressure tank positioned above the tokamak. Boiling about 50-m³-water would remove the inboard decay heat for the first day. Perhaps even an external large water tank could be considered. However, such a system would be a temporary solution, because the water supply would be limited, and might have to be actively initiated and controlled. Furthermore, adding either air or water to the system increases compatibility problems with liquid metal test modules and the graphite tiles (Sec. 5.2.4.2).
5. Add an active emergency coolant system. A simple sump pump to the top of the inboard shield may be all that is required, rather than a high pressure injection system. However, if the coolant passages in the inboard side were damaged, adequate cooling could be difficult. Furthermore, this might only add more water to the torus, possibly increasing chemical reaction concerns.

Detailed examination of these potential solutions is needed. The available simple calculations do show that a problem exists, but TIBER II resources were insufficient to resolve the concern. The viability of the high-tungsten-content inboard shield requires finding a satisfactory solution to removing decay heat during a LOCA.

Even if decay heat does not raise the structural material hot enough to mobilize radioactivity, a coolant spill mobilizes the tritium and activation products in the shield water coolant. Most of the corrosion products would not be expected to vaporize with the water so that the corrosion product contribution would be expected to be far below the theoretical value of 25 mSv

indicated in Sec. 5.2.2.2 for 50% release of all corrosion products. However, the tritiated water would boil. For illustrative purposes, if 25% of the decay heat released over the first day (33 GJ) boils water, then 6.8% of the water would boil and the offsite dose from tritium at peak concentration (30 Ci/liter) would be 20 mSv (2 rem), above the design target.

In summary, it appears that loss-of-shield coolant accidents have the potential to cause offsite doses above the 10 mSv (1 rem) design target, either via volatilization of inboard shield structure or boiling the tritiated water. Volatilization of the shield may even exceed the 2-Sv acute fatality threshold. In future studies, more analyses and/or reduction of decay heat problems are required.

5.2.3.6 Graphite Tile Chemical Reactions

Reactions of hydrogen (H, D, or T) with carbon can form both methane and acetylene. Which forms, and how much, is sensitive to the temperature of the carbon and the form and temperature of the hydrogen (monatomic or diatomic). Monatomic hydrogen is the more reactive for low temperature graphite. Plasma particles intercepting graphite limiters are monatomic as are energetic charge-exchange neutrals.

Most investigations of chemical sputtering of graphite by light-ion bombardment has focused on the formation of methane, which is the greatest problem because it increases the graphite sputtering yield well above that observed by physical sputtering only. Methane is formed at temperatures of 200-800°C, and formation rate peaks at 550-600°C with about 1 keV incident hydrogen energy. In the 200-700°C range, sputtering coefficients fall within 10⁻² to 10⁻¹ for H⁺ and D⁺. [21]

Fear of chemical sputtering of graphite led experimenters to coat limiter blades with TiC. This, however, was not very successful because of the tendency of the coating to flake off with high heat fluxes. Graphite limiters are now being used without coatings, and chemical sputtering does not appear to be a problem in a tokamak plasma. [22]

Under routine operation, neither the chemical sputtering erosion of the carbon first wall tiles nor the small amount of methane it produces are expected to be a concern. Water coolant spraying from a broken pipe onto the hot carbon tiles (at 1400°C), however, could create hydrogen and carbon monoxide by the water gas reaction:

This reaction only takes place above about 1000°C. The maximum amount of carbon reacted before the 10-mm thick tile would cool below 1000°C was estimated to be 0.3 g/cm², or 15% of the total thickness. If the entire tile-covered inboard first wall (50 m²) were to be sprayed, up to 150 kg of carbon could react, producing 25 kg of H₂ and 350 kg of carbon monoxide. At standard temperature and pressure, this is two times the amount of hydrogen necessary to fill the vacuum chamber; accidental air ingress could lead to a hydrogen explosion. See Sec. 5.2.4.2 for more details on preventing air ingress and hydrogen combustion. This concern must be further investigated, including a transient calculation to determine how much graphite would react before it cooled below 1000°C. If necessary, water flow limiting devices must be used to prevent significant sprays of water on the carbon tiles, or either the carbon tiles or water coolant could be eliminated.

Carbon could burn should air penetrate the plasma chamber. If sufficient air is present, the carbon reacts to form dioxide, releasing 33 MJ/kg carbon. If there is excess carbon, the monoxide reaction prevails and the energy release would be 9.4 MJ/kg. Adiabatic flame temperatures are 1200°C for the monoxide reaction and 2180°C for dioxide. Some local melting of metal alloys could occur if in direct contact with the dioxide flame but not with the monoxide flame. Unless the carbon is not only pulverized but entrained in the air, burning rates should be low enough to preclude any pressure buildup. It is unlikely, therefore, that carbon used for first wall armor, and divertor plates will pose any additional hazard for damaging the reactor in the case of air in-leakage accident. Tritium in the graphite, however, might be released.

5.2.3.7 Beam and Pellet Misfire Effects

During operation, neutral beams and DT fuel pellets will be injected into the plasma. However, misfires of these systems could damage the machine. The kinetic energy of fuel pellets in existing test stands are capable of penetrating 3 mm (1/8 in.) steel. TIBER pellet injectors could have even more kinetic energy. Thus, operation of the pellet injector without the plasma present, e.g., during shutdown or after a plasma disruption, is expected to damage the target wall area. A cold, rather than hot, plasma might also allow the pellet to strike the other side. Although interlock systems might be

expected to reduce the frequency of this event, say from "anticipated" to "low probability," it still appears sufficiently likely that a pellet-dump should be provided opposite pellet injectors, similar to neutral beam dumps. This pellet-dump should be easily replaceable.

The current TIBER pellet system calls for 5-mg pellets injected at 1-2 km/s, for 20 J/pellet. If, however, pellets were replaced with compact toroid (CT) fueling,[23] then the presence of a hot plasma would not be required to prevent striking the other side of the torus. Only the continued presence of magnetic field would be needed to dissipate the CT. CT fueling would certainly be stopped well before the magnetic fields were lost in some accident.

Two actual neutral-beam events highlight the need for beam-dumps and interlock systems. In JET, malfunction of cryogenic systems in a neutral beam caused the beam to move off-target, burning a 10-cm² hole in a water-cooled Inconel duct wall. This flooded the torus with over a cubic meter of water, which escaped from the torus by breaking a diagnostic window not built to withstand the water pressure head.[24] In TFTR, despite interlock systems, a neutral beam was fired when a valve was incorrectly closed, destroying the valve.[25]

Clearly, off-normal events of neutral beams and fuel pellet injectors have the potential to damage the machine. Careful attention to possible failure states is required to adequately limit damage and provide for repair.

5.2.3.8 Summary of Transient Analyses

Table 5.2-4 summarizes the preceding transient analyses, together with a conservative estimate of the likelihood of the event. The major concerns are:

1. Decay heat removal from the inboard side during a loss-of-coolant.
2. Vaporization of tritiated shield water coolant.
3. Provision for quick (about 1 second), benign, and passive plasma termination in the event of LOCA or LOFA in the divertor, shield, or test modules.
4. Control of tokamak erosion dust.
5. Design tolerance of plasma disruptions.

Table 5.2-4 Summary of conservative estimates of hypothetical accident consequences in current TIBER design.

Hypothetical transient	Estimated consequences		Worst case frequency regime ^b
	Maximum off-site dose (mSv) ^a	Onsite	
Plasma disruption, with vacuum boundary failure	10	Small	Anticipated
Tokamak dust mobilized from vacuum boundary failure	75	Hall contaminated	Low probability
Large structural piece falls into plasma, is vaporized (if outboard steel)	1100	First wall coated	Very low probability
All highly vulnerable tritium released	8		Low probability
Decay heat boils 6% of tritiated water during LOCA	20		Low probability
1000 g H ³ in U-bed released if major air ingress to bed	500	U-bed destroyed	Very low probability
Inboard LOCA heats up structure, with no corrective action	2000	Considerable inboard damage	Very low probability
Divertor LOCA, plasma burn Unknown lasts more than 1 s	Unknown		Damage likely
Water sprays on carbon tiles, makes H ₂ ; air ingress causes explosion Very low	Unknown		Damage likely probability
LOCA spills 20% of corrosion products, some volatilize	<< 10		Low probability
Pellet injector fires when plasma not present	Low	Target area damaged (c)	Anticipated
Neutral beam fires when plasma not present	Low	Target area damaged (c)	Anticipated

Table 5.2-4 Summary of conservative estimates of hypothetical accident consequences in current TIBER design. (NOTES)

- a. Values shown are generally quite conservative. All radioactivity mobilized/volatilized is assumed released to the environment under worst-case weather conditions. 10 mSv (1 rem) is the design target for maximum offsite dose; 2000 mSv (200 rem) is the threshold dose beyond which acute fatalities are possible.
- b. Conservative estimates from engineering judgement. See Table 5.1-1 for definition of frequency regimes and examples.
- c. Unless replaceable beam/pellet dump provided.

Without further analyses, one cannot determine whether the consequences from the associated transients will be acceptable from either the public safety or machine damage standpoints.

5.2.4 TEST MODULE ACCIDENTS

The preceding section ignores the potentially dominant hazards associated with the blanket test modules. The spectrum of materials in these modules will likely include several coolants (pressurized water, pressurized helium, flibe, lithium, and LiPb), several breeding materials (lithium, LiPb, flibe, and several ceramics), and multiple structural materials (V-alloys, low activation ferritic steels). Each has favorable and unfavorable safety and environmental characteristics.[12] It is beyond the scope of the TIBER II effort to examine these characteristics and possible interactions with the basic TIBER machine. Rather, a preliminary examination was made to find any hazards or concerns that could cast doubt on the fundamental TIBER design, i.e., low pressure/low temperature water-cooled shield surrounding several different test modules. Analyses focused on two physical mechanisms, thermal and chemical interactions.

5.2.4.1 Thermal Interactions

One class of potential hazards involves the thermal mismatch between the test modules and basic machine. Test modules will operate at commercially relevant temperatures and pressures, with first wall temperatures as high as 700°C (flibe module) and pressures as high as 15 MPa (water module). The modules will be surrounded by the shield, which is cooled by water at 40-70°C and 0.2-0.3 MPa pressure. The issues are very similar to possible interactions

between water-cooled limiters ($< 100^{\circ}\text{C}$ water) and various types of blankets in the BCSS (see Sec. 5.4.3.5 of Ref. [12]). Several interactions may be possible: (a) cold shield water could thermally shock test modules, (b) hot fluid from test module, dropping into a cold water pool, could initiate a steam explosion, and (c) release of pressurized module coolant into the vacuum chamber could overpressurize it.

Thermal Shock Potential. With a temperature difference of over 600°C between shield water and the first wall of some test modules, the potential for cold shield water thermally shocking the test modules is at least as high as limiter/blanket thermal shock in the BCSS. In most designs, such thermal shock poses minimal public safety concerns because of compatibility between fluids in various plasma-facing components. In TIBER, however, the incompatibility between liquid-metal test modules and the water coolant highlights any mechanism, like thermal shock, whereby the leakage of one fluid could propagate to the leakage of the other. Spill of shield water over the face of a test module could thermally shock (and fail) the test module wall. A helium-cooled or liquid metal-cooled shield would not pose this concern.

Steam Explosions. Accidents in existing tokamaks, all water cooled, have lead to water in the torus. The same should be expected of TIBER. If a simultaneous failure of a test module were to spill/spray a hot fluid (lithium, LiPb, flibe) into the water, there is the potential for a steam explosion. Two possible pathways for this simultaneous failure are (a) cold water thermally shocks test module first wall and (b) earthquake. The possible consequences from resulting steam explosions were not studied in TIBER II, being beyond the scope of available resources. A helium-cooled shield would not pose this concern.

Overpressure of Torus. The low pressure single-phase coolants (lithium, LiPb, and flibe) do not pose hazards of overpressurizing the torus simply by leaking into it. All three do pose steam explosion hazards; the liquid metals pose chemical interaction hazards. However, pressurized water and helium test modules may pose an overpressure hazard. Test module designs have coolant volumes in the range $1\text{--}2\text{ m}^3$. TIBER and the BCSS are compared as follows:

TIBER - 1000-kg water at 300°C, 15 MPa into 150-m³ vacuum chamber with most steel wall temperature near 100°C (surface of carbon tiles at 1400°C)
BCSS - 2000-kg water at 100°C, 4.2 MPa into 950-m³ vacuum chamber with steel wall temperatures of 550°C

In the BCSS, the leak of one limiter was thought capable of raising the torus pressure to 600-800 kPa. Using the same methodology, release of 1000-kg water from a test module would raise the torus pressure to 500-1300 kPa if the resulting steam/water mixture attains temperatures in the range 150-200°C. Pressure relief of the vacuum chamber appears warranted.

5.2.4.2 Chemical Interactions

Three types of test modules may pose significant chemical reaction hazards, (a) lithium-cooled, (b) LiPb-cooled, and (c) water-cooled LiPb-breeder. Preliminary criteria to judge the acceptability of using these modules in TIBER include:

1. Keep the maximum possible production of hydrogen gas below the explosive limit (18%) in the vacuum volume, preventing a hydrogen explosion in the torus in the event of hydrogen filling the torus volume coupled with contact with air and an ignition source. A worst-case mixture of hydrogen and air in a 150-m³ vacuum vessel would release 0.5 GJ of energy (100 kg-TNT).
2. Keep the maximum possible production of hydrogen gas a factor of 10 below the flammability limit (4%) in the reactor hall. The factor-of-10 margin is designed to prevent locally higher concentrations from exceeding this limit, preventing hydrogen combustion or explosions in the reactor hall.
3. Keep the maximum possible mobilization of radioactivity below that which could cause an offsite prompt dose of 10 mSv (1 rem). This is a TIBER design target.
4. TIBER design should be tolerant of test module failure.

Early in the study, some parameters were assumed to estimate the acceptable quantity of liquid metals in a test module. The data, modeling, and assumptions given in Ref. [26] were used. No credit was taken for active

systems, consistent with TIBER design targets. A test module is assumed to fail and spill the total volume of its liquid metal. Given the numerous sources of water in the TIBER design, water is assumed available for reaction in a worst-case accident.

The resulting initial guidance to test-module designers is summarized in Table 5.2-5. The parameters initially assumed were too optimistic (too favorable) in two areas--the vacuum vessel and reactor hall volumes are too high. The correct vacuum chamber volume is about 150 m^3 ; the reactor hall is $48,000 \text{ m}^3$. However, the fluence to any given test module would be less. Most importantly, nitrogen was adopted as the cover gas within the reactor hall, reducing lithium-air and hydrogen combustion concerns. One can assess how successful the final design was in allowing liquid metal testing in a water-cooled machine.

Lithium-Cooled Test Module. The module holds about 0.8 m^3 of lithium; piping to the module contains another 0.13 m^3 . Processing subsystems associated with the module, located below the reactor hall, contain another 1 m^3 . About $1 \text{ m}^3 \text{ Li}$ (500 kg) is at risk.

Complete reaction of lithium with water is expected to produce $3.6 \times 10^4 \text{ mole H}_2/\text{m}^3 \text{ Li}$. At standard pressure and temperature, the potential hydrogen gas would be $860 \text{ m}^3 \text{ H}_2$. Thus, if air could enter the vacuum chamber concurrent with a lithium-water reaction, the air-hydrogen mixture could explode, given an ignition source. Fortunately, no air is in the building. Possible pathways to allow air into the building include failure of the building and failure of any ducts/ports, e.g., vacuum ducts, that might penetrate the reactor hall.

Hydrogen produced from $1 \text{ m}^3\text{-Li}$ could raise the hydrogen concentration in the reactor building to only 1.8%, a safety factor of 2.2 below the flammability limit of 4% hydrogen in air. Thus, one would not expect major hydrogen combustion (or explosion) hazards in the building itself.

Lithium-air reactions can produce temperatures of 1300°C at the localized combustion zone.[26] Scaling up from lithium-fire tests at 45 and 100 kg-Li, the available product of surface area and time is under $30 \text{ m}^2\text{-h}$. Using the available data base [16,17] for volatilization of PCA in air suggests that about 1.8×10^{-4} of the outboard first wall BHP could be mobilized. If

Table 5.2-5 Initial guidance to test module design to limit lithium and lithium-lead volumes in test modules.^a

Criterion	Limit on Lithium (m ³)	Limit on LiPb (m ³)	Comments
Hydrogen concentration in vacuum chamber below explosion limit	0.06	0.5	Perhaps can discard if nitrogen in building and no pathways into torus
Hydrogen concentration in building below 1/10 of flammability limit	0.9	7	Perhaps could relax if nitrogen in building
Mobilization of radioactivity < 10 mSv			LiPb limits are equivalent activated material
lithium-air/structure	>100	n/a	
lithium-water/structure	10	n/a	
chemical reactions mobilize tritium	150	1200	
chemical reactions mobilize water			
radioactivity	?	?	
LiPb-water mobilize			
LiPb activity	n/a	4	

a. Early in the study, this guidance was given to test module designers, based on assumptions for first wall area (200 m²), vacuum vessel volume (300 m³), reactor hall volume (2 x 10⁵ m³), air as building atmosphere, and neutron flux (1.5 MW/m² for 2 full years).

released to the environment, this produces a maximum offsite dose of only 7 mSv, below the 10-mSv design target.

In fact, complete reaction of 1 m³-Li with possible reactants will release only about 1/5 of the decay heat released over the first day after shutdown: lithium-nitrogen (5 GJ), lithium-oxygen (22 GJ), lithium-water (no hydrogen combustion--16 GJ; with subsequent hydrogen combustion--25 GJ). Both decay heat and lithium-air fires release energy somewhat slowly, compared with hydrogen combustion.

Overall, the dominant concern from a lithium test module is preventing hydrogen combustion. Design features, which should reduce the frequency of the lithium-water-air reaction required to produce hydrogen combustion, include

(a) nitrogen as building atmosphere, (b) minimizing lithium volume, (c) double-containing the test module at all points except the first wall. Analysis is insufficient to know if the resulting risk is acceptable from the public or machine damage standpoints. However, it does appear that testing of $1\text{-m}^3\text{-Li}$ modules in TIBER might be possible, with proper design.

If cost effective, an inert cover gas like helium would make a better cover gas than nitrogen, since lithium does react with nitrogen. Use of nitrogen rather than air does help prevent lithium-air and hydrogen-air reactions. Note, carbon dioxide is not an inert gas with respect to lithium; they react violently.[26]

Lithium-Lead Cooled Test Module. The complete reaction of $1\text{ m}^3\text{-LiPb}$ leads to a factor of 8 less hydrogen than $1\text{-m}^3\text{-Li}$, because of the lower lithium atom density. Still, intrusion of air into the vacuum chamber concurrent with a water-LiPb reaction could lead to a hydrogen explosion. However, the hydrogen concentration in the building overall is limited to about 0.2%, 20 times below the 4% flammability limit. In both cases (torus, building), the hydrogen concerns are qualitatively the same as for a lithium-cooled module.

Unlike lithium, the dominant radioactivity mobilization concern for LiPb is mobilization of LiPb activity itself. Thus, the level of activation of the LiPb is crucial. If the LiPb test module is exposed to neutrons for only a short time, the radioactivity mobilization concern is decreased. Similarly, if the lead-related activation products (mercury, thallium, bismuth, and polonium isotopes) are removed from the LiPb, the concern is decreased, but at increased expense.

The BHP of LiPb exposed at the first wall at 5 MW/m^2 for 2 years is $2.4 \times 10^6\text{ km}^3\text{-air/m}^3\text{-LiPb}$. [12] For an assumed factor-of-10 flux reduction for LiPb in the test zone versus the first wall, and a maximum first wall fluence of 3 MW-yr/m^2 , one $\text{m}^3\text{-LiPb}$ might have a BHP of 8×10^4 . The release fraction for polonium, which is the dominant LiPb activation product, from LiPb has not been measured. However, the measured release fraction for thallium and mercury, with similar vapor pressures, are in the range of 0.1 to 1%.[26] If 1% of the LiPb radioactivity were conservatively assumed to be mobilized during an accident, the resulting maximum offsite dose would be about 9 mSv, essentially the same as the maximum offsite dose from a $1\text{-m}^3\text{-Li}$ fire. In this estimate, no

credit is taken for on-line removal of LiPb activation products or reducing the fluence below 3 MW-yr/m^2 .

As with a lithium-cooled module, further analyses are required to determine if the risk of a LiPb-cooled module in a water-cooled ETR is acceptable. In both cases, the dominant concern is preventing hydrogen combustion within the torus.

Water-Cooled LiPb-Breeder Test Module. The consequences to the machine or public are similar to the case of a LiPb-cooled module with an important difference--here a high pressure water source exists within the test module. Therefore, the likelihood of a water-LiPb reaction is higher than in a LiPb-cooled module.

Summary of Liquid-Metal Test Modules. The three types of liquid metal-test modules all pose significant chemical reaction concerns, especially production of hydrogen gas from liquid metal-water reactions. If hydrogen combustion/explosions can be prevented, the offsite radiological consequences appear below 10 mSv (1 rem) --if the volume of liquid metal is kept to about 1 m^3 . However, if hydrogen combustion/explosions occur (inside the torus), the machine would likely be significantly damaged, which might also lead to unacceptable offsite consequences.

Further analysis would be required to know if the combination of limiting the liquid-metal volume and module design adequately reduces the risk of liquid-metal testing in a water-cooled machine. Three critical design details are (a) using nitrogen as the reactor hall cover gas, (b) eliminating penetrations through the reactor hall wall (e.g., vacuum ducts) whose failure might allow air into the hall, and (c) limiting the volume of liquid metal to about 1 m^3 . Eliminating these features could eliminate liquid-metal testing in TIBER.

At present, the most worrisome penetration through the building wall is the neutral beam lines. Detailed design should adequately limit the frequency of air intrusion into the vacuum chamber via the neutral beam lines.

The highest risk (consequence times frequency) of the test modules may be the water-cooled LiPb-breeder module. In severe accidents, all three liquid-metal modules could produce enough hydrogen gas to cause an explosive mixture

(given air intrusion) in the vacuum chamber, but the water/LiPb module appears to have the highest chance of producing the hydrogen.

From the standpoint of module-machine interactions, a helium-cooled basic machine would be significantly more attractive. Substitution of helium for water would reduce risks from both thermal and chemical interactions.

5.3 PERSONNEL SAFETY AND MAINTENANCE EXPOSURE

This section provides some design guidance with respect to the radiation environment associated with the TIBER torus and associated systems and support facilities. Available TIBER analyses are insufficient to predict the actual level of radiation hazards that will arise during operation of the TIBER facility. These will be determined by the reality of the final TIBER mission (and design), and the cost/benefit of ongoing modifications to the torus and associated systems.

5.3.1 DESIGN APPROACH FOR MAINTENANCE ACCESS

As presently envisioned, in any fusion power facility, the bulk of maintenance work in the reactor hall will require remote handling systems, because of the high activation levels of first wall and blanket/shield components. The biological shielding around the torus necessary for routine shutdown access is generally viewed as prohibitively expensive. However, some shielding is necessary to limit neutron damage and thermal load on magnet materials/systems, and more is needed to avoid unacceptable activation of structural material in the reactor hall.

Accordingly, some degree of personnel access may be possible, and is likely desirable to:

1. Optimize performing small, relatively intricate tasks that would not be cost effective to perform remotely, e.g., connecting coolant lines, and
2. Ensure a large degree of operational flexibility in response to unanticipated tasks for which no remote handling capability exists.

The issue of hands-on versus all-remote maintenance has been addressed and investigated in numerous other studies such as INTOR, STARFIRE, MINIMARS, and initial TIBER studies.[27,15,5,28] Findings from these studies, as well as discussions with TIBER design team, indicate that the facility layout should embody in its overall design approach, some degree of operational/maintenance flexibility. Major maintenance tasks associated with the torus that can be foreseen as likely, or even remotely necessary, will be designed for remote handling. (However, minor maintenance tasks outside of the torus hall and the

hot cells should be performed hands-on. Design of balance-of-plant and support systems should be such that fields from transported activation products, or direct activation will not preclude access.) However, design should be such that short-term contingency access will be possible in response to maintenance problems that were not foreseen, and for which remote handling systems were not provided. Dose rates in these locations should be 0.1-1 mSv/h (10-100 mrem/h), depending on location and occupancy factors, perhaps 4 weeks after shutdown. Because the reference scenario is fully remote maintenance, resort to this contingency manned access would be expected very infrequently. In areas such as the torus hall, contingency access may require the use of local or portable shielding, even accounting for the presence of magnet shielding. The current TIBER design does not include provision for contingency manned access but future studies should. Contingency access would require portable air supplies since the building cover gas is nitrogen.

The greater proportion of radiation dose accumulated by fusion facility staff is unlikely to result from exposures in high field areas such as the reactor hall. Experience indicates that most person-Sv (man-rems) are accumulated through integrated exposure to lower radiation field areas such as heat exchanger and circulating equipment rooms, auxiliary equipment rooms and waste handling/cleanup system rooms. In these locations, activated corrosion products that are transported outside of the primary reactor shield enclosures result in radiation fields which, while not high enough to justify the cost of remote systems for maintenance, are nonetheless significant enough to require radiological work planning and dose control.

For areas that may have some radiation sources, but where routine access is envisioned, dose rates should be in the range 5-100 μ Sv/h (0.5 to 10 mrem/h), again depending on projected occupancy. At this preliminary stage, the working areas at TIBER are grouped into three major types as indicated in Table 5.3-1.

5.3.2 OCCUPATIONAL DOSE CRITERIA

Due to the preliminary nature of facility design and layout, it is very difficult to estimate the accumulation of person-Sv by facility staff in a

Table 5.3-1. Classification of TIBER radiation work areas.

-
- Zone I Non Radiation Area--An area that does not contain systems and equipment that are potential sources of radiation, and for which controlled access is unnecessary. This includes areas accessible to visitors without accompaniment, and work locations of personnel not trained and classified as radiation workers, administrative areas, cafeterias, workers' rest rooms, and change rooms.
- Zone II Normally Accessible Radiation Area--An area that contains a minimum number of systems/equipment that are potential sources of radiation. Equipment in this area requires frequent access for inspection, operation or maintenance. Radiation workers will have unlimited access, in spite of administrative controls to limit non-radiation workers. This would include the inactive workshops, stores, control room, chemical and health physics labs, and all general access corridors in the facility.
- Zone III Limited Access Radiation Area--This would include areas wherever equipment will have radiation fields, or where the potential for radioactive contamination is persistent. Normally, most Zone III areas would only be accessible during TIBER shutdown. This would include all coolant equipment rooms, active workshops, waste management systems areas, and tritium systems rooms.^a
- Zone IV No Access Radiation Area--An area with dose rates too high to allow manned access, even during shutdown. This would include the plasma chamber.
-

- a. The torus hall, normally inaccessible even during shutdown, could be classified as Zone III during major maintenance outages or under contingency conditions.

typical year of operation. However, some preliminary dose rate targets assigned to zones or areas, along with estimates of overall facility person-Sv

can be advanced as part of the iterative approach to establishing appropriate occupational dose performance requirements.

The International Commission on Radiological Protection (ICRP) judges the acceptability of the level of worker risk in a nuclear facility by comparing that risk to that for other occupations recognized as having high standards of safety. Target values of personnel exposure are presented to assist future refinement of the TIBER design. The target for the most exposed work group should be set at a level that represents the desired maximum level of occupational risk at TIBER associated with exposure to radiation. It should also be sufficiently below the regulatory limits to provide adequate confidence that the limit will not be exceeded.

In terms of defining the ALARA level of worker risk, the most appropriate design target at this stage in the objective put forward by the Department of Energy for all contractors involved in the design of nuclear facilities:

"On-site personnel exposure levels less than one fifth of the permissible limits.....should be used as a design objective."

The rationale for this 10-mSv (1-rem) design target is given in Ref. [29]. Although this definition of ALARA may not be the final word for TIBER, since the cost/benefit is unique to each facility, it is clearly the most appropriate at this stage in design.

Derivation of these limits must address the difference between the most exposed work group (e.g., mechanical maintainers, control technicians) and the total work force which includes technical support and administrative staff. Retrospective studies at Ontario Hydro (and elsewhere) indicate that, in general, when the most exposed work group averages 10 mSv/yr (1.0 rem/yr), the total exposed workforce averages 5 mSv/yr (0.5 rem/yr) of occupational dose.[30]. The ICRP guidelines state that the level of worker health risk associated with this level of exposure is equivalent to that for workers in conventional, safe industries.

The following assumptions were included in deriving more specific dose targets:

1. General Radiological Assumptions

- The dose targets (average) for facility workers are 10 mSv/yr (1.0 rem/yr) for most exposed workers, and 5 mSv/yr (0.5 rem/yr) for all exposed staff.
- The MPC-air for tritium oxide is $5 \mu\text{Ci}/\text{m}^3$ in controlled areas.
- The average protection factor for plastic suits is 100, based on the Ontario Hydro dosimetric database for tritium oxide uptake.
- The average decrease in working efficiency resulting from the wearing of gloved plastic suits is 30%. This is based on experimental and operational investigations at Ontario Hydro.
- About 75% of the annual individual whole body dose equivalent results from exposure to penetrating radiation from activation products. The remaining 25% is due to intake of radioactive material, primarily tritium. This is based on Ontario Hydro operational/maintenance experience relating to systems carrying tritiated water.

2. General Operational Assumptions

- TIBER will be staffed for 8-hour shifts, 24 hours per day, 7 days per week; anything less would not reflect the large capital investment of this research facility.
- TIBER availability will be targeted at 30% averaged over facility life.
- The total exposed work force will consist of approximately 150 persons. This is based on considerations of the staffing requirements for like-sized early fission plants, and analogous research facilities. An approximate breakdown of personnel by specialty is given in Table 5.3-2. Of these, about 25 persons would be primarily concerned with maintenance in Zone III (see Table 5.3-1), including contingency access to the reactor hall. This group is considered to be the most exposed population to whom the design objective of 10 mSv/yr (1.0 rem/yr) would apply.
- The total annual work time of an individual member of the TIBER staff is 2000 hours. This is based on fifty weeks of 40 hours each. This is an accepted value for use in radiation protection assessments.

Table 5.3-2. Composition of the radiation exposed TIBER workforce.

Work group	Approximate number of persons
Senior operators and supervisors	15
Operations	30
Mechanical maintenance	25
Electrical and instrumentation maintenance	30
Service maintenance (balance of plant)	20
Scientific and technical	<u>30</u>
Total radiation workforce	150

- The occupancy of the most exposed work group in Zone III would be about 800 hours/yr.
- Maintenance tasks which are labor intensive will be performed by removing components to hot cells, or low dose rate areas. Alternately, local shielding will be provided where possible to reduce staff exposure.
- All maintenance tasks which involve removal of shielding from the torus, vacuum chamber disassembly or sector removal will be performed remotely. Work outside shielding will be designed for remote maintenance, with provision for manned intervention in the case of unforeseen contingencies.

5.3.3 PRELIMINARY OCCUPATIONAL DOSE TARGETS

Before appropriate area dose rate targets can be derived, it is necessary to differentiate among radiation safety requirements for various work areas in the TIBER facility. Optimization of these targets would take into consideration the different hazard potentials in different equipment locations, including the potential for intake of radionuclides.

In terms of dose rate/MPC-air design targets for these areas, the proportion of dose which should be assigned to penetrating radiation versus

that for internal dose (tritium) is difficult to determine at this time. Based on operating experience, and an estimate of the relative costs associated with dose control, a ratio of 3/1 for external/internal dose will be assumed for any subsequent derivation of allowable area dose rates and airborne tritium concentrations.

Based on a reasonable assignment of area occupancies, average dose rates/MPC targets which will complement the 10 mSv/yr dose target are listed in Table 5.3-3. Note that these are averaged values. Contact dose rates for specific equipment, or local MPC-air levels, may be higher if the anticipated exposure/occupancy is proportionally lower. Also, note that no provision is made to set MPC-air limits for other airborne contaminants at this time, since penetrating radiation and tritium are judged to be the major dose contributors.

5.3.4 CONCLUSIONS

For the TIBER design study, a design target of 10 mSv/yr (average) for the most exposed work group is recommended. While design information is too preliminary to perform the required cost/benefit assessments, this target is reasonably consistent with ICRP and DOE guidelines relating to application of ALARA principles in nuclear design. In addition, this target is in agreement with what is presently considered by the Health Physics community to be excellent performance at operating nuclear power facilities. As such, it provides a reasonable default value for operations/maintenance staff at any future fusion power facility under consideration.

No major design or technological impediments are envisioned which would impede the application of the above dose target or the values listed in Table 5.3-3. Therefore, the implementation of radiation safety requirements should not constrain TIBER design beyond what is already considered acceptable at existing nuclear facilities.

Table 5.3-3. Annual average occupational dose for most exposed work group.

Radiological work area	Occupancy (h/yr)	Dose rate (μ Sv/h)	External dose(mSv)	Effective MPC-air ^a	Internal dose(mSv)
Zone I	600	0.5	0.3	--	--
Zone II	600	2.0	1.2	0.03	0.5
Zone III	800	7.5	6.0	0.1	2.0
Subtotal			7.5		2.5
Total yearly individual dose estimate 10 mSv (1.0 rem)					

a. This refers to concentrations to which the individual is exposed. If protective equipment is used, work area levels can be higher.

5.4 EFFLUENT CONTROL

Routine airborne radioactive contamination within controlled areas of the TIBER facility will include tritium, activated corrosion and erosion products, and air activation, discussed in the following subsections.

5.4.1 TRITIUM CONTROL

Tritium is judged to be the more likely critical radionuclide with respect to routine environmental impacts. Thus, TIBER effluent analyses focused on tritium.

5.4.1.1 Sources of Tritium Effluents

Major sources of airborne elemental tritium are leakage and permeation from the fueling systems, exhaust gas cleanup, vacuum/plasma chamber, and related tritium processing systems. Of specific concern is the potential leakage (as HTO) from shield and blanket module coolant lines and heat exchangers.

Chronic leakage from water-based coolant systems will be the primary source term for HTO into the containment. Canadian experience [31] with Pickering "A" CANDU units (540 MWe/unit, 287 Mg-water in moderator, 137 Mg-water in primary coolant) provides some insight. Total water leakage is 7.5 kg/h. Most of this is recovered by air dryers in the reactor hall so that the total water loss from the unit is 0.5 kg/h; 90% of the loss is from the primary coolant rather than the moderator.[31] The TIBER shield coolant conditions are similar to the CANDU moderator. Scaling to the 200-Mg TIBER shield coolant system, 840 g-water/day should be lost from the shield system to the environment, assuming continual air drying in the reactor hall.

Because TIBER is not a CANDU, however, this important estimate may be too conservative for two reasons:

1. High availability (often greater than 90%) CANDU systems cannot be maintained as often as systems in a fusion ETR, which would have an availability target (average) of about 30%. More frequent identification and correction of leaks in TIBER than CANDU's may further reduce TIBER water leakage from CANDU experience.

2. CANDU leak-tightness technology is based on equipment engineering that is 20 years old. Tighter leakage specifications can be realized at future fusion facilities, due to advances in the design, fabrication, and leakage performance of present-day components and systems.

On the other hand, the 840-g/day loss estimate may not be conservative since a fusion ETR will involve new components and technology, and using existing components in new ways. Thus, actual experience in an ETR may not match current CANDU experience.

A recent analysis for NET [32] estimates the total escape rate as 1-2 kg/day, not too dissimilar from the above 0.8 kg/day estimate by scaling CANDU experience. Thus, the estimated range appropriate for TIBER is 0.8 to 2 kg/day, assuming that very high leak tightness components are specified. With the peak 2-kg/day estimate and an average 10 Ci/liter tritium concentration, the shield water coolant system is expected to cause about 20 Ci/day effluent to the environment. Almost all of this will be airborne.

5.4.1.2 Estimate of Waterborne Tritium Effluent

Emissions to the environment via waterborne pathways are limited to two basic pathways: (a) permeation through heat exchangers from the shield coolant to the secondary coolant loop, and (b) spills from active coolant systems, coolant storage vessels, and Atmospheric Detritiation System (ADS) condensate. The former case is not anticipated to be a significant source for TIBER due to the low pressure and temperature of the shield coolant. Based on experience with CANDU moderator systems which have comparable temperature and pressure and tritium concentrations, only pinhole leaks in heat exchanger tubes will result in measurable tritium in secondary coolant circuits. In the latter case, facility design and layout will be such that spills of radioactive liquids can only drain to active liquid collection systems for appropriate treatment.

Accordingly, for the case of chronic emissions, there does not appear to be any significant impediment to keeping resultant offsite doses through waterborne releases to well below the target of 40 μ Sv/yr (4 mrem/yr) to the most exposed member of the public.

5.4.1.3 Estimate of Airborne Tritium Effluent

Based on the tritium inventories in various torus, fueling, and associated systems (Table 5.2-1), the potential for chronic escape and the experience gained from comparable systems/facilities; the following is an abbreviated list of the anticipated range for chronic escape of tritium to the environment from the major contributing sources:

<u>Source</u>	<u>Tritium Escape (range) Ci/day</u>
Shield coolant systems	8-20 (uncertain by factors of several)
Blanket test modules	1-5
Fueling system	1-3
Tritium process systems	1-3
Hot cells	0-2
Misc. (cleanup and waste systems)	0-2
Total	11-35 (mean of 23 Ci/day)

Note that no distinction is made between elemental tritium and HTO, since the residence time within confinement would likely result in the oxidation of much of the T₂ that escapes. In any case, a worst case assumption of complete conversion is not a major constraint considering the relatively low escape estimates. Some margin exists between the peak 35-Ci emission estimate and the 40-Ci/day design release target. However, the large uncertainty in estimating shield water leakage does give some concern that tritium effluents could be adequately controlled.

Overall, it does appear likely that tritium effluents can be controlled, assuming that the shield water coolant system is built to CANDU-like leak-tightness and that building gas dryers are routinely operated to collect 90-99% of the HTO leaking into the building.

It should be noted that a pressurized water coolant system would be expected to have an order of magnitude higher leakage rates, which might require tritium concentration in the water to be an order of magnitude lower. Thus, adequate control of tritium effluents from a pressurized-water version of TIBER could be prohibitively expensive.

5.4.1.4 Chronic Offsite Dose Due to Tritium

Based on the tritium release/offsite dose ratio calculated in Sec. 5.1.2, the release of chronic tritium escape into the environment (via the stack, and

without ADS operation) would result in a dose to the maximally-exposed individual of from 2--10 $\mu\text{Sv/yr}$ (0.2--1.0 mrem/yr). This is equivalent to, and likely better than, the dose performance of existent research/nuclear facilities maintaining large inventories of tritium.

5.4.1.5 Tritium Hazard Levels.

Estimates of routine tritium escape into containments from all sources range from 100 to 150 Ci/day. With a building volume exchange rate of 1%/day (480 m^3/day), the tritium concentration would be of order 0.1 Ci/ m^3 without active air dryer operation, much higher than the MPC for tritium (5 $\mu\text{Ci}/\text{m}^3$ in controlled areas) in the reactor hall and associated areas. Since these areas are not intended to be accessible during operation, these levels are acceptable. However, as noted above, routine operation of air dryers is expected to be needed to adequately control effluents.

Maintenance activities are expected to release much greater quantities of tritium into areas which may require access by facility personnel. Depending on which systems were opened for maintenance, releases into containment could increase to 500 Ci/day or greater. However, this source can be controlled via proper bakeout of large components prior to disassembly, use of contained transfer units where practical, and use of local temporary confinement linked to an atmospheric detritiation (ADS) system.

If non-routine entry into the reactor confinement area shortly after shutdown proves necessary, use of plastic suits with clean air supply will allow adequate protection against airborne contamination, even if the contamination levels are 50 times MPC, since the protection factor is 100 or greater even for the present generation of suits. Alternately, the ADS system can be used to reduce levels to less than 1.0 MPC (after about 24 hours of continuous operation) before entry by facility staff.

Normal good practice in design would require that ventilation air flow would go from less contaminated to more contaminated areas, with the bleed flow from the contaminated area going to a monitored stack for controlled release. Operation of ADS systems in this loop would be optional, depending on the buildup of atmospheric tritium.

5.4.1.6 Tritium Outgasing from Surfaces

Tritium adsorption into concrete surfaces will be controlled via the application of a steel liner to the surface of the torus hall and associated areas. This will provide a barrier to HTO, and will trap any HT that permeates into the metal surface such that it will not reach the concrete in any significant quantity before the end of facility life. As a result, although outgasing from walls will occur during atmospheric detritiation, it should not significantly hamper the reduction of MPC-air levels prior to entry by maintenance staff. In addition, the decommissioning waste volume would be reduced.

Although high quality epoxy paints and composite materials under development may prove cheaper and more effective as barriers to tritium than steel, performance data is insufficient at this time for any other recommendation.

5.4.1.7 Tritium Systems Room

The intent is to isolate tritium systems associated with exhaust gas cleanup, purification and other fuel processing systems in a separate confinement area than the torus hall. This would help to ensure that common mode events would not result in tritium from these systems will not be vulnerable along with that from torus systems. The room that has been assigned to house the tritium systems has dimensions of about $60 \times 35 \times 25 \text{ m}^3$. This is judged to be more than adequate for all tritium process systems that do not have to be located with torus systems. All systems in this room would be doubly contained, with the room confinement filling the role a tertiary containment barrier. Routine releases of tritium are not anticipated to be greater than 3.0 Ci/day . [33] Normally, ADS operation would not be required for emissions at this level, especially since they would likely be in the form of elemental tritium.

5.4.2 ACTIVATION PRODUCTS

Activation product sources include (a) activation and corrosion products in coolants, (b) accumulated erosion "tokamak dust," (c) air activation, and (d) test modules.

5.4.2.1 Activation Products in Coolants

Corrosion products originate in circulating systems such as heat removal loops, and become activated by neutron irradiation. These would normally escape via leaks or through normal maintenance activities. Typical radionuclides are Mn-56, Co-58, and Co-60. Among TIBER fluid streams, the largest activation product inventory should be in the shield coolant system, since it is the largest such system. The shield coolant inventory was estimated (Table 5.2-3).

As indicated in Sec. 5.2.2.1, the TIBER corrosion product inventory is about the same as for STARFIRE.[15] However, because the volume of STARFIRE's coolant system is about 2-3 times higher than TIBER's, the concentration suspended in the TIBER shield water at any one time will be 2-3 times that predicted for STARFIRE. Other calculations [34] performed using STARFIRE estimates [15] shows that at 10 Ci-H³/day effluent, the offsite dose from effluents from processing the coolant are about 15 times higher than tritium. The tritium and corrosion product concentrations in TIBER are both higher than STARFIRE, so the ratio of corrosion products to tritium could be about 10.

However, the TIBER corrosion product inventory estimates are too conservative in that no credit was taken for on-line routine removal of corrosion products or the low (30%) availability, which allows decay of shorter-lived isotopes like Mn-56 between operating campaigns. Since on-line removal of corrosion products is expected, the actual TIBER corrosion product inventory will be less than in Table 5.2-3. Still, the offsite dose from coolant corrosion products may be comparable to the tritium dose and should be further examined in future studies.

Another radioactivity source in the coolant is production of C-14 from addition of lithium nitrate to the water to breed tritium. Scaling from Ref. [34] suggests that the effluent dose from this C-14 will be low (4 μ Sv/yr), but not insignificant; C-14, however, is a waste management issue (Sec. 5.5.1).

5.4.2.2 Activated Erosion Products

Activated erosion products consist of various materials in the first wall which become activated, and which subsequently mechanically erode or are sputtered off as a result of plasma operation. The quantity and composition of this activated dust is highly dependent on first wall or armor material choice,

and would therefore be very difficult to characterize at this time. However, regardless of composition, mobilization and subsequent dispersal of this material to the vault atmosphere cannot occur unless there is a breach in the plasma vessel, along with a violent egress of particulates. Since this is an unlikely event, this is not viewed as a potential source of routine atmospheric contamination until such time as the torus is disassembled. At that time, loose material can be collected with remote systems before the pressure boundary is breached. In addition, transfer chambers or temporary confinement barriers will be used to control the spread of particulates while components are being transferred or maintained.

Still, the total mass of first wall material eroded over the TIBER lifetime is likely to be significant. The estimate for NET is about 300 kg.[20] Careful control of this material will be required.

5.4.2.3 Air Activation Products

Due to interaction between the flux of escaped neutrons and the constituents of the normal torus hall atmosphere, the most significant long-lived radionuclides would be Ar-41 and C-14; shorter-lived isotopes will include N-16 and N-13. TIBER air activation calculations for C-14 and N-16 (Sec. 3.3.2) show the building gaseous inventory will only be 0.1 Ci of N-16 and 0.002 Ci of C-14. In the TIBER building ($4.8 \times 10^4 \text{ m}^3$), the resulting air concentration is already below the MPC-air for public exposure; thus, with the further dilution from the building to the environment, the offsite doses should be insignificant. However, it was beyond the scope of the study to estimate neutron streaming effects, which would be expected to further increase air activation levels. Still, although the emission of these radionuclides will have to be monitored in the building exhaust, they are not envisioned as being significant contributors to offsite dose to the public.

Although the present TIBER design uses nitrogen, rather than air, as the cover gas in the reactor hall, activation calculations with air are adequate. The present ventilation system (Sec. 5.6) only removes oxygen and water vapor from the building, leaving nitrogen and argon. Since nitrogen is already 78% of air, a 99%-N₂ atmosphere will not differ significantly in activation characteristics from air. Oxygen is not a major contributor to air activation products, other than short-lived N-16.

Because air activation products are found to be insignificant, the partition of effluent dose (Table 5.1-3) in future studies for shielded, superconducting machines does not need to include a contribution for air activation.

5.4.2.4 Test Modules.

TIBER resources were inadequate to estimate the possible contribution to effluents from operation of the various blanket test modules. However, they are not expected to add significantly to the total activation product effluent because (a) collectively they are small relative to the basic TIBER machine and (b) each is presumably based on a technology thought capable of extrapolating to commercial-scale systems that could meet stringent regulatory requirements. On the other hand, by nature, test modules may not perform as expected, so routine monitoring of effluents from test modules will be necessary. The assumption that test modules will not be major activation product effluent sources should be checked in future studies.

5.5 WASTE MANAGEMENT AND DECOMMISSIONING

Besides accidents, effluents, and maintenance, proper design minimizes waste and decommissioning hazards. Before an ETR could be operated, detailed plans for handling wastes and for decommissioning would likely be required. To reduce hazards (real and perceived), the targets (not requirements) are to qualify all wastes as low-level and to be able to return the site to non-nuclear use.

5.5.1 LOW-LEVEL WASTE (LLW) EVALUATION

Waste sources will include (a) disposal of the shield and divertor following operation, (b) processing of coolant water during operation, and (c) disposal of test modules. Once radioactivity is generated, it must go somewhere. Processing coolant water to reduce hazards from accidents shifts the radioactivity to a waste hazard. In the U.S., radioactive waste is either low-level or high-level. No high-level disposal site is likely to operate this century in the U.S. Clearly, all fusion machines, including an ETR, should avoid producing any high-level waste.

Tritiated wastes are a concern, but tritium alone does not make waste high-level. Tritiated wastes were not studied in TIBER since it is believed that related problems are solvable with generally known techniques.

5.5.1.1 Evaluation Methodology

Two pathways exist to avoid high-level waste disposal: (a) qualify waste as low-level or (b) recycle it. Although recycling is likely relevant for later fusion power plants, recycling is thought less applicable to TIBER because all components are one-of-kind and a fusion-specific recycling capability will not yet exist. Therefore, TIBER analysis focused on evaluating whether TIBER wastes could qualify as low-level, i.e., pass the isotope concentration limits in 10CFR61.[4]

Three problems exist with the current 10CFR61: (a) errors in the calculations of isotope limits, (b) no limits for many fusion-specific isotopes, and (c) uncertainties in implementation. Using the 10CFR61 methodology, Fetter [35,36] has calculated correct values for currently-listed isotopes and new values for unlisted isotopes. As shown in Table 5.5-1, the concentration limits are determined by the waste form. The more leachable the

waste form (and chemical element involved), the lower the concentration limit. Fetter's values all assume that the waste is embedded in metal; the range in values results from different assumptions as to the corrosion and leaching rates.

The Waste Disposal Rating (WDR) is defined as the specific activity, A_i , divided by the isotope concentration limit, C_i , summed over long-lived isotopes (half-life over 5 years):

$$WDR = \sum A_i/C_i.$$

Thus, wastes with WDR under one qualify as low-level waste. For present purposes, the WDR is evaluated using both the legal and Fetter concentration limits.

The lack of some fusion-specific isotopes, and possible errors, in 10CFR61 create an uncertainty. The Fetter-corrected values for isotopes are sometimes higher (easier to meet) than the current legal limits. Unless and until the legal values are corrected, one could not dispose of wastes as low-level unless they met the 10CFR61 values. Of course, instead of relaxing the isotope concentration limits, regulatory officials could decide to change the underlying criteria and leave the limits as they are.

On the other hand, 10CFR61 states that wastes containing isotopes (except alpha-emitters) not listed in 10CFR61 legally qualify as low-level waste regardless of concentration. However, an actual attempt to dispose of fusion waste with high concentrations of long-lived isotopes not listed in 10CFR61 would likely be difficult.

Three other uncertainties with implementing 10CFR61 are:

1. Currently in the U.S., one cannot dispose of chemically hazardous wastes in a LLW disposal site, or vice versa. Thus, one is not sure how "mixed" wastes (chemically and radiologically hazardous) from TIBER would be handled. They would likely have to be handled separately from non-chemically hazardous wastes.
2. The 10CFR61 methodology is based on isotope concentrations. In principle, one could lower the concentration by diluting waste, i.e. adding new mass to the system. However, informal conversations that S. Fetter [37] has had with NRC officials and that S. Piet [38] has

Table 5.5-1 Isotope concentration limits for low-level wastes.^a

Isotope	Half-life (yr)	Isotope concentration limit (Ci/m ³)		
		In activated metal		Non-metal
		10CFR61	Fetter	10CFR61
C-14	5,730	80	700-7,000	8
Ni-59	76,000	220	900	(d)
Ni-63	100	7,000	1.0e6-1.0e7	700
Sr-90	29	(d)	1.0e6-9.0e6	7,000
Nb-92 (c)	3.7e7		0.2	
Nb-94	20,000	0.2	0.2	(d)
Mo-93 (c)	3,500		300	
Tc-99	213,000	(d)	0.2-2	3

- a. Only the least stringent concentration limits, for Class C wastes, are considered here. Few of the TIBER wastes would qualify for the most stringent, least hazardous, Class A.
- b. Fetter [36] has calculated limits for isotopes in activated metal with a range of leachability and corrosion characteristics; the range shown is from worst metallic waste form to best form.
- c. Isotope not listed in 10CFR61.[4]
- d. Waste form not explicitly indicated in 10CFR61 for this isotope; here it is assumed that the value for the other waste form applies.

had with INEL waste management personnel suggest that dilution would be disapproved.

- Another possible method of lowering concentrations would be to average concentration over the waste that would be disposed of in a single container, e.g., a 55-gallon drum. Some averaging would likely be approved. However, it is not known if officials would approve of carefully re-arranging waste from different parts of the facility to tailor individual waste drums to just meet the isotope concentration limits.

5.5.1.2 Structure Evaluation

The major components to be disposed are:

1. Shield structural walls and coolant piping,
2. Beryllium balls in first 410 mm of the outboard shield,
3. Steel balls in the outboard shield,
4. Tungsten blocks in the inboard shield, and
5. Tungsten divertor plates.

The divertor plates were not analyzed; depending on alloying constituents and impurities, the plates could qualify as LLW. The tungsten shield blocks qualify as LLW unless alloying or impurity elements force the waste to be high-level (HLW); the current analysis assumed W-2Ni-1Fe. The beryllium balls qualify as LLW, but "mixed" LLW. The two major uncertainties evaluating the shield components are (a) which of the shield components will be disposed of separately and (b) how large a volume may one average over in calculating isotope concentrations. Table 5.5-2 shows several cases with averaging over either the first wall or the entire inboard, or outboard, shield component:

- Concrete-filled--Shield disposed of as unit. Beryllium balls, tungsten blocks, and steel balls stay in shield pieces; coolant spaces flushed to remove water and filled with concrete.
- Crushed--Shield disposed of as unit. Beryllium balls, tungsten blocks, and steel balls stay in shield pieces; water removed; pieces crushed to reduce volume.
- Empty and Crushed--Beryllium balls removed from outboard side; tungsten blocks removed from inboard side. Steel structure and steel balls are crushed to reduce volume.

The concrete-filled case has the greatest reduction in isotope concentration but involves dilution of the system by new mass; it is unlikely to be approved. Note, the cost of waste disposal scales as the volume; hence the concrete-filled case would be the most expensive.

The crushed and empty-and-crushed cases do not involve dilution but are based on averaging over the shield components. If chemically-toxic beryllium

Table 5.5-2 Waste disposal rating evaluation of shield components.^a

Component ^b	<u>Concrete-filled</u>		<u>Crushed</u>		<u>Empty-and-Crushed</u>	
	10CFR61 values	Fetter values	10CFR61 values	Fetter values	10CFR61 values	Fetter values
Outboard first wall	23.6	22.3- 57.9	23.6	22.3- 57.9	23.6	22.3- 57.9
Average outboard ^c	0.11	0.08- 0.21	0.17	0.13- 0.33	0.23	0.18- 0.46
Inboard first wall	15.3	14.1- 36.4	15.3	14.1- 36.4	15.3	14.1- 36.4
Average inboard ^d	0.27	0.32- 0.79	0.34	0.40- 0.99	2.6	3.1- 7.5

- a. Low values preferred; WDR under one indicates qualification as 1. Both legal [4] and Fetter [36] isotope concentration limits considered. Fetter values range from worst metallic waste form to best form. Concrete-filled, crushed, and empty-and-crushed refer to different scenarios for handling waste; see description in text. The concrete-filled scenario is unlikely to be approved. WDR calculated at shutdown; however, WDR is not sensitive to waiting even 10 years for disposal. The PCA composition was taken from the BCSS [12], which has low Nb content. Attaining low levels of unfavorable elements like niobium could be difficult and expensive.
- b. 100% of the C-14 in the shield water coolant and corrosion layer is assumed to be flushed from the shield before disposal; 90 to 99% removal of C-14 needed for C-14 not to add significantly to the WDR values given here.
- c. The contributors to WDR are Ni-59 (0.94%), Ni-63 (38.9%), Nb-94 (51.3%), and Tc-99 (8.8%), using 10CFR61 values.
- d. The contributors to WDR are Ni-59 (0.56%), Ni-63 (21.2%), Nb-94 (65.3%), and Tc-99 (12.9%), using 10CFR61 values.

and lead (used to thermally bond the tungsten blocks to structure) can be disposed of with the shield steel, then the crushed case may be applicable. If, however, the mixed wastes (beryllium, lead-contaminated tungsten) have to be handled separately, then the empty-and-crushed case is applicable.

Table 5.5-2 indicates that the structural shield waste is LLW unless:

1. One cannot average over the entire sector,
2. One removes the tungsten blocks and disposes of them separately, (i.e., chemically-toxic lead forces separate disposal of tungsten),
3. A high activation element, e.g., rhenium, is added to the tungsten, assumed here to be W-2Ni-1Fe without any impurities,
4. The niobium content in the steel is higher than assumed here (0.03 wt%), or
5. 1-10% of the C-14 produced by the lithium nitrate is not removed from the shield coolant passages.

It seems likely that one of these conditions could exist, i.e., likely that the shield structure and/or tungsten shield blocks would be HLW. The lower WDR for the outboard shield, versus inboard, results from the steel balls diluting the high concentration at the first wall.

The individual tungsten divertor plates and shield blocks may be LLW depending on composition.

5.5.1.3 Coolant Evaluation

To provide for tritium breeding, lithium nitrate is added to the shield water coolant at $160 \text{ g-LiNO}_3/\text{liter-water}$. The resulting nitrogen in the water leads to C-14. This C-14 will be eventually removed in coolant processing units and appear in associated waste streams, e.g., resin beds and filters. 18,800 Ci of C-14 are produced over the TIBER lifetime for an average of about 2000 Ci/yr. For a 1000-MWe/3000-MWh water-cooled fusion plant, the estimated volumes of coolant-associated wastes are $205 \text{ m}^3/\text{yr}$ of solidified concentrates and $155 \text{ m}^3/\text{yr}$ of filters, filter sludges, resins, etc.[34] For TIBER, a scaled-down estimate (scaled using thermal power level) is a total of about $40 \text{ m}^3/\text{yr}$ of coolant-associated wastes. Thus, the average C-14 concentration in waste will be 50 Ci/m^3 . However, one expects that the C-14 will preferentially

concentrate in some portions of the wastes, with a correspondingly higher concentration, easily as high as 500 Ci/m^3 .

Unlike structure waste disposal, these wastes will not be in non-leachable metal so that the more stringent isotope concentration limits apply. The 10CFR61 limit for C-14 is 8 Ci/m^3 ; therefore, at least some coolant wastes from TIBER operation will not be LLW because of C-14 production.

If Fetter's calculations are ultimately adopted into 10CFR61, the C-14 waste may qualify as LLW. Fetter's C-14 concentration limits for isotopes in activated metal range from 700 Ci/m^3 for the worst case to $7,000 \text{ Ci/m}^3$ for the best case for a perfect waste form that did not corrode and had a low leach rate. With special processing, C-14-contaminated coolant wastes might be made into waste forms approximating metallic and qualify for isotope concentration limits over 1000 Ci/m^3 , in which case coolant waste would be LLW. This processing would likely be expensive and would succeed only if the NRC significantly changed 10CFR61.

The other long-term radioactivity in the water is corrosion products. Since the shield structure itself, averaged over the volume, apparently qualifies as LLW, then the metal corrosion products in the water, and wastes associated with them, should also qualify as LLW. This agrees with the experience with water-cooled fission reactors; coolant-associated wastes are LLW. Specifically, the WDR for the corrosion product inventory in Table 5.2-3 is only 0.007 assuming disposal in 40 m^3 of waste. The waste disposal problem from the TIBER shield coolant is production of C-14 from the lithium nitrate, rather than corrosion products.

5.5.1.4 Test Module Evaluation.

Although direct evaluations of various test modules were not performed for TIBER, one can estimate the WDR for some materials by scaling from the BCSS.[12] The fluence seen by any individual test module is less than 3 MW-yr/m^2 , versus 10 MW-yr/m^2 in the BCSS. The BCSS WDR was calculated using exposure at the first wall, versus averaging over a test module. In Table 5.5-2, the ratio WDR for first wall to average inboard/outboard ranges from 5 to 270. Thus, depending on design and how different materials in the test module are processed for disposal, the WDR for TIBER (averaged over test module) is 15--800 times lower than the WDR for BCSS materials exposed at the first wall.

From this scaling, using several materials in test modules would not produce HLW: V15Cr5Ti, Tenelon (a manganese-based austenitic steel), low-activation ferritic steel, beryllium, lithium oxide, flibe (LiF-BeF₂), and elemental lithium.

Depending on design and waste processing, using these materials in test modules may generate HLW: PCA, HT-9, lithium aluminate, and LiPb.

These approximations agree with the preceding structural evaluation--beryllium is LLW, and PCA (in TIBER) may qualify as LLW depending on how the waste is processed.

5.5.2 DECOMMISSIONING APPROACH

The decommissioning program for a fusion ETR would have to:

- Meet regulatory requirements, e.g. personnel exposure limits,
- Define and document a decommissioning scenario, and
- Establish and document an organizational structure to implement the decommissioning program.

5.5.2.1 Decommissioning Scenario

During the conceptual design phase, a reference decommissioning scenario should be produced. This scenario should consider all relevant IAEA guides as well as other applicable regulatory guides. At a minimum, the scenario must include a decommissioning schedule and conceptual dismantling and disposal plans. The scenario should be in sufficient detail to allow a personnel dose estimate to be made. The decommissioning scenario, which will reflect best current engineering practice and current decommissioning philosophy, will be reviewed as part of the overall design review and approval.

5.5.2.2 Implementation of a Decommissioning Program

The program will take advantage of existing design verification programs as much as possible. It will commence at the conceptual design phase with the definition of a decommissioning scenario. The scenario will then be used at subsequent design phases to insure decommissioning is given proper consideration. Existing verification programs will be modified to accommodate the decommissioning program. The level of detail to which decommissioning is

addressed should recognize the lengthy time span between design and decommissioning, and the changes in technology and philosophy that may occur.

Based on the reference decommissioning scenario, decommissioning requirements for the reactor must be defined and recorded. The design description document of those systems in which decommissioning was identified as a constraint will contain a section in which its decommissioning is outlined. This section will also describe the effect of its design on the decommissioning of other systems or components. Throughout the operating life of the reactor, any modifications must take decommissioning into account to the same degree as for the original design. Records relating to decommissioning should be kept for the life of the facility.

5.5.2.3 Decommissioning the TIBER Design.

A cursory examination of the TIBER design indicates that decommissioning should not pose unusual problems, with three possible exceptions:

1. The contamination of tritium systems by gamma-emitters could make decommissioning of the tritium systems more difficult than if they were only contaminated by beta-emitting tritium.
2. The contamination of components by C-14 may complicate disposal.
3. Any significant deformation of components during an accident could make removal/recovery/decommissioning difficult. The TIBER maintenance philosophy is that all components can be removed remotely--if the original design configuration is maintained. However, if an accident caused components to deform or weld against adjacent components, then decommissioning would be more complicated.

5.6 SAFETY CONSIDERATIONS IN BUILDING DESIGN

Some scoping calculations investigated how the reactor hall building could be configured to improve safety and environmental performance, but at minimum cost and complexity.

5.6.1 ROLE OF THE BUILDING IN SAFETY

The building must separate the reactor hall from the external environment, in both directions. In TIBER, this includes

1. Controlling routine effluents to the environment,
2. Controlling accidental releases to the environment,
3. Preventing external forces and missiles (seismic, winds, floods, aircraft) from damaging the machine, and
4. Controlling ingress of oxygen into the building.

The building has two accident functions, structural integrity and confinement/containment integrity.

Even if TIBER achieves "passive safety," the building has a structural integrity function. Many passive safety functions of a design are only valid if the machine geometry remains basically intact.[2,3] This is even more true of active safety systems. The reactor hall building must help maintain the basic machine geometry by protecting it against external events. Therefore, there are two design targets related to external events:

1. The safety-related features of the fusion machine must withstand all credible earthquakes.
2. The building must withstand credible external events (earthquake, aircraft, winds, floods) to the extent of not collapsing and failing the fusion machine itself.

These do not apply to the containment/confinement function, only the structural function of the building. Quantification of these design targets depends on the country, regulatory body, and site.

The building also has confinement/containment functions, even given the

10-mSv design target to limit the maximum possible offsite dose without active means, i.e., "passive safety":

1. The building helps limit effluents during routine operation,
2. If the machine design does not achieve passive safety, the building may have to limit accidental radioactivity releases, and
3. The ALARA principle may lead to requiring additional radioactivity control even if the building achieves passive safety.

Thus, if the machine does not achieve the 10-mSv design target, the building provides a backup. If the machine does achieve the 10-mSv design target, one might be able (or forced by ALARA to) further reduce offsite doses by cost-effective building design.

5.6.2 POSSIBLE OVERPRESSURE SOURCES

Before proceeding further in examining the building, one must identify possible overpressure sources in the building. Note, however, that the following calculations are generally simplistic and only serve to identify whether additional work is needed.

5.6.2.1 Test Module Overpressure Sources

Possible test module overpressure sources include loss of coolant (water, helium, flibe, LiPb, or lithium) and chemical reactions (LiPb, lithium). Using the methodology in Ref. [39], a loss-of-coolant from a test module containing 2 m^3 -water at 15 MPa and 300°C could raise the building pressure ($4.8 \times 10^4 \text{ m}^3$ volume) about 5 kPa. The overpressure from a helium or flibe test module would be insignificant, under 1 kPa.

Without chemical reactions, spills of LiPb or lithium would not increase the building pressure. Scaling down from past lithium-air fire calculations, [26] one finds that the 1 m^3 -Li from a module, if burned, could raise the building pressure only 13 kPa. Although lithium-air reactions are not considered likely, because the building cover gas is nitrogen, lithium-water/steam reactions are likely capable of causing similar over-pressures. Overpressures from lithium-nitrogen reactions would be lower than from lithium-air. However, a more serious issue is hydrogen combustion via LiPb-water, lithium-water, or carbon tile-water reactions. Fortunately (Sec. 5.2.4.2), the

total amount of hydrogen available in the building is below the flammability limit, even if air ingress occurred. Thus, the only major hydrogen issue is combustion/explosion within the torus, which would require air ingress from outside the building into the torus. The most likely pathway for air ingress to the torus, from outside the building, is a break in the neutral-beam lines. Such an event could damage the torus but would be unlikely to damage the building.

Thus, neglecting hydrogen combustion in the torus, the maximum overpressure from failure of all test modules is under 20 kPa. In fact, if lithium and water-cooled test modules are not in service simultaneously (a good idea), the maximum overpressure of test modules at any one time is under 15 kPa.

5.6.2.2 Cryogen Release Overpressure

Scaled to TIBER, the building overpressure resulting from cryogen helium release is 7-12 kPa/Mg-He-released [27,39,40] depending on the heat transfer from the coils to the helium. In the BCSS design, cryogen release from one TF coil could produce 7 kPa.[12]

Because of the high decay heat in the inboard shield, complete release of all magnet helium (all TF coils, inboard OH coils) should be assumed credible in the current design. During a full-scale LOCA in the inboard shield, the tungsten-generated decay heat will heatup the inboard legs of the superconducting coils. Furthermore, a loss-of-site power condition will lead to cryogen heatup in minutes to hours depending on design.[10] If all cryogen is released, the building overpressure could exceed 100 kPa. However, initially, the building pressure might fall as the atmosphere is initially cooled. More detailed calculations are needed.

5.6.2.3 Shield Coolant Overpressure

Without heat transfer to the water, a loss-of-coolant from the shield will not pressurize the building because of its low temperature and pressure. However, if heat transfer to either water coolant or building atmosphere is significant, then the resulting overpressures may be a concern. Fortunately, because the total heat capacity of the building is high, about 2 GJ/°C, and the release of decay heat is slow (130 GJ in first day), the percent of decay heat transferred to gases is expected to be low, under 10%. However, further

analyses of loss-of-coolant in the inboard shield are needed to determine where that decay heat goes, specifically how much it may heat the building atmosphere.

Treating the building atmosphere as an ideal gas, if 10% of the 130 GJ of decay heat released within 1 day after shutdown is transferred to the building cover gas (nitrogen), the overpressure would reach 75 kPa. Unlike a pressurized-water release, however, this pressure buildup would be fairly slow. Note, steam's specific heat is higher than nitrogen or air, so a steam-nitrogen mixture would exhibit lower overpressure for the same heat transfer.

5.6.3 UNDERGROUND SITING CONSIDERATIONS

In the last decade, there have been several major studies performed to examine the feasibility and cost/benefit of underground siting for fission plants. Two of these comprehensive studies, performed by the California Energy Commission [41] and Ontario Hydro [42] are judged to have results and conclusions relevant to the siting of a TIBER facility:

Advantages of an Underground Site:

1. Increased strength of containment structures
2. Enhanced protection against external hazards
3. Reduced seismic motion (especially in deep rock caverns)
4. More effective barriers against air/water radionuclide escape
5. Natural shielding.

Disadvantages:

1. Major increases in capital costs of construction
2. Longer construction schedule
3. More difficulty/expense in operations/maintenance
4. New safety issues (radionuclide migration, rock stability, ground activation).

The most significant finding was that the capital cost of underground construction would be higher than for surface siting, with no clearly significant benefit; that is, for a shallow excavate-and-cover concept the costs were estimated at 14 to 25% more expensive than surface siting. For deep caverns (in granite) the cost was estimated at 31 to 36% higher.

The Ontario Hydro study showed that precambrian rock was capable of sustaining the mechanical and thermal stresses associated with major energy releases from fission reactors, and that it would provide a tight, massive barrier against radiation release. However, the need for substantial overpressure protection does not arise in the present TIBER design concept; deep rock placement would represent a major over-design of the containment.

The California study noted that the use of a filtered atmospheric venting system on a surface plant design resulted in reductions to public risk (due to major accidents) almost as broadly effective as underground siting, yet at only 2% increase in base costs. This study goes on to state that underground siting in California should not be mandated because of "the existence of what appears to be moderately effective and less expensive alternatives."

The conclusion at this time is that underground siting would not be a cost effective design alternative for the TIBER/ETR next-step machine.

5.6.4 CONTAINMENT VERSUS CONFINEMENT

One can consider three general containment/confinement approaches:

1. Containment building--Building is designed to withstand any credible internal pressure source without failure or release of radioactivity to the environment. This is a PWR-type building in the U.S.
2. Filtered, vented confinement building--Building includes a filtered vent system to relieve pressure. The building must still be designed to withstand credible driving pressures, i.e., whatever pressure drop exists through the filter system. Thus, one ensures that pressure is relieved by gases passing through the filter system, rather than leaking.
3. Simple confinement building--Building includes exhaust/filter system, but is not designed to withstand credible driving pressures. This is appropriate if confinement of routine effluents is a concern, but accidental releases are not.

Note, any of these can achieve adequate structural integrity; the present issue is the approach to controlling radioactivity. As indicated in Sec. 5.6.1, the containment/confinement integrity of the TIBER reactor building is only considered a backup to proper machine design. Therefore, the intent is to find

the lowest-cost option that would insure keeping maximum offsite doses under 10 mSv (1 rem). As Sec. 5.6.2 indicates, possible overpressures could exceed 100 kPa (cryogen release or decay heat); therefore, a containment structure for TIBER could be undesirably expensive. Thus, the TIBER approach was to attempt design of a filtered, vented confinement building.

5.6.5 FILTERED VENT SYSTEM

The key issue determining acceptability of a filtered, vented confinement building, as opposed to an expensive pressure-tight containment building is whether an adequate filter system can be designed. TIBER resources were not adequate to design such a system but were sufficient to indicate an approach that should work.

Possible accidental insults to the building and filter system include:

1. Overpressure sources exceeding 100 kPa
2. Radioactivity sources
 - Elemental tritium
 - Tritiated water
 - Volatilized metal oxides
 - Volatilized metal dust
 - LiPb and lithium combustion products
3. Hot, humid environment, e.g., inboard LOCA dominated accident
4. Cold, dry environment, e.g., cryogen-release dominated accident

Furthermore, one would like to make the system as passive as possible to increase reliability and demonstrability.

The wide range of insults listed above suggests that the filter system be a combination of specialized processing and filtration systems for routine operation with a generic system for accidents. The suggested system is shown in Fig. 5.6-1.

5.6.5.1 Specialized Processing Units

During operation, specialized optimized systems will process the tokamak exhaust, shield coolant, and building atmosphere. The building atmosphere processing unit is responsible for (a) converting elemental tritium to HTO, (b) removing (tritiated) water vapor, (c) removing oxygen, and (d) pulling a

slight negative pressure on the building. The building in-leakage rate is assumed to be 1% volume/day or $480 \text{ m}^3/\text{day}$. Depending on the efficiency of oxygen removal at low oxygen concentration, one would either

1. Simply pull a slight pressure on the building. Process enough building gas to keep oxygen content to about 1%, exhaust $480 \text{ m}^3/\text{day}$. Exhaust from the building would be enriched in oxygen and water vapor.
2. Bleed nitrogen into the building, while pulling slight pressure on building atmosphere. Building exhaust would therefore exceed $480 \text{ m}^3/\text{day}$ but limiting the nitrogen content becomes more likely.

A commercially available unit [43] can provide $2000 \text{ m}^3\text{-N}_2/\text{day}$ at 95-97% purity for \$110,000 by separating oxygen and water vapor from air. The cost may be somewhat higher in the TIBER application because of the desire to keep the oxygen content so low. Unfortunately, it appears that argon in air would follow the nitrogen so that argon activation products would still be produced.

5.6.5.2 Generic Accident Filter System

Without significantly perturbing the above systems, each optimized for a special chemistry and purpose, one also needs an accident filter system capable of handling the various insults noted above. This accident system is not intended to handle routine conditions. Thus, individual effluent streams must be processed before exhausting into the accident system as implied in Fig. 5.6-1. Routine effluent streams feed into the accident system so that active isolation and control during accidents is avoided. That is, the correct gas flow is established during routine operation. In an accident, no active systems have to operate because the building is routinely isolated except for the processing systems that already exhaust into the accident filter system.

The accident filter system should be capable of handling all possible conditions, except it will not remove elemental tritium. Inside the reactor hall, the total elemental tritium inventory is under 540 g, most at low vulnerability, e.g., trapped in beryllium balls (Table 5.2-1). In an accident, some of this tritium would be converted to HTO inside the building; this converted tritium would be trapped by the filter system. However, some of this tritium, likely under 1%, would be converted to HTO after it leaves the filter

Figure 5.6-1. Filtered, vented confinement system to control accidental radioactivity releases.

system and is in the environment. Therefore, the maximum offsite dose from an incredible total release of the 540 g H^3 would be under 3 mSv (300 mrem). (540 g H^3 as HT with 1% conversion is the same hazard as 5.4 g H^3 as HTO. This times 0.5 mSv/g H^3 gives 2.7 mSv.)

A passive system to capture this elemental tritium has not been found. An active catalyst bed to convert elemental tritium to HTO could be inserted before the accident filter system. However, currently known catalysts would likely be fouled by one of the possible accidentally conditions, e.g. steam or lithium-air fire aerosol.

The first high efficiency particulate air (HEPA) filter is expected to adequately handle most accidents. Such filters have > 99% efficiency in capturing TiO_2 , MoO_3 [44], and lithium-air fire aerosols.[45] Presumably, they would have <99% removal efficiency for most or all TIBER aerosols. The pressure drop can reach 6-9 kPa.[46] The HEPA filter should be sized to accommodate at least a $1\text{-m}^3/\text{s}$ flow to handle possible production of steam from decay heat. This is only a scaleup of 2.2 times a HEPA filter successfully tested with lithium-air fire aerosols.[45]

If the HEPA filter clogs, the submerged gravel scrubber (SGS) (Fig. 5.6-2) is an inline backup. A HEPA sized to $1\text{ m}^3/\text{s}$ would only handle about 8 kg of lithium-air fire aerosol. Complete combustion of the $1\text{ m}^3\text{-Li}$ in a lithium test module would produce about 100 kg of aerosols. Most would likely plateout inside the building. If not, the first HEPA would clog and exhaust gas would transfer to a bypass line leading to the SGS. A 20-m^3 water pool SGS, a factor-of-45 scaleup from tests,[45] would capture the lithium-air aerosols at at least 99.6% efficiency. The SGS has a 7-kPa pressure drop. It should remove any aerosol or gas soluble in water. Tritium that would otherwise be released to the environment (< 40 Ci/day) will buildup in the pool to 0.7 Ci/liter in a year, eventually requiring processing.

A LOCA-dominated accident would increase the temperature of exhaust gases, slowly raising the SGS pool temperature. A 20-m^3 -water SGS initially at 20°C could absorb 5% of the first day's decay heat before it started to boil. If future design found this inadequate (unlikely), the SGS could be increased in volume.

A cyrogen-dominated accident would initially lower the building atmosphere temperature, raising a concern that the SGS pool could freeze. If additional calculations showed that the pool temperature could be sufficiently lowered by

Figure 5.6-2 Schematic of a submerged gravel scrubber.

a cryogen release, then addition of antifreeze to the pool might solve the problem.

Finally, a second HEPA filter may be warranted to provide a final chance to capture radioactivity. Because the mass loading on this HEPA filter would be low, its pressure drop would be small compared to other parts of the accident filter system--first HEPA at 9 kPa, SGS at 7 kPa.

This accident filter system is in some ways similar to one proposed for German pressurized water reactors.[47] The system is designed to operate with 600-kPa driving pressure, considerably higher than the TIBER system. The PWR system includes a stainless-steel fiber prefilter, a HEPA or HEPA-like stainless-steel filter, an iodine filter (molecular silver sieve), and possibly a final HEPA filter. Obviously, the exact design is not useful for TIBER because the possible accident conditions are significantly different. This system is sized for 0.8-1.4 m³/s, the same as the TIBER system, and costs about \$1 million.

In summary, there are good reasons to believe that a passive, low cost accident filter system could be designed for a TIBER-like device. The building pressure required to drive gases through the system is under 20 kPa. Thus, if the building is designed to 30 kPa, with rupture disks opening to the filter system also set at 30 kPa, the confinement building approach should be adequate. Instead of an expensive containment building designed to withstand overpressures over 100 kPa, TIBER uses a filtered vented confinement building designed to 30-kPa overpressure. There is apparently little cost incentive to reduce the overpressure capability below about 30 kPa. CFFTP analysis for NET [48] indicates that there is a breakpoint or "knee" in the curve of cost versus design overpressure at about 30 kPa. Above this overpressure, "the building design gets disproportionately more expensive because standard construction (to provide shielding and keep the ceiling up) will not also cope with the dynamic load" imposed by the overpressure.

The present confinement building concept should capture 99% of all accidental radioactivity releases, except elemental tritium, without depending on active safety systems. Further detailed design would be required to verify this, especially the lack of dependence on active controls and isolation valves.

If additional heat removal capability is needed from the building, a NaK-cooled passive, natural circulation system used in EBR-II (Experimental Breeder

Reactor) could be useful. A factor-of-4 scaleup of the 350-kW EBR-II system [49] should be adequate to remove all the decay heat.

Finally, either a confinement or containment building will fail if penetrations through the building walls allow gas to escape. Proper ETR design will eliminate as many penetrations as possible and design the rest to 30 kPa overpressure. The most worrisome penetrations are the neutral-beam lines.

5.6.6 TRITIUM CLEANUP SYSTEM

The tritium cleanup system is described in Sec. 3.5. For many facilities, e.g. the Tritium Systems Test Assembly, the capacity of the emergency building atmosphere detritiation system is sized to allow manned access to the building after a certain period of time. In TIBER, however, there is no plan or ability for manned access to the reactor hall, so the manned access criterion for tritium cleanup does not apply. Instead, control of radioactivity to the environment will be the appropriate criterion.

For example, make the following assumptions:

1. Building leak rate is 1% volume/day,
2. The offsite dose target for anticipated accidents is 0.1 mSv (10 mrem) (Sec. 5.1), and
3. The highest "anticipated" tritium release into the building is 10 g-H^3 as HTO.

The reference site assumptions (Sec. 5.1) indicate a maximum offsite dose of 50 nSv/Ci H^3 or 0.5 mSv/g H^3 . Thus, the dose target implies a target to limit the tritium released to 0.2 g, or 2% of the 10 g assumed to be released. Thus, the tritium would have to be cleaned up in about 2 days. With more detailed design, this approach could determine the appropriate capacity of the tritium cleanup system. Lowering the tritium inventory, or its release frequency, decreases the appropriate tritium cleanup system capacity.

5.7 SAFETY AND ENVIRONMENTAL CONCLUSIONS AND RECOMMENDATIONS

TIBER II safety and environmental (S&E) analyses were sufficient to begin examination of several innovative features. However, many unknowns remain. The analyses are generally conservative; in some cases too conservative because of lack of resources for more detailed calculations or because of poorly understood underlying phenomena. Also, analyses focused on public safety; future analyses should include more consideration of investment protection.

5.7.1 COMPARISON OF DESIGN WITH DESIGN TARGETS

The final TIBER design was compared with pre-established design targets for accidents, personnel exposure, effluents, and waste management. No problem has been uncovered that would prohibit meeting probable S&E regulatory requirements; however, protection against some problems may be insufficient, possibly invalidating the present design concept. Also, several design targets are not met, as indicated below. These discrepancies define major issues (Sec. 5.7.2), needed future work (Sec. 5.7.3), and form the basis for overall S&E conclusions (Sec. 5.7.4).

5.7.1.1 Comparison with Accident Design Targets

Complex accident transients are difficult to analyze. Accordingly, with limited resources, TIBER II accident analyses were only adequate to search out major problem areas and were insufficient to resolve all of them. Some conservative results are listed in Table 5.7-1. The table shows both "maximum" and "average" offsite doses. "Maximum" refers to worse case weather with no credit for the confinement building; it is appropriate for determining whether the 10-mSv (1 rem) offsite dose limits for all credible events has been met. "Average" refers to average weather and takes a factor of 100 credit for confinement building, except for elemental tritium; it is appropriate for determining whether the various probabilistic offsite dose limits (Fig. 5.) for different accident frequency regimes have been met. The table shows that some design targets have not been met.

Table 5.7-1. Summary of conservative accident analyses and associated desired design features.

Hypothetical transient	Maximum ^b offsite dose(mSv)	Average ^c offsite dose(mSv)	Machine ^d damage	Desire design features
Anticipated Transients Possible frequency 1-10/yr. Offsite dose target for this frequency regime ≤ 0.1 mSv (10 mrem). Offsite dose target ≤ 10 mSv for any credible event. Offsite dose requirement ≤ 250 mSv.				
Plasma disruption vaporizes material, coupled with loss of vacuum boundary	10	<0.01	Nil?	Design for many disruptions and highly reliable vacuum boundary
Pellet injector misfire	Nil?	<0.001	Target damaged	Target area (pellet dump) should be easily replaceable; interlocks needed
Neutral beam misfire	Nil?	<0.001	Target damaged	Target area (beam dump) should be easily replaceable; interlocks needed
Low probability transients Possible frequency 10^{-2} to 10^{-4} /yr. Offsite dose target for this frequency regime ≤ 1 mSv. Offsite dose target ≤ 10 mSv for any credible event. Offsite dose requirement ≤ 250 mSv.				
Air ingress mobilizes tokamak dust	<u>75</u>	0.08	Low	Limit accumulation of dust during operation
All highly vulnerable, tritium released	8	0.08	Varies	Limit inventory, double contain highly vulnerable tritium
Inboard shield LOCA boils tritiated water	20	0.02		Find way to remove or reduce decay heat, or reduce tritium in water
U-bed burns with air ingress, releases H ₃	<u>1000</u>	1	U-bed destroyed	Prevent air ingress to U-bed; find alternative to U-bed
Shield coolant spills, mobilizes corrosion products	$\ll 10$	<0.01	Low	Remove corrosion products during operation

Table 5.7-1. Summary of conservative accident analyses and associated desired design features (continued).

Hypothetical transient	Maximum ^b offsite dose(mSv)	Average ^c offsite dose(mSv)	Machine ^d damage	Desired design features
Low probability transients Possible frequency 10^{-2} to 10^{-4} /yr. (continued)				
Divertor LOCA, plasma burns for > 1 s	Unknown	Unknown	Divertor fails	Divertor easily replaceable; find quick, passive, benign plasma shutoff method.
Very low probability ansients Possible frequency 10^{-4} to 10^{-6} /yr, Offsite dose target for this frequency regime ≤ 10 mSv. Offsite dose target ≤ 10 mSv for any credible event. Offsite dose requirement ≤ 250 mSv.				
Lithium-air fire; test module fails, air ingress into building	7	0.007	Moderate	Nitrogen cover gas; limit penetrations through reactor hall walls; passive dump tanks; partially double-walled modules
LiPb-water reactions; test module fails; water spills	9	0.009	Low?	Partially double-walled modules; passive dump tank
Hydrogen reacts; Li or LiPb module fails; water spills; air ingress	Unknown	Unknown	Major	Nitrogen cover gas; limit penetrations through reactor hall walls; passive dump tanks; partially double-walled modules
Hydrogen reacts; Water spills and reacts with C-tiles; air ingress	Unknown	Unknown	Major	Nitrogen cover gas, limit penetrations through hall walls
Structural component falls into plasma and is vaporized	≤ 1100	1	First wall coated	Reliable structural integrity; recoverable first walls
Inboard shield; LOCA; no corrective action	≥ 2000	2	Melting	Find way to remove or reduce decay heat

Table 5.7-1. Summary of conservative accident analyses and associated desired design features (continued).

- a. Transients conservatively classified by engineering judgement into possible frequency regimes, see Table 5.1-1. Frequency regime classifications take no credit for active safety systems. Underlined dose values do not meet a design target.
- b. These offsite dose estimates are conservative and take no credit for plateout within the building or capture by the confinement building. All mobilized radioactivity is conservatively assumed released to the environment. These values are compared against the 10-mSv dose limit for any credible event.
- c. These offsite dose estimates take a factor of 10 credit for having average weather conditions versus worst-case weather conditions and a factor of 100 credit for the confinement building (except elemental tritium releases). These values are compared against the dose limit for each frequency regime.
- d. In most cases substantial contamination of the reactor hall could result.

The accident frequency regimes were conservatively estimated by engineering judgement and take no credit for active safety systems. Failure rates of mature technologies may not be appropriate for an experimental machine that is pushing the frontiers of technology and physics. In fact, one might argue that if no failures occurred, especially in test modules, it would indicate that the machine design and operation were not sufficiently ambitious.

5.7.1.2 Comparison with Personnel Design Targets

Estimating personnel exposure requires predicting maintenance operations and duration, which is beyond the scope of TIBER II. However, no particular problems with the design are apparent. The major concern is whether contingency manned access can be allowed inside the reactor hall. Otherwise, if an unforeseen event damaged the machine beyond the repair capabilities of automated equipment, one would have to choose between abandoning the facility or giving undesirably high exposure to repair personnel.

5.7.1.3 Comparison with Effluent Design Targets

Based largely on experience with TSTA tritium operation and CANDU moderator water control, appropriate effluent design targets should be met.

5.7.1.4. Comparison with Waste Management Design Targets

Operation of the present TIBER II design will likely generate high-level waste (HLW), violating the design target to produce only recyclable materials or low-level wastes (LLW). Wastes from shield coolant processing will likely be sufficiently contaminated with C-14 to be HLW. Depending on how wastes are processed, how the isotope concentrations can be averaged, and material compositions, some structure and some test modules may be HLW.

5.7.2 MAJOR ISSUES

Three major complex issues become apparent in the S&E analyses:

1. Is adequate protection provided the inboard shield and divertor against loss-of-coolant accidents (LOCA), i.e., is the LOCA risk acceptable?
2. Are liquid-metal cooled test modules (and carbon tiles) sufficiently compatible with the water-cooled basic TIBER machine, i.e., is the chemical interaction risk acceptable?
3. Can generation of high-level waste be avoided?

None of these questions was definitively resolved in the study. However, the possible steps toward favorable resolution were identified and are shown in Figs. 5.7-1, 5.7-2, and 5.7-3. Generally, a mixture of (a) more detailed analyses, (b) improved design, (c) more research, and (d) changed or clarified administrative constraints are recommended to attack the problems. That is, successful resolution will require progress on multiple fronts, because each involves uncertain feasibility and/or undesirable cost-impacts.

Figures 5.7-1--5.7-3 indicate specific items that are either required or recommended for favorable resolution of these major issues. These help define future work. More details can be found in Secs. 5.2.3, 5.2.4, and 5.5.1, respectively, for these issues.

In Fig. 5.7-1, if generated, decay heat in the inboard shield must go somewhere. If it remains in the shield, the shield will melt resulting in damage and serious radioactivity mobilization. If the heat warms the inboard leg of magnets, they may be damaged and will release significant cryogens into the building, pressurizing it.

Figure 5.7-1 Possible pathways toward successful resolution of loss-of-coolant risks.

Figure 5.7-2. Possible pathways toward successful resolution of chemical reaction risks.

Figure 5.7-3 Possible pathways toward avoiding production of high-level wastes.

In Fig. 5.7-2, the TIBER II approach was to first select a basic machine technology, water coolant, and then attempt to design test modules around that choice. Test module needs should be given earlier attention in determining basic machine choices.

In Fig. 5.7-3, the requirements for qualifying wastes as low level (10CFR61) include uncertainties and errors. These should be resolved.

5.7.3 FUTURE WORK

A simple comparison of some safety-relevant parameters among CIT, TIBER, and possible commercial fusion reactors (Table 5.7-2) shows that TIBER is an appropriate aggressive step in fusion technology. However, considerable safety and environmental research will be needed to take that step with confidence and without mishap. Higher priority items for future work to resolve problems and produce a superior fusion product are listed in Table 5.7-3.

5.7.4 SAFETY AND ENVIRONMENTAL CONCLUSIONS

Several general conclusions can be reached from TIBER S&E analyses:

1. Uncertainties exist in regulatory requirements for a fusion ETR, which vary with time, country, and site. Nevertheless, appropriate design targets can be derived from potential requirements, incorporating design margins and the As-Low-As-Reasonably-Achievable (ALARA) principle. Future ETR design efforts must incorporate similar design targets.
2. No problem has been uncovered in the TIBER II design that would prohibit meeting probable S&E regulatory requirements. However, protection against LOCA's and liquid-metal/water chemical reactions may be insufficient, possibly invalidating the present design concept. Also several worthwhile design targets are not being met.
3. Protection against loss-of-coolant accidents in the inboard shield and divertor plates should be increased, including
 - Find and demonstrate a rapid (about 1 s), passive, and benign plasma shutoff method.
 - Unless more detailed analyses indicate the existing design is adequate, either reduce the amount of high-decay-heat tungsten in the inboard shield or provide passive heat removal mechanisms. The

Table 5.7-2. Comparison of some safety-related parameters among CIT, TIBER, and possible commercial fusion designs.

Safety Parameter	CIT	TIBER II ^a	Possible Commercial ^b
Tritium inventory (g)	20	1,800-3,700	2,000-10,000
Neutron fluence (MW-yr/m ²)	0.008	3.3	20-40
Availability (%)	< 0.01	30	65-90
First wall activity (Ci/kg)	300	11,000-21,000	20,000-100,000
(rem/kg-released)	0.05	2.5-5	5-20
Total activity (MCi)	1	850	1,000-10,000
Operational mode	Pulsed	Steady state	Steady state
Magnets	Normal	Superconducting	Superconducting
Liquid metal volume (m ³)	0	1-5	200-800 (c)
Blanket/shield cooling	Inertial	Convection	Convection
Blanket/shield coolant	Li ₂	Water	Liquid metal, flibe, helium, or water
Structural materials	Inconel	Austenitic steel (d)	V-alloy or ferritic steel

a. For first wall activity, the range is between the inboard and outboard first walls. Other ranges are caused by design uncertainties.

b. Range results from differing concepts and design uncertainties.

c. If liquid metal is used as coolant or breeder.

d. Basic machine has austenitic steel structure; however V-alloy and ferritic steel are used in test module structure.

Table 5.7-3. Summary of major future work to favorably resolve safety and environmental issues for a fusion ETR.

More Detailed Analyses:

1. Heat removal from inboard shield during loss-of-coolant.
2. Behavior and consequences from injection of foreign materials (coolant, air, dropped probes, fallen tiles) into the plasma.
3. Perturbation by test modules of machine performance and safety.
4. Performance of passive, filtered vented confinement system.
5. Potential for, and control of, plasma overpower transients.
6. Behavior of corrosion products.
7. How to minimize the risk from incorrect computer or human control.

Design Improvements to Consider:

(Analyses and/or research will show which are necessary in future designs.)

1. Select different salt in aqueous salt approach, replacing lithium nitrate, which produces C-14.
2. Reduce tungsten in inboard shield, reducing decay heat.
3. Provide double first wall for liquid metal test modules, reducing frequency of spills.
4. Choose shield technology more compatible with test modules, possibly helium coolant.
5. Eliminate liquid metal testing.
6. Find better structural material with lower long-term and short-term activation, lower decay heat.

Research Data:

(Some research will not be completed until initial checkout tests in an ETR. For example, the safety analyses of an ETR will assume some method of plasma control and shutoff, which will have to be validated by checkout tests.)

1. Examine quick, passive, benign methods to turn off plasma when off-normal conditions in shield, divertor, or test modules require.
2. Study and report off-normal events at JET and TFTR. (Reporting of off-normal events at TSTA already in place).
3. Study tritium inventory and behavior in plasma-facing materials.
4. Study activation product behavior, especially "tokamak dust" and volatilization of test module materials, PCA, tungsten, and plasma-facing materials.
5. Study consequences of liquid metal/water reactions in TIBER-relevant geometries.
6. Study carbon-tile chemical reactions, with air and water.

Table 5.7-3. Summary of major future work to favorably resolve safety and environmental issues for a fusion ETR (continued).

Change or Clarification of Administrative Constraints

1. Report off-normal events at JET and TFTR.
 2. Correct 10CFR61 concentration limits, especially C-14.
 3. Clarify position on "mixed" (chemically and radiologically hazardous, e.g., beryllium) wastes.
 4. Clarify position on averaging over large waste structures, e.g., sector-sized shield pieces.
 5. Define external events that ETR must handle, e.g., seismic, aircraft impact, winds, floods.
-

tungsten inboard shield causes TIBER to have a worse decay heat problem than most past commercial fusion designs or NET. The viability of the tungsten shield concept requires finding a satisfactory way to remove decay heat.

4. The present design uses nitrogen as a reactor hall atmosphere, limiting liquid-metal volumes in test modules to about 1-m^3 each, and partial double-containment of test modules to reduce the risk from liquid metal (lithium, lithium-lead) and carbon tile chemical reactions. Risk assessments are required to determine if present design is adequate.
5. More analyses are required to determine if the design is adequately tolerant of plasma disruptions.
6. Adequate control of personnel exposure is considered likely. However, contingency manned access to the reactor hall should be provided because of the likelihood of unforeseen damage states.
7. Adequate control of routine effluents is considered likely, assuming tritium concentrations of 10 Ci/liter-water in low pressure shield coolant and very high leak-tightness components and air detritiation.
8. Some High-Level Waste (HLW) would likely be produced from TIBER operation, especially C-14 from coolant processing, other possible HLW are:
 - PCA shield structure if averaging is limited or W-blocks removed
 - Tungsten blocks and divertor plates if composition poor
 - Test modules (PCA, HT-9, LiAlO₂, LiPb).

Unfortunately, the role of S&E analyses is to first find problems before being able to resolve them. Distinctly positive aspects of a design often receive less attention. Key positive features of the TIBER II design from the S&E perspective include:

1. TIBER operation will be steady state, increasing reliability and decreasing failure rates compared with pulsed operation.
2. Designing TIBER for multiple physics operating points increases design margins and flexibility for actually operating the machine.
3. The capability for replacing any failed (but not deformed) components substantially increases the chance of completing the research mission. It also decreases motivation for attempting to run the machine in a partially failed, and possibly hazardous, state.

5.7.5 SAFETY AND ENVIRONMENTAL RECOMMENDATIONS

Items for future work associated with these findings were listed in Table 5.7-3. The major S&E recommendations are:

1. Consider the range of desired nuclear technology testing and the materials, size, and geometry of test modules before selecting a technology for the basic machine. The present water-cooled shield concept has limited compatibility for testing lithium or lithium-lead test modules. More compatible shield concepts should be pursued, including helium coolant. This becomes a necessity if sector-sized test modules are used.
2. Keep the TIBER features of steady state operation, multiple operating points, and complete disassembly capability.
3. Consider safety ramifications of key design decisions. For example, if the plasma current increases, the disruption effects increase. If the major radius increases, the first wall surface area increases, along with the radioactivity inventory and the amount of activated erosion dust. However, increased major radius might allow decreased use of tungsten in the inboard shield, reducing decay heat and LOCA concerns.

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6.0 PHYSICS AND TECHNOLOGY R&D NEEDS FOR THE TIBER (ETR/ITER)

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6. PHYSICS AND TECHNOLOGY R&D NEEDS FOR THE TIBER (ETR/ITER)

6.1 PHYSICS R&D

At the completion of the TIBER/ETR study, and the beginning of the new ITER study, it is appropriate to reassess the physics R&D needs. In each section, critical issues are identified and opportunities to test those issues in existing or planned devices are suggested. Inclusion of a given experiment in this assessment does not imply that a group has made a commitment to carry out the research. Figure 6.1-1 indicates a schedule for the ETR (TIBER or ITER) which calls for full steady-state current drive after an initial operational phase with inductive drive (the inductive pulsed phase beginning ~1998 and the steady-state phase beginning after the year ~2002). Thus, the level of importance in resolving a given critical physics issue is indicated in Tables 6.1-1 to 6.1-5 separately for the initial and later operational phases. A short list of major contributing experiments is provided in Table 6.1-6, accompanied by a simple formulary for normalizing the confinement and beta capabilities of those experiments. For brevity, we have not attempted to discuss the role of every possible contributing experiment, even though many small Tokamaks could undoubtedly contribute valuable data for the ETR.

For each mode of operation envisioned there are physics issues relating to start up, control burn, disruption, and plasma turn off. For these different modes of operation, control will be exercised by various machine "knobs" such as pellet fueling rate and current profile control (when current drive is used) to avoid the various instabilities and to optimize the desired plasma operating point. Some of these issues are discussed in this report. With respect to ignited burn [the initial inductively driven Phase I and part of Phase II of TIBER (see Fig. 6.1-1), beginning ~1998], data from CIT during 1993-1998 will clearly be important to resolving the questions related to operation of an ETR plasma in which alpha heating dominates. With respect to steady-state current drive in TIBER (Phase III) beginning ~2002, data from DCT, a proposed (but not yet funded) new steady-state superconducting device in the US, could provide experience in 1995-1998 in optimizing current drive,

ETR SUPPORTING PHYSICS RESEARCH AND DEVELOPMENT

Figure 6.1-1. ETR supporting physics research and development schedule.

Table 6.1-1. ETR physics R&D--confinement.

	Critical issue	Facilities	Contribution	Contribution
			to ETR Phase I	to ETR Phase II
1.	Sawtooth transport--comparison of theory and experiment, α -effects	DIII-D, TFTR, C-MOD, CIT, T-14, T-15, PBX, ASDEX/u, JET, JT-60	useful	important
2.	Transport theory for ETR conditions, understanding of H-mode* phenomena	DIII-D*, TEXTOR, TFTR JET*, JT-60*, ASDEX/u* JFT2-M*, PBX*, CIT	crucial	important
3.	Global empirical scaling of H-mode confinement	DIII-D, C-MOD, CIT, JET, JT-60, ASDEX/u, JFT2-M, PBX, ASDEX	crucial	important
4.	Effect of heating method on H-mode confinement	DIII-D (ECH, NBI), PBX (NBI+RF), JET (NBI+ α), CIT (ICRF or ECRH α), ASDEX (LH, ICRF, NBI), ASDEX/u (LH, ICRF, NBI)	crucial	important
5.	Improvements to τ_E/I_p H-mode)	DIII-D, JET, JT-60, ASDEX/u, DCT, PBX, CIT, C-Mod	useful	important
6.	Effects on ion thermal transport on Ti profile	DIII-D, TFTR, JET, TEXT, PBX	important	useful
7.	Electron thermal transport at very low collisionality	DIII-D, TFTR, T-15, MTX, DCT, CIT	important	crucial
8.	DT isotope effects	TFTR, JET, CIT, T-14	useful	useful
9.	Alpha heating physics	TFTR, JET, CIT, T-14	crucial	important
10.	Fast ion transport at high beta (ballooning)	TFTR, DIII-D, CIT	important	important
10.	Single fast particle confinement (ripple loss)	TFTR, DIII-D, JET, CIT	important	important
11.	Plasma edge transport	DIII-D, PBX, C-MOD, TFTR, JET, ASDEX/u, JFT2-M, JT-60, CIT, TEXT	important	useful
12.	Particle transport (understanding of)	All present experiments	crucial	crucial
13.	Impurity transport (understanding of)	TEXTOR, ASDEX/u, TFTR DIII-D, C-MOD, CIT TORE-Supra, DCT	important	crucial

Table 6.1-2. ETR physics R&D--stability.

	Critical Issue	Facilities	Contribution to ETR Phase I	Contribution to ETR Phase II
1.	Dependence of beta limits on plasma geometry, with divertors	DIII-D, PBX, JET, C-MOD, ASDEX/u, CIT	crucial	crucial
2.	Dependence of beta limits on current profile, $q(\psi)$ shape	DIII-D, JET, ASDEX/u, TFTR, PBX, TORE-Supra, MTX, DCT	crucial	crucial
3.	Techniques for disruption stabilization at high beta and low q_a	DIII-D, JET, T-15, PBX, MTX, DCT	crucial	crucial
4.	Dependence of current limits on plasma geometry and divertors	DIII-D, PBX, JET, C-MOD, ASDEX/u, CIT, DCT	crucial	crucial
5.	Vertical stability at high elongation	DIII-D, JET, CIT, DCT	crucial	crucial
6.	Control of tearing modes	DIII-D, JET, T-15, TFTR, PBX, MTX, DCT	important	important
7.	Avoidance of kink modes	DIII-D, PBX, ASDEX/u, CIT	important	important
8.	Control of ballooning modes	DIII-D, PBX, T-15, ASDEX/u, TFTR, JET, CIT	important	important
9.	Fast ion driven modes	TFTR, DIII-D, JET, PBX, CIT	important	important
10.	Stability for $\tau \gg L/R$	TORE-Supra, DCT	important	crucial
11.	Second stability	PBX, ATF	useful	useful

Table 6.1-3. ETR physics R&D--impurity control and fueling.

Critical issue	Facilities	Contribution to ETR Phase I	Contribution to ETR Phase II
1. Minimal impurity level ($Z_{eff} < 1.5$)	TFTR, DIII-D, PBX, MTX, CIT, JET, JT-60, ASDEX, ASDEX/u, TORE-Supra, TEXTOR, T-15, T-14, DCT	important	crucial
2. Adequate lifetime for first wall components due to erosion	Test facilities at SNL and UCLA, DIII-D, PBX, JET, DCT, ASDEX, ASDEX/u, JFT2-M, CIT JT-60, TFTR, JET, TORE-Supra, TEXTOR,	useful	crucial
3. Compact poloidal divertor for impurity control	DIII-D, JET, JFT2-M (only modest elonga- tion is possible, $k \sim 1.4$), PBX, ASDEX/u C-Mod, CIT, DCT	crucial	crucial
4. High recycling operation of poloidal divertor to raise density, lower temperature	DIII-D, JET, JFT2-M, PBX, ASDEX/u, CIT, Alcator C-Mod, CIT, DCT	important	crucial
5. Helium exhaust	DIII-D, JET, JFT2-M, PBX, ASDEX/u, CIT, Alcator C-Mod, DCT	crucial	crucial
6. Impurity transport	DIII-D, JET, JFT2-M, PBX, ASDEX/u, CIT Alcator C-Mod, DCT	important	crucial
7. Sufficient pellet pene- tration and particle transport for adequate refueling	TFTR, JET, JFT2-M, PBX, ASDEX/u, DCT Alcator C-Mod, CIT,	important	crucial
8. Minimal fast alpha flux on first wall	TFTR, JET, CIT for ripple losses; PBX, DIII-D, TFTR, JET, JT-60, ASDEX, ASDEX/u, TORE-Supra, CIT, DCT for fishbone losses	important	crucial
9. Minimal damage of first wall components due to disruption heat loads	DIII-D, TFTR, JET, JFT2-M, PBX, ASDEX/u, TORE-Supra, C-Mod, CIT, DCT	important	crucial
10. Acceptable tritium retention in first wall materials	Laboratory tests of the tritium reten- tion; TFTR, JET, CIT, T-14	important	crucial
11. Impurity control with the special conditions required for current drive	JT-60, PBX, ASDEX, DCT	useful	crucial

Table 6.1-4. ETR physics R&D--heating and current drive.

	Critical issue	Facilities	Contribution to ETR Phase I	Contribution to ETR Phase II
1.	ECH Heating	DIII-D, T-15, MTX	important	included in current drive
2.	High energy NBI heating	TFTR, JET, DCT	important	included in current drive
3.	Ion Bernstein wave heating	DIII-D, PBX, C-Mod	useful	not needed unless also drives current
4.	EC current drive (steady-state)	DIII-D, MTX, COMPASS, DCT	NA*	crucial
5.	NBI current drive (steady-state)	TFTR, JET, DCT, DIII-D	NA	crucial
6.	LH current drive (steady-state)	JT-60, JET, ASDEX/u, MTX, TORE-Supra, PBX, DCT	NA	crucial
7.	Low density current ramp-up	JT-60, JET, ASDEX/u, TORE-Supra, DCT	important	important
8.	Suppressing M=1 (saw-teeth) and m=2 disruptions with current drive	JT-60, JET, ASDEX/u, MTX, TORE-Supra, DCT	crucial	crucial
9.	Amplification of current drive by the bootstrap effect	significant in any Tokamak at $\epsilon\beta_p > 1/3$	useful	important
10.	High beta optimization using current profile control	DIII-D, MTX, DCT, TORE-Supra	important	crucial
11.	Fueling and divertor heat loads with steady state current drive	DCT	important	crucial

*The ETR first phase does not call for current drive because (a) dominant alpha heating physics for long pulses at ignition is desired to be studied, and (b) more development time is then allowed for higher efficiency current drive methods. However, if more efficient current drive were developed in time (e.g., beat waves in MTX), it could still be used in the ETR First Phase.

Table 6.1-5. ETR physics R&D--operational scenarios.

	Critical issue	Facilities	Contribution to ETR Phase I	Contribution to ETR Phase II
1.	Pure inductive ramp-up, ignition, and (~ 200 sec) burn	DIII-D, TFTR, JET, JT-60, T-15, CIT, T-14	useful	NA
2.	LH-assisted ramp-up (~ 800 sec) ignited burn	JT-60, JET, TORE-Supra, DCT	important	important
3.	LH-assisted ramp-up, $Q > 20$ inductive + LH assisted burn (~ 1600 sec)	JT-60, JET, TORE-Supra, DCT	important	important
4.	Full noninductive steady state $Q > 5$, CW	TORE-Supra, DCT	NA	crucial
5.	Plasma startup and turnoff	DIII-D, TFTR, JET, JT-60, T-15, CIT, T-15, DCT	important	important
6.	H-mode τ_E control	DIII-D, JET, JT-60, PBX, CIT, DCT	important	crucial
7.	Plasma position control with strong shaping	DIII-D, JET, PBX, CIT, DCT	crucial	crucial
8.	Variable fueling control	DIII-D, JET, T-15, TFTR, JT-60, TORE-Supra, DCT	important	crucial
9.	Current profile control	DIII-D, MTX, TORE-Supra, DCT	important	crucial
10.	MHD activity monitoring and disruption prevention	DIII-D, MTX, TORE-Supra, JET, JT-60, DCT	crucial	crucial
11.	Alpha heating effects on operating point control and burn control	JET, TFTR, CIT, T-14	crucial	crucial

Table 6.1-6. Short list of tokamak experiments potentially contributing to the ETR.

	CIT	DIII-D	TFTR	MTX	C-MOD	TEXTOR	JET	ASDEX-U	TORUS SUPRA	PBX	JFT2-M	JT-60	T-14	T-15	DCT
<u>GEOMETRY</u>															
R (m)	1.75	1.67	2.60	0.64	0.64	1.75	3.00	1.65	2.25	1.65	1.30	3.00	1.25	2.43	1.67
a (m)	0.55	0.67	0.96	0.12	0.21	0.46	1.20	0.50	0.80	0.30	0.35	0.90	0.27	0.70	0.46
Elongation	2.0	2.50	1.00	1.00	1.80	1.00	1.60	1.60	1.00	2.3	1.70	1.00	1.70	1.00	2.4
Volume (m ³)	20.9	37.00	47.30	0.18	1.00	7.31	136.44	13.03	28.43	6.7	5.34	47.97	3.06	23.5	17.0
B _{tor} (T)	10.0	2.20	5.00	9.00	9.00	2.60	3.40	3.50	4.50	1.0	1.40	4.50	14.7	3.5	4.0
I _p (MA)	9.00	3.50	3.50	0.30	3.00	0.50	4.00	2.88	1.70	0.8	0.50	2.10	3.85	1.5	4.0
q(a)	3.0	4.35	3.39	3.6	2.75	3.63	5.15	2.86	4.93	1.1*	2.98	3.49	2.38	2.8	2.5
t pulse (s)	4.7	1.50	3.00	0.10	1.00	3.00	15.00	5.00	30.00	1.0	1.00	10.00	2.0	5.00	CW
Edge(L,D,P)	L,D	L,D	L,P	L	L,D	P	L,D	L,D	P	D	L,D,P	L,D	L	L	L,D
<u>POWER</u>															
P _{oh} (MW)	1.65	1.8	1.8	0.2	1.5	0.3	2.0	1.0	0.9	0.4	0.3	1.1	1.9	1.0	NA
P _{nb1} (MW)		14.0	27.0			3.0	18.0	6.0	7.0	8.0	1.6	20.0		9.0	19.0
P _{ich} (MW)	20.0	9.0	10.0		5.0	4.2	20.0	6.0	12.0	1.0	3.0	5.0			
P _{lhh} (MW)		0.0		2.0					8.0	2.0	0.6	15.0			6.0
P _{ech} (MW)		2.0		2.0									20.0	5.0	5.0
P _{other} (MW)	60 ()										0.4		40 ()		
P _{total} (MW)	61.7**	26.8	38.8	4.2	6.5	7.5	40.0	13.0	27.9	11.4	5.9	41.1	1.8**	15.0	30.0
<u>PLASMA PROPERTIES</u>															
n _{max} (10 ²⁰)	9.47	2.48	1.21	6.63	21.65	8.75	0.88	2.55	0.85	2.8	1.30	0.83	16.8	0.97	6.0
n _{oper} (10 ²⁰)	3.4	1.86	0.91	1.40	4.00	0.56	0.66	1.91	0.63	2.1	0.97	0.62	12.6	0.73	1.11
H-mode mult.	1.2	2.00	1.00	1.00	2.00	1.00	2.00	2.00	1.00	2.0	2.00	2.00	1.00	1.00	2.0
H x τ_{aux} (s)	0.57	0.309	0.131	0.007	0.094	0.021	0.376	0.223	0.052	0.11	0.048	0.162	0.22	0.082	0.41
beta (%)	4.8	11.54	1.07	0.50	1.89	0.81	2.39	4.55	0.62	46.8	6.67	1.71	3.8	0.98	6.1
n(Te+Ti)(keV-10 ⁻²⁰)	66.0	9.38	4.51	7.0	25.69	0.92	4.63	9.36	2.13	7.8	2.20	5.81	126	2.0	16.2
Ti (keV)	9.7	2.52	2.49	2.5	3.21	0.81	3.49	2.45	1.68	1.86	1.13	4.69	5.0**	1.4	7.4

* Actual MHD q is higher due to bean shape

** Only ohmic and alpha power is included for τ_E in CIT and T-14, as far as this formula is used. T-14 has additional heating due to compression

Calculations of plasma parameters for Table 6.1-6

UNITS

All quantities are given in the following units in the tables and formulas:

Length - meters
Magnetic Field - tesla
Plasma Current - MA
Power - MW
Energy - MJ
 τ_E - seconds
 n - 10^{20} m^{-3}
 T - keV

FORMULAS

Plasma volume

$$V = 2\pi^2 R K a^2 = 19.74 R K a^2$$

Plasma q at edge (approximate, with correction for finite aspect ratio)

$$q = 2.5 \frac{B_t a^2}{I_p R} \frac{(1 + K^2)}{[1 - (a/R)^2]^2}$$

Ohmic heating power

$$P_{oh} = 0.5 I_p$$

(assumes 0.5 V loop voltage during auxiliary heating)

Plasma confinement time (Kaye-Goldston scaling)²

$$\tau_E = [(1/\tau_{ohmic})^2 + (1/\tau_{aux})^2]^{-1/2},$$

where

$$\tau_{ohmic} = 0.07 n R^2 a q^*$$

$$\tau_{aux} = 0.056 I_p^{1.24} R^{1.65} K^{0.28} n_{oper}^{0.26} \left(\frac{A_1 A_2}{2}\right)^{1/4} / (p_{total}^{0.58} a^{0.49} B_t^{0.09})$$

$$q^* = 2.5 a^2 B_t (1 + K^2) / (I_p R)$$

are the equations for neo-Alcator ohmic scaling, L-mode auxiliary heating scaling and "circular" q equivalent, respectively.

Plasma pressure

Using $P_{te}/V = 3/2 n (T_e + T_i)$, and converting to units given above

(n in 10^{20} m^{-3} , T in keV) gives

$$n(T_e + T_i) = 41.7 P_{total} \tau_E / V$$

Plasma beta (%)

$$\beta = \frac{P_{total} \tau_E}{V B_t^2 / 2 \mu_0} = 2.5 \frac{P_{total} \tau_E}{V B_t^2} \times 100$$

Maximum density attainable (Greenwald scaling, for auxiliary heated plasma)

$$n_{max} = I_p / \pi a^2$$

Maximum normal operating density

$$n_{oper} = 0.75 n_{max} \text{ or smaller, if higher } I \text{ is desired.}$$

and current profile control and impurity control related to these steady-state operational phases of TIBER and ITER.

Together, this R&D needs assessment addresses issues in three of the six major research areas described in the Technical Program Activity (see Technical Program Activity Final Report, ANL/FPP-87-1, January 1987): Confinement Systems, Physics of Burning Plasmas, and Plasma Technology. Of course, the TIBER/ETR and ITER design studies themselves address another TPA research area, Systems Design and Analysis. This R&D section does not address the other TPA areas of Nuclear Technology and Fusion Materials, but these will be addressed for TIBER in a separate Volume II, to be published by Argonne National Lab. Many of the critical physics issues discussed here are common to those identified in the TPA report. What we have tried to emphasize here are the key driving issues for the TIBER (and anticipated ITER) designs. We have tried to connect the identified key issues with specific recommended machines suitable (but not necessarily yet committed) to addressing those issues. We have also tried to consider all Tokamak facilities, including those abroad, which would be suitable since TIBER and ITER are intended to be built and operated internationally. The next step for future work would be to take these identified R&D needs and work out an appropriate experimental implementation plan. Presumably, such a supporting implementation plan will be done shortly as part of the international ITER program. There may be an attempt to coordinate the carrying out of such an international ETR R&D plan, so that there could be some specialization by some countries on portions of the needed R&D. We hope this R&D needs assessment will provide a useful input from the US perspective in any such international ETR R&D planning effort. Also, in case some specialization for each country is called for, we recommend the US emphasize, for its major contributions to ETR physics R&D, four physics performance-driving elements which dominate many of the critical issues identified in Tables 6.1-1 to 6.1-5:

- (1) Strongly shaped plasmas with divertors for impurity control.
- (2) Steady-state current drive.
- (3) Advanced fueling (higher velocity pellet or plasmoid injection).
- (4) Alpha physics.

Many existing US Tokamaks could usefully address parts of these four primary elements, as well as many other generic ETR issues in confinement and beta. However, a relatively complete coverage of these four driving elements would need the combination of CIT and DCT in the 1990's, as called for in the TPA.

6.1.1 CONFINEMENT

For clarity, in discussing this area, it is convenient to divide the plasma into three regions:

- Center $q < 1$.
- Confinement zone $1 < q < q_a$.
- Edge $q \approx q_a$.

It seems that all future machine designs have identified H-mode confinement quality as a necessary ingredient for device success. Accordingly, many sections of Table 6.1-1 deal with issues specific to the H-mode. Good H-mode confinement is most important for any initial ignited (or very high Q) phase of an ETR [equivalent to at least 1.7-2.0 times τ_E (Kaye-Goldston)]. For the current drive mode at $Q \sim 5$ to 10 in the steady-state phase of the ETR, the H-mode requirement is less, equivalent to 1.2 to 1.5 times τ_E (Kaye-Goldston).

6.1.1.1 Sawtooth Transport

In the central region, confinement is generally dominated by sawtooth activity. It is important to understand the characteristics of sawteeth (amplitude and frequency), not only because they contribute to heat and particle losses, but also because they have a strong influence on profiles, and on fusion power. It is observed that their characteristics change in low resistivity plasma conditions, and it is important to validate theories of this region. Continued studies of sawtooth suppression, using current drive, are important. Sawtooth amplitudes in the presence of substantial alpha heating should be encountered and measured in CIT, T-14, TFTR and JET prior to ETR operation. The role of sawtooth activity on central ion particle confinement may also be important, particularly to aid alpha ash transport from the fusion source in the central region into the confinement zone nearer to the edge.

6.1.1.2 Transport Theory

While empirical scalings are used for convenience in discussing tokamak performance, in fact, there is a deeper understanding of the underlying transport mechanisms.

Under a variety of conditions the ion thermal diffusivity is in agreement with the predictions of the Chang-Hinton neo-classical theory. There is mounting evidence that deviations from neo-classical behavior are a result of ion pressure gradient (η_i) modes. An issue for ETR is whether a judicious choice of fueling, and current drive can create peaked density profiles without sawteeth (as in TFTR supershots) to maintain a low level of η_i transport.

Various theoretical models have been proposed to explain the anomalously high electron transport, and both electrostatic and electromagnetic instabilities have been invoked to explain the observations. Dissipative Trapped Electron (DTE) modes exhibit some of the features of the observed ohmic and L-mode scalings. In some cases measured density fluctuations have the form and magnitude predicted by such electrostatic theories. However, mode behavior is not correlated with the L to H transition, and like many other theories, which have a zero or negative dependence on atomic mass, there is no explanation of the common empirical observation that global confinement improves with atomic mass. A thermal diffusivity resulting from electrostatic or magnetic turbulence may be the cause of the enhanced losses. Validation of this latter possibility awaits the development of a method for measuring fine-scale magnetic fluctuations in the confinement zone.

There needs to be an emphasis on studies to understand the origin of the H-mode, since all ETR designs would appear to need some degree of H-mode confinement, particularly for ignition, and to a lesser degree for a steady-state driven burn. Such studies should define the basis for expecting H-mode to actually be achieved in ETR, a basis better than simple extrapolation. Basically all of the divertor machines that have achieved or can be expected to achieve H-mode will contribute to this physics endeavor. Even a machine like JT-60 which has failed to achieve the H-mode contributes important data to understanding the physics of the H-mode. The research program in this area consists largely of making increasingly detailed edge plasma measurements in the various devices listed. The real challenge in this area lies in the

plasma theory, however. The number of detailed observations on H-mode exceeds the number of theories by quite a large factor at present.

6.1.1.3 Global Empirical Scaling Studies

Global empirical confinement scalings are commonly used to describe the confinement in tokamaks. In studies for TIBER and the CIT, a combination of ohmic, Iteo-Alcator scaling, auxiliary heated (such as Kaye-Goldston scalings) has been used. Refinement of present uncertainties will require a larger database than presently available:

- (1) Validation of the empirical scalings, notably H-mode, under ITER plasma conditions (R , R/a , b/a , q , ...) near operational limits;
- (2) Testing of alpha effects on transport - should alpha power be treated like other power in empirical scalings?;
- (3) Conditions for the sustainment of the H-mode; and
- (4) Confinement scaling with current drive.

In particular, the scaling of the H-mode with plasma current is one of the most important factors upon which future ETR's rely. DIII-D, JET and JT-60 all have sufficient high current to demonstrate this current scaling in a regime close to the ETR applications.

6.1.1.4 H-Mode with Different Heating

There needs to be experimental verification that the H-mode can be produced and sustained with the types of heating that are candidates to be used on the ETR. The same difficulty applies to CIT. There needs to be a research program to demonstrate that H-mode can be compatible with ICRF alone for CIT, or with alternative heating such as ECH. For ETR the prime candidate heating systems are combinations of lower hybrid with ECH or negative ion beams for their combined heating and current drive application. At some point the properties of H-mode plasmas with alpha heating will need to be studied. This data appears likely to come from JET and from CIT. The start date for DT operation in JET is listed as 1992 and 1995 for CIT.

6.1.1.5 Improvements to τ_E/I_p (H-mode)

The quality of confinement available with H-mode, while good, should not be regarded as the best that can be accomplished in the Tokamak. An active program of further improving the confinement following along the H-mode path is very important for a machine like ETR. If the current drive efficiency turns out to be not too high, then ETR will desire to run at lower current in order to achieve a substantial capital Q, but with current scaling of confinement there will then be a penalty in τ_E . This coefficient τ_E/I_p is referred to here as the confinement quality factor. For steady-state devices running with modest Q in the range of 10, the capital Q can be shown to be proportional to this confinement quality factor times the current drive efficiency. DIII-D has an active program in this area of confinement improvement. We expect that JET will turn to such methods of improving confinement in the 1990 time frame. Both ASDEX/upgrade and an upgrade of JT-60 can be expected to explore improvements on H-mode confinement. Finally, DCT in the mid 1990's would provide a needed testbed for H-mode optimization in steady-state with current profile control.

6.1.1.6 Ion Thermal Transport

The key ingredient in understanding thermal ion transport is an ion temperature profile measurement. Although such measurements are difficult, DIII-D has an excellent capability in this area and will likely make major contributions to the study of thermal ion transport. When the TFTR charge exchange recombination system begins reliable operation, TFTR will also be able to make major contributions in this area, although not with H-mode plasmas.

6.1.1.7 Electron Thermal Transport

The steady-state phase of an ETR will require operating at low density and very high temperature, which will place the collisionality deep into the trapped particle regime. It is a hotly debated theoretical issue as to whether mechanisms like the collisionless trapped particle modes which give χ_e proportional to $T_e^{3/2}$ will in fact be found in these collisionality regimes. The key factor in performing these experiments is the ability to achieve high electron temperatures and to vary the electron temperature with all the other parameters remaining constant. With strong ECH heating and neutral beam

injection heating in the 1990 time frame, DIII-D should be able to make such studies in an optimal manner. MTX, with intense FEL heating in the burst mode, may also be able to achieve high $T_e > 10$ keV in the early 1990 time frame. TFTR already operates plasmas of low collisionality but needs a means independent of neutral beam injection to vary the electron temperature. The ICRF heating coming on to TFTR in 1989 will contribute substantial electron heating at low density. The alpha particle heating will also predominately heat electrons in CIT and T-14.

6.1.1.8 DT Isotope Effects

With H-mode operation, the ASDEX team has found that τ_E is linearly proportional to the ion mass. This strong dependence favors DT operation over what we see in present day hydrogen and deuterium experiments. The first machines to test these isotope effects with the reactor relevant ion species will be TFTR, JET, CIT, and T-14.

6.1.1.9 Alpha Heating

It remains a very important question whether alpha heating will show the same confinement degradation as auxiliary heating methods. A first look at alpha heating in tokamak plasmas will be obtained in TFTR in the DT phase in 1990. The basic mission of JET and T-14 is to demonstrate significant alpha heating of the plasma and, of course, CIT with ignition will be the milestone in this particular area.

6.1.1.10 High Beta Fast Ion Transport

Theoretically, it has been suggested that centrally peaked pressure distributions of superthermal ions can drive ballooning modes. These ballooning modes in the interior can then cause fast ion transport which will broaden the heating profile and in an extreme case might lead to fast ion loss. In the reactor situation, of course, the alpha heating is strongly peaked near the center of the plasma and, therefore, such modes represent a particular hazard to alpha heating. A reacting plasma however, is not necessary to study this physical process. Any tokamak which can achieve central beta values up to the beta limit for that device and which also has a source of significant superthermal particles should be able to study this particular phenomenon. TFTR has obtained preliminary evidence for such fast

ion transport. TFTR can do the most elegant experiment in this field by means of tritium beam injection into a deuterium target plasma. However, this experiment is somewhat tangential to the main line DT, $Q = 1$ effort. In TFTR's limited 40 week DT phase in 1990, there is some doubt whether experiments of this nature can be fit in. DIII-D has requisite high beta and, with strong centrally peaked neutral beam injection, can study the effects of such interior ballooning modes on the neutral beam injected fast ions.

6.1.1.11 Single Fast Particle Confinement

Ripple losses are not generally considered to be a problem at the ETR level of ripple, however, concern has arisen recently about the possibility of severe losses of trapped, energetic, alpha particles under these conditions. Recently theoretical work has suggested ripple loss, for example, might be exceedingly detrimental to alpha particle confinement. Because of the DT operation in TFTR and the excellent condition there of fusion reaction product studies we can expect TFTR to make major contributions in this area. Other tokamaks which have high current and the ability to produce fusion reaction products at least in test quantities can also contribute in this area. DIII-D, JET and T-14 can contribute in this area, and of course CIT and T-14 will be most relevant.

6.1.1.12 Edge Transport

Most tokamak data indicates that transport in the edge plasma is caused by different instabilities than those in the central confinement region. Candidate modes include resistivity gradient driven (rippling) modes and micro-tearing modes. Because the edge conditions set the boundary conditions for the central confinement region, they can have a large impact on global confinement - the L-H mode transition is a clear example of this. Numerous explanations have been given for the transition, and the effect of trapped ions and ballooning modes are both possible contributors to the phenomenon.

6.1.1.13 Particle Transport

The particle diffusivity is also anomalous, and typically $D \sim \chi_e$. The understanding of particle transport is complicated by the presence of a strong flow term, which, depending on plasma conditions, can lead to a net outward or inward flow of particles. Such a flow is predicted by both neo-classical and

electrostatic turbulence theory. It is of major importance to the understanding of fueling needs. This may be most critical for the steady-state ETR phase, both because the long term wall recycling may be different in steady-state, and because the lack of toroidal electric field with current drive may change or eliminate the inward flow of particles.

6.1.1.14 Impurity Transport

It is critically important to obtain a better understanding of impurity transport for, while effects attributable to neo-classical phenomena have been seen, the transport is often anomalous. It is essential to obtain a better understanding of the conditions for impurity accumulation, which often occurs under conditions of optimum (H-mode) plasma confinement, and development of techniques to avoid accumulation. In addition, most experiments have operated on a time scale short compared to plasma-wall equilibration, and ultra-long pulse, high power density, diverted/pumped limiter operation is needed to prepare for the ETR.

6.1.2 STABILITY

A clearer understanding is required of the limitations on disruption free, high beta operation set by MHD instabilities. The Troyon formula $\langle \beta \rangle = CI/aB$ is a first simple guide to such limits. However, further calculations are required to delineate how the factor C (presently taken to be a constant) depends upon R/a , q_a , κ , δ , ι and on the shape of the current and pressure profile. The current profile with steady-state current drive will likely differ from ohmic profiles in ignited operation, and in fact may need to be optimized with current drive spatial control to achieve maximum C at minimum of q_a simultaneously, and avoid disruptions.

6.1.2.1 Dependence of Beta Limits on Geometry and Divertors

The key issue here is to define the optimal plasma shape within the context of "normal" plasma shapes in tokamaks. For ETR, elongations should be investigated up to 2.5. DIII-D has explicit plans to investigate this regime and is considering whether discharges with elongation three can be operated. If one counts elongation measured around the curvature of the bean, PBX contributes high elongation data. Triangularity and indentation are closely related factors in optimizing the plasma shape for high beta. Theoretically

the crescent shape is expected to have the highest beta although the magnitude of the advantage predicted does not seem compelling in regard to machine design. Both DIII-D and PBX should contribute in this area of high triangularity and indentation. There remains an explicit issue of beta limits in limiter vs divertor plasmas. In Doublet III the highest achieved beta was in the limiter case, owing to the ability to operate higher current per unit field in limiter vs divertor. Whether there is any other explicit dependence of the beta limits on details of the flux surface topology remains to be seen. Experimental results that will contribute in this area should be obtained from DIII-D, PBX, ASDEX/upgrade and CIT. All of these devices should have sufficient power to get to the beta limit at the toroidal fields they expect to operate.

6.1.2.2 Dependence of Beta Limits on Current Profile Shape

Theoretical studies have shown beta limits can vary as much as a factor of two for a given geometry, corresponding to Troyon coefficients $C = 0.02$ to 0.04 , depending on the current profile shape ($j(\phi)$ or $q(\phi)$) and the pressure profile. Both the ignition and current drive ($Q > 5$) phases of ETR operation depend critically on achieving some degree of current profile optimization to reach required betas ($C \gtrsim 0.03$ to 0.035), and would greatly benefit from fully optimized profiles ($C \gtrsim 0.04$) using some form of current drive. In the 1990's, current drive with either lower hybrid, neutral beams or ECH will be available in DIII-D, JET, ASDEX/upgrade, TFTR, PBX, TORE-SUPRA, MTX and DCT, to explore higher betas through current profile control.

6.1.2.3 Techniques for Disruption Stabilization

The common present experimental approach to disruption avoidance is to operate at betas well below the Troyon limit and at safety factors q_a well above three. However, this will not suffice for either ignited or current-driven ETR operation, which must operate at $\beta \gtrsim 0.03$ I/aB and at $q_p \lesssim 3$ to reach acceptable performance. Thus, some means of active disruption stabilization is needed. Some form of localized heating (with inductive E) or localized current drive near the $q = 2$ surface is needed, to reduce the steepness of the current profile locally near $q = 2$. Experiments with localized ECH, or lower hybrid edge current drive, such as DIII-D, JET, T-15, PBX, MTX and DCT, should be able to address this need. (CIT will leave ICRF

and possibly ECRH for localized heating, as well as pellets and current ramp-up to freeze in favorable profiles.)

6.1.2.4 Dependence of Current Limits on Geometry and Divertors

Since ETR performance is sensitive to the upper limit on operating current or equivalently the lower limit on q_a , we need more database on those limits from machines that have the needed strong plasma shaping and divertors: DIII-D, PBX, JET, C-MOD, ASDEX/upgrade and CIT. The maximum performance and maximum current limit would need careful current profile control, a technique expected to be first tested in DIII-D, JET and PBX, and then further refined in DCT.

6.1.2.5 Vertical Stability

Vertical stability is a specific issue when we are talking about elongations as high as 2.5. DIII-D will first address this issue experimentally in 1988. Control of the highly shaped PBX plasma will also contribute to the understanding of the control vertical instabilities. However, since vertical stability is very intimately connected with the machine design and the poloidal coil system layout the real answer to this question should be derived from computation and modeling work. JET, CIT, ASDEX/upgrade, C-MOD and DCT can provide more data later on these large elongations.

6.1.2.6 Control of Tearing Modes

Tearing modes are the dominate MHD mode that all Tokamaks must deal with. Tokamaks that can have active control of these modes by localized heating and current drive are DIII-D, JET, T-15, PBX, MTX and DCT. An effect of the bootstrap current, first observed on TFTR, may also be to modify the stability of bearing modes.

6.1.2.7 Avoidance of Kink Modes

Wall stabilization is often discussed for stabilizing plasmas against external kink modes, however, this may be limited to the skin time of the wall. PBX is a key experiment to address this because of recently added close conducting walls. DIII-D and ASDEX/upgrade will help define kink mode limits accessible to a Tokamak without effective wall stabilization.

6.1.2.8 Control of Ballooning Modes

Both avoidance and control of ballooning are possible. The control aspect derives from the fact the ballooning modes are related to the pressure profile. Localized heating methods to alter the pressure profile can be effective in controlling ballooning modes. As we mentioned in connection with the fast ion transport by ballooning modes, we are mainly concerned here with the plasma interior. Consequently the high beta experiments DIII-D, PBX, MTX and ASDEX/upgrade, and TFTR and JET should contribute in this area of ballooning mode studies.

6.1.2.9 Fast Ion Driven Modes

This topic is discussed in Sec. 6.1.1.10, above.

6.1.2.10 Stability for $\tau \gg L/R$

The achievement of high fluence with high availability for nuclear testing in the steady-state phase of ETR operation will require demonstrating stability for times much longer than the resistive skin time. This is a critical area in which some new device on the way to the ETR is called for, such as DCT. With replacement of resistive PF coils to superconducting PF coils, TORE-SUPRA may address this, but not with relevant elongated, diverted plasmas.

6.1.2.11 Second Stability

For given current drive efficiency in terms of amps per watt, the steady-state operation of the ETR could have a higher $Q = P_{\text{fusion}}/P_{\text{current drive}}$ if the ETR device could run with relatively lower plasma current. The confinement problem that may result is discussed elsewhere. Along with the confinement problem however, is the fact the toroidal beta limit which is proportional to I/aB also decreases as the current is decreased. Hence operation in a second stable region and confinement improvement are two related elements that must be tied together to be useful for steady-state ETR, if the current drive efficiency turns out to be modest. The PBX tokamak and the ATF tokamak are at present the only experiments with the announced goal of studying second stability, and testing should occur within the next two years. Second stability refers to ballooning modes. It is not at all clear that there exists a second stable regime to external kink

modes. If ETR is designed to run within the constraints of the Troyon beta limit for the first stable region, then the emphasis on the device design will be to operate at the highest possible plasma current. The highest beta will be achieved based on the highest available operating plasma current. The key issue then becomes not the beta limit but the current limit.

6.1.3 IMPURITY CONTROL/FUELING

Impurity control and fueling are critical issues for an ETR due to the requirements for reliable long pulse (> 200 s) and steady-state operation. The major issues are:

- (1) maintaining a sufficiently low impurity level so that central radiation losses are not a problem and that dilution of the fuel by impurities does not impact the plasma beta,
- (2) maintaining a sufficiently low net erosion rate of the first wall components so that the lifetime of the first wall components is long enough to allow successful operation of the tokamak,
- (3) exhaust of the helium created by the fusion reaction so that the plasma can be run for the required pulse length, and,
- (4) adequate fueling of the plasma with deuterium and tritium so that long pulse operation can be sustained.

The reference impurity control system relies upon the use of a compact poloidal divertor. The physics issues associated with the successful operation of a compact divertor include the physics of particle and energy transport in the central plasma, the transport of energy and particles along and across the flux surfaces in the edge plasma, the transport of neutral atoms and molecules, the effects of atomic and molecular collisions, the effects of collisions of the plasma ions and neutral atoms with the wall including reflection, sputtering, and implantation, the transport of ions and atoms within the wall materials, the transport and atomic physics of impurities, and the physics of the sheaths that are formed where the plasma contacts the wall. The reference method for impurity control is to operate the divertor in a high recycling regime in which intense localized recycling produces a cool, dense plasma at the divertor plate, thus providing a low temperature to minimize the sputtering and a high neutral density to allow

successful pumping of the helium. The key physics R&D items involve verifying the utility and feasibility of high recycling operation of divertors and characterizing the physics of divertors. The most appropriate experiments to resolve these questions are long pulse divertor experiments with high levels of auxiliary heating.

There are a number of auxiliary questions concerning impurity control including the choice of appropriate materials, discharge cleaning techniques, and impurity control with current drive at low densities.

Fueling is also a major issue. While gas fueling is successful on present experiments, central fueling with pellets has demonstrable advantages for profile control, and improved fueling efficiency. Thus the development of a high speed pellet injector or other central fueling technique would be important for fueling an ETR.

6.1.3.1 Minimal Impurity Level ($Z_{\text{eff}} < 1.5$)

Successful operation of an ETR will require minimal impurity radiation with maximum impurity levels of less than 1-2% of low Z impurities (C, O), less than .1-.2% of medium Z impurities (Fe, Ni, Ti, etc.), less than .01-.02% of high Z impurities (W, Mo, Ni, etc.), and less than 5% of He. These restrictions are due to avoiding excessive radiation levels and avoiding excessive depletion due to impurities of the allowed plasma pressure. Thus R&D needed most is the development of successful techniques for tokamak operation with low impurity content and high heating power.

The facilities and timescales for developing these techniques are:

U.S.

TFTR (1983-1990) - demonstration of impurity control feasibility of graphite limiters with high heating power (30 MW NBI, 5 MW ICRF).

Study of effects of impurity level on fusion yield (DD and DT).

DIII-D (1986-1991) - demonstration of impurity control capability of graphite tiles and poloidal divertors with high heating power (10 MW NBI, 2 MW ECRH).

PBX (1987-1991) - study of impurity control capability of graphite tiles and poloidal divertors with high heating power (7-8 MW NBI).

MTX (1989-) - study of impurity control capability of limiters with high power ECRH in a dense plasma.

C-MOD (1991-) - study of divertors at high power density with ICRF heating.

CIT (1993-1998) - study of impurity control capability of divertors with very high heating power (20 MW ICRF, 60 MW of alpha heating), study of effects of impurity level on fusion power.

DCT (1995?-) - evaluation of steady-state impurity control with divertors and 20 MW of NBI, ECH and LH power.

Foreign

JET (1984-1994) - comparison of impurity control with divertors and limiters and high power heating (10 MW NBI, 15 MW ICRF?). Study of effects of impurity level on fusion yield (DD and DT).

JT-60 (1986-1993?) - evaluation of impurity control with divertors and limiters with high power heating (10 MW NBI, 20 MW LH, 10 MW ICRF).

ASDEX (1980-1989) - continued evaluation of impurity control issues with divertor and high power heating (3 MW NBI, 3 MW ICRF, and 3 MW LH).

ASDEX/upgrade (1988-1995) - evaluation of impurity control with divertors and limiters with high power heating (6 MW ICRF, 6 MW NBI).

TORÉ-Supra (1988-1995) - evaluation of impurity control with limiters and high power heating, emphasis on long pulse length issues (30 s).

TEXTOR (1984-1990) - evaluation of impurity control with limiters and modest levels of ICRF heating (2 MW).

T-15 (1989?) - evaluation of impurity control with limiters and high heating power (14 MW)

T-14 (1989?-) - evaluation of impurity control with limiters and alpha heating (~ 30 MW P_α).

6.1.3.2 Adequate Lifetime for First Wall Components due to Erosion

The ETR first wall components must have a reasonable lifetime (greater than one year) due to erosion by the plasma. The major source of erosion is likely to be sputtering by ions and fast neutrals. The loss of surface material by sputtering is likely to be ameliorated by re-deposition of the

sputtered material on the first wall components. The maximum net erosion rate that will probably be acceptable is 1 cm/year. The R&D needed is therefore to demonstrate tokamak operation with conditions that will likely lead to low net erosion rates. Study and characterization of erosion and redeposition phenomena in tokamak experiments and test facilities.

Test facilities at SNL and UCLA can explore some of these issues. Plasma characteristics needed for low erosion can be determined from modeling studies. Achievable plasma characteristics should be determined from measurements on divertor experiments (DIII-D, PBX, JET, ASDEX, ASDEX/upgrade, JFT2-M, JT-60 and CIT) and limiter experiments (TFTR, JET, TORE-Supra, TEXTOR and T-15). Actual data on erosion lifetimes would require a high power, steady-state experiment such as DCT.

6.1.3.3 Compact Poloidal Divertor for Impurity Control, Enhanced Confinement

The present ETR designs incorporate a compact poloidal divertor for impurity control and confinement enhancement. The design assumptions made and the parameters chosen need to be consistent with the performance of divertors in operating tokamaks. The particular issues are: the minimum distance allowable from the null point to the divertor plate, the potential need for mechanical and/or plasma baffling of the neutral backflow, the dependence of the divertor operating parameters on the magnetic configuration and other operating parameters (elongation, single null or double null configuration, plasma density, impurity level and radiation losses, heating power, etc.), the potential need for local pumping, variation of radial decay lengths for power and density, role of impurity and hydrogen radiation in the divertor power balance, compatibility of RF (particularly ICRF) heating with divertor operation, compatibility of divertor operation with various current drive scenarios, and operational limits for divertors in terms of maximum current, maximum beta, and maximum plasma density. Experiments are needed that will address all of these issues. Ideally, those experiments should be capable of operating with significant elongation ($k \sim 1.6-2.2$) and compact divertor configurations.

The experiments capable of providing the needed data are DIII-D, JET, JFT2-M (only modest elongation is possible, $k \sim 1.4$), PBX, ASDEX/upgrade, C-Mod and CIT. They should have extensive diagnostics of the divertor plasma, and have sufficient flexibility to explore the limits for successful divertor

operation in terms of magnetic configuration, current, etc. DIII-D, JET, JFT2-M and PBX will be operating in the next few years (1987-89). ASDEX/upgrade and Alcator C-Mod will produce useful data in the 1990-93 time frame and CIT will provide data in the 1993-1997 time period. DCT could test the steady-state performance of divertors in the 1995-1998 time period.

6.1.3.4 High Recycling Operation of Poloidal Divertor to Raise Density, Lower Temperature

The present ETR strategy for impurity control is the use of a high recycling poloidal divertor. This mode of operation produces a cool, dense plasma at the divertor plate, thus reducing or eliminating sputtering as a concern for impurity production and erosion by reducing the plasma temperature and the sheath potential at the divertor plate, and reducing the pumping requirements for helium by increasing the neutral density near the divertor. Such operation has been observed on many divertor experiments. However, recent results on DIII-D and JET indicate that this type of operation may not occur during "H-mode" operation. In addition, the presence of H_{α} oscillations during "H-mode" operation significantly changes the characteristics of the diverted plasma. If the cool, dense plasma required for low sputtering in the divertor chamber does not exist during "H-mode" operation, then a different impurity control strategy will have to be developed for the ETR. Well diagnosed experiments are needed with divertor tokamaks to explore the characteristics of diverted plasmas during "H-mode" operation and in the presence of H_{α} oscillations and high power heating.

The experiments capable of providing the needed data are DIII-D, JET, JFT2-M, PBX, ASDEX/upgrade, Alcator C-Mod and CIT. Resolution of the questions will require extensive diagnostics for the divertor plasma. At present, the best divertor diagnostics are installed or planned for DIII-D, ASDEX/upgrade and Alcator C-Mod. DIII-D, JET, JFT2-M and PBX will be operating in the next few years (1987-89). ASDEX/upgrade and Alcator C-Mod will produce useful data in the 1990-93 time frame and CIT will provide data in the 1993-97 time period. DCT could provide data in steady-state in the 1995-1998 time period.

6.1.3.5 Helium Exhaust

Helium pumping must be possible with a modest sized (~ 35 – 100 kliter/s) pumping system for the ETR. This requires that the neutral gas density be in the 10^{-3} to 10^{-2} Torr range to achieve the gas throughput necessary to exhaust the helium at the rate it is created and to maintain $n_{\alpha}/n_e < 5\%$ in the main plasma. Also, $n_{\text{helium}}/n_{\text{hydrogen}}$ in the exhaust gas needs to be in the 5% range. Measurements of the helium concentration in the plasma center, in the plasma edge and in the divertor are needed in operating tokamaks. The helium could be introduced as an impurity in the 1–10% range. The helium concentration in the gas and the gas pressure in the divertor chamber should also be measured. Part of the gas in the divertor chamber will be very energetic and non-isotropic, and measurements of the distribution function of the unthermalized, hot gas from the divertor recycling region would potentially allow a reduction of the pumping speed requirements because the hot gas would have a higher throughput than thermalized, cold gas.

The experiments capable of providing the needed data are DIII-D, JET, JFT2-M, PBX, ASDEX/upgrade, C-Mod, CIT and DCT. Resolution of the questions will require diagnostics for measuring the central and edge helium and hydrogen ion density, such as charge exchange recombination spectroscopy and diagnostics for measuring the helium concentration of the diverted plasma and the recycling gas in the divertor chamber. In addition, measurements of the velocity distribution of the recycling gas would allow the ITER design team to make better estimates of the pumping speed required for adequate helium exhaust. At present, the best diagnosed divertor is on DIII-D, and good diagnostics are planned for ASDEX/upgrade and Alcator C-Mod. DIII-D, JET, JFT2-M and PBX will be operating in the next few years (1987–89). ASDEX/upgrade and Alcator C-Mod could produce useful data in the 1990–93 time frame and CIT could provide data in the 1993–1997 time period. DCT could provide steady-state data in the 1995–1998 time period. Such measurements are not a high priority item for existing machines, so encouragement is needed for these machines to address this issue.

6.1.3.6 Impurity Transport

Impurity transport in diverted plasmas, particularly during "H-mode" operation, needs to be such as to avoid central peaking of the impurity concentration for impurities generated at the plasma edge and wall. Such

central peaking is often observed with divertor experiments. In addition, there needs to be sufficiently rapid impurity transport for the central He ions to ensure the helium generated at the plasma center is removed at a large enough rate to ensure that $n_{\alpha}/n_e \sim < 5\%$ at the center. Experiments are needed which can characterize the behavior of impurities in divertor experiments, particularly with "H-mode" operation, and in very long pulse or steady-state.

The experiments capable of providing the needed data are DIII-D, JET, JFT2-M, PBX, ASDEX/upgrade, Alcator C-Mod and CIT. Resolution of the questions will require diagnostics for measuring the impurity content of the plasma. One of the best diagnostics for this purpose is absolutely calibrated charge exchange recombination spectroscopy with a doping beam. At present, the best diagnostics for this purpose are installed or planned for DIII-D, JET, PBX, ASDEX/upgrade and CIT. DIII-D, JET and PBX will be operating in the next few years (1987-89). ASDEX/upgrade will produce useful data in the 1990-93 time frame and CIT will provide data in the 1993-1997 time period. Steady-state H-mode impurity behavior could be tested in DCT in the 1995-1998 time period.

6.1.3.7 Sufficient Pellet Penetration and Particle Transport for Adequate Refueling

The ETR will need to be fueled during the pulse for long pulse and/or steady-state burn. The reference fueling method is pellet injection. The pellet velocity required for penetrating to the plasma center is on the order of 100's of km/s, which is beyond the limits of present solid pellet injectors. Thus fueling the plasma center will require transport of the fuel ions from near the edge to the plasma center by some plasma transport processes. If such processes do not work under steady-state ETR conditions, other techniques need to be developed, such as advanced plasmoid fuel injectors using lasers or plasma accelerators.[1,2] The penetration physics for pellets and plasmoids and the particle transport processes involved in fueling the plasma center need to be determined by experiments with high power plasma heating and high speed pellet or plasmoid injection. The understanding and modeling of pellet and plasmoid penetration need to be improved through continued analysis of fueling experiments, including possible effects due to energetic ions and alpha particles. In particular, the pellet and plasmoid

velocity and penetration requirements for fueling ITER and the particle transport physics of fueling for large, high power tokamaks need to be better understood.

Pellet and plasmoid injection experiments are needed on large, high power experiments. Pellet experiments capable of providing the needed data are either in operation or planned for TFTR, JET, JFT2-M, PBX, ASDEX/upgrade, Alcator C-Mod, and CIT. DIII-D could also provide data but no pellet experiments are funded for DIII-D. TFTR, DIII-D, JET, JFT2-M and PBX will be operating in the next few years (1987-89). ASDEX/upgrade and Alcator C-Mod will produce useful data in the 1990-93 time frame and CIT will provide data in the 1993-97 time frame. Plasmoid experiments capable of providing the needed data could be attempted on TFTR, JET, ASDEX/upgrade or DIII-D. Demonstration of initial plasmoid injection in these present experiments could then be further evaluated in CIT in 1993-1997 and in DCT in 1995-1998.

6.1.3.8 Minimal Fast Alpha Flux on First Wall

A significant fast alpha flux on the first wall can lead to significant first wall erosion and possibly first wall damage. The source of such alpha losses could be due to toroidal field ripple and other departures from axisymmetry, or instabilities such as fishbones. The possible fast alpha losses and their impact distribution on the first wall need to be determined by modeling studies and confirmed with experiments operating tokamaks.

All tokamaks with high power heating could potentially contribute to this issue. The most likely candidates are TFTR and JET in the short term and CIT and T-14 in the long term since they will have significant alpha particle populations. Characterization of fishbone and other MHD effects should be possible on any tokamak that will operate near the beta limit, i.e., PBX, DIII-D, TFTR, JET, JT-60, ASDEX, ASDEX/upgrade, TORE-Supra, CIT and DCT.

6.1.3.9 Minimal Damage of First Wall Components due to Disruption Heat Loads

The Peak loads due to plasma disruptions often cause damage to first wall components in presently operating tokamaks. A large number of disruptions in the ETR would likely lead to a very limited first wall lifetime, requiring frequent repair or replacement of many of the first wall components. The likely timescales, magnitude and location of the peak heat fluxes from disruptions need to be determined for the ETR. In addition, the number of

disruptions and their frequency need to be estimated. Techniques need to be developed to minimize the number of disruptions, and the operational limits on current, field, elongation, beta, impurity level and other plasma parameters need to be determined.

The experiments capable of providing the needed data are DIII-D, TFTR, JET, JFT2-M, PBX, ASDEX/upgrade, TORE-Supra, Alcator C-Mod and CIT. Resolution of the timescale and heat flux questions will require diagnostics dedicated to the analysis of disruptions such as IR TV cameras, fast IR systems, etc. At present, the best disruption diagnostics are installed on TFTR. Some have been planned for DIII-D. All of the other experiments could pursue the heat load and timescale issues, but disruption studies are not often given enough priority to get adequate resources to obtain definitive data. The operational limits necessary to avoid a large number of disruptions can be mapped out on the larger, high power, elongated divertor tokamaks such as DIII-D, JET, JFT2-M, PBX, ASDEX/upgrade, Alcator C-Mod and CIT. TFTR, DIII-D, JET, JFT2-M and PBX will be operating in the next few years (1987-89). ASDEX/upgrade and C-Mod will produce useful data in the 1990-93 time frame and CIT will provide data in the 1993-97 time period. The control of disruptions with current profile control could be tested in DCT in the 1995-1998 time period.

6.1.3.10 Acceptable Tritium Retention in First Wall Materials

Many candidate first wall materials such as graphite absorb hydrogen, deuterium, and tritium so that a significant quantity of tritium may be tied up in the first wall. If the quantity of absorbed tritium is very large, it impacts the allowed tritium inventory, and may lead to excessively expensive and difficult tritium inventory requirements. The tritium absorption properties of the candidate wall materials need to be evaluated, and the likely tritium absorption in the ETR needs to be evaluated in light of the recycling experience on large, high power tokamaks with various types of first wall materials.

Laboratory tests of the tritium retention properties of candidate wall materials need to be performed. In addition, models of the hydrogen recycling and retention need to be developed and calibrated on experiments on large, high power tokamaks. Experiments on TFTR, JET, CIT and T-14 with tritium will be particularly valuable in this regard.

6.1.3.11 Impurity Control with the Special Conditions Required for Current Drive

Current ramp-up and transformer recharge often require special plasma conditions, such as operation at very low densities, to be efficient. At such low densities, it is not likely that a high recycling divertor will work. The erosion of the first wall may then be unacceptably large. In addition, steady-state current drive will undoubtedly lead to increases in the net power loading on the first wall components since the first wall will intercept not only the alpha heating power but also the current drive power. Experiments are needed with current ramp-up and/or transformer recharge with divertor experiments. The exact conditions at which impurity control may be possible with a high recycling divertor and efficient current drive need to be determined. In addition, the feasibility of impurity control with double the ignited alpha power needs to be assessed.

The major near term experiments capable of assessing the feasibility of high recycling divertor operation with current drive for limited pulses are JT-60, PBX and ASDEX. These experiments will be operating in the next few years. Extension of this test for steady-state could be provided by DCT in the 1995-1998 time period.

6.1.4 HEATING AND CURRENT DRIVE

There is a growing realization in the Tokamak community that the probability of getting reliable, disruption free operation at low q_a and high β will likely require some degree of current profile control. Thus, quite aside from the need for steady-state in the nuclear test phase of the ETR, some use of current drive, in addition to the necessary heating to ignition, may also be critical to achieving a controlled ignited burn for long pulses, that is, for times longer than the plasma skin time L/R . Since all presently envisioned methods of current drive would also couple heat into the bulk plasma from the current-carrying tail electrons or from beam ions in the case of NBI, it seems preferable to emphasize R&D on systems which would both heat and drive current, such as NBI, LH and ECH. Even for the ETR inductively driven first phase, approximately 50 MW of heating would be needed for ~ 50 s, and it would be helpful in that 50 MW could also ramp up the current and shape the current profile prior to ignition, so that higher beta would be allowed at least for a 100- to 200-s skin time. For profile control reasons, it will

probably be optimum to have more than one type of heating and current drive system, within a reasonable total of 50-70 MW.

In this context, ICRH would become a candidate for part of the auxiliary heating and current drive systems if (a) the physics R&D program would show that it could heat ions selectively in an initially densified core, to aid ignition, (b) that it could still couple to a 10^{20} m^{-3} diverted plasma in a sustained H-mode, and (c) that it could also be a competitive candidate for current drive with a minimum efficiency $\eta_{20} I_{MA} R_m / P_{MW} \gtrsim 0.4$. So far, progress in (b) and (c) have been discouraging, so combinations of NBI, LH and ECH are recommended for R&D emphasis towards ETR auxiliary systems.

6.1.4.1 ECH Heating

Recent and anticipated advances in frequency and power for microwave sources have rekindled interest in ECH, and several multi-megawatt ECH heating experiments will be forthcoming in the next few years, particularly in DIII-D, MTX, T-10 and T-15, and COMPASS. The key aspects of these heating experiments to look for is the effect on confinement and beta when the resonance zone (heating position) is varied, effect on the H-mode (in DIII-D), effect on confinement with large $N_\parallel$ (more distributed deposition over the profile), and ability to heat very high density cores with pellets (250-500 GHz experiments on MTX).

6.1.4.2 High Energy NBI Heating

There will be several experiments in the next few years with beam energies above 100 keV (TFTR, JET and TORE-SUPRA). Since even higher energy beams are also a candidate for current drive in the ETR, there are questions concerning fast ion confinement at high beta (see Secs. 6.1.1.10 and 6.1.2.9) and the possibly larger role of the beam ion contribution to beta (Does the anisotropic $\beta_\parallel$ portion of beam ions count fully in the beta limit?). Another question is the potential role of multi-step ionization processes which could increase the beam energy requirements.

6.1.4.3 Ion Bernstein Wave Heating

Although Ion Bernstein wave heating is currently a dark horse candidate for heating and current drive in the ETR, it might prove to have unique advantages in initiating ignition in pellet-densified plasma cores. This

technique is being actively developed on DIII-D, PBX and would be further developed on C-Mod.

6.1.4.4 EC Current Drive

EC current drive experiments will begin on DIII-D (using CW gyrotrons) in 1988 and in MTX (using pulsed FELS) in 1989. For DIII-D, the key issue to demonstrating high EC current drive efficiency is to get high $\langle T_e \rangle > 5$ keV, possibly with the aid of neutral beam heating. For MTX, the key issue to getting high current drive efficiency at moderate T_e of a few keV, is to find ways to exploit nonlinear effects with high FEL intensities, such as (a) mode-conversion parallel beat waves or (b) asymmetric, stochastic heating.

6.1.4.5 NBI Current Drive

The physics of neutral beam current drive can be investigated with present positive ion injection and high power machines with strong coinjection. Both DIII-D and TFTR should make contributions in this area. These near-term NBI current drive experiments should seek conditions of high β_p (for larger bootstrap effects), and at maximum pulse lengths to minimize inductive effects in calibrating the current drive efficiency. Combinations of NBI current drive with ECH can be attempted in DIII-D, combinations with ICRH in TFTR, and combinations with lower Hybrid in JET (when lower Hybrid is added in 1990) and TORE-Supra. Tests with higher energy negative ion beams must, of course, wait for their development by the mid 1990's, and so could be first tested in DCT in the 1995-1998 time period. Recent proposals for multi-MeV NBI current drive using RFQ Linacs need to address the critical issue of anomalous ion drag for beam ions above the Alfvén velocity (> 750 keV in the ETR).

6.1.4.6 LH Current Drive

There will be a large set of lower hybrid current drive experiments in the next few years, greatly expanding the database for the ETR: in order of lower hybrid power, LH experiments will be performed in JT-60 (already has achieved 2 MA, but must be demonstrated at higher density), JET, TORE-Supra, ASDEX/upgrade, PBX and MTX. The primary issues are

- (1) LH current drive efficiency at $\langle n_e \rangle > 1 \times 10^{20} \text{ m}^{-3}$.

- (2) Scaling of efficiency with high T_e (this is mainly where further improvements can be expected on big machines).
- (3) Quality of the launched N_e spectrum.
- (4) Loss of fast electrons near the plasma edge, and the effect on limiters and divertors (hot spots).
- (5) Compatibility of high power ($>$ several kW/cm²) waveguide grills in devices using graphite limiters, divertors, or armor tiles (JT-60 will be a key test--this was a real limitation in Alcator C past experiments).

As there is not a good basis to expect lower hybrid current drive to be accessible to the axis at $\langle n_e \rangle \gtrsim 1 \times 10^{20} \text{ m}^{-3}$ in an ETR (because in part the Landau damping is too strong in the center, and in part due to alpha damping, experiments using combinations of LH in the edge with either EC or NBI in the core are more recommended than LH alone.

6.1.4.7 Current Drive used for Startup and Current Ramp-up at Low Density

There is a need for a lower hybrid system to both perform current ramp-up at low density as well as current drive in steady-state at higher densities. 3 GHz is certainly an appropriate frequency for ramp-up, but it is too low for the higher density steady-state. Since it is not convenient to have two separate systems, a better frequency would be 4.6 GHz.

One problem with this high frequency is the size of the individual waveguide and the walls since the latter have to be cooled. From a recent conversation with ORNL engineers (D. Nelson and J. Yugo), it turned out that it is possible to have "septums" 2 mm thick, hence high frequency is feasible.

Beneficial data input-facilities would be JT-60 has been ramping up the current (50 + 300 kA) in a PDC plasma with LH and the transformer, but with lower frequency LH. TORE-Supra and FTU should be able to give very useful information at a higher LH frequency and density. JET could have some input, depending on how low in density it can go.

Since full steady-state current drive with a combination of LH and either NBI or ECH is likely, current ramp up experiments with NBI (TFTR) and EC (MTX) should also be encouraged.

In reality, the experimental program is relatively weak in this area. Although many facilities have the needed capacity, the experiments may not be viewed as high priority due to other program issues.

6.1.4.8 Current Drive Applications to Sawteeth and Disruption Control

A major motivation for current drive in the ETR is sawteeth suppression and disruption control. Past experiments on PLT, ASDEX, Petula and Alcator C have shown that sawteeth ($m = 1$) modes can be suppressed, with large increases in control $T_e(0)$. At the same time, ASDEX and Petula have shown that by suppressing sawteeth, the $m = 2$ mode was destabilized. The interpretation is that the current profile is flattened, so that $q(a) \gtrsim 1$. Theoretically, current profiles stable to both $m = 2$ (disruptions) are possible in low q_a plasmas, and so the combination of lower Hybrid with another, localized current drive such as EC is needed. EC in particular could be used localized to the $q = 2$ surface to stabilize disruptions, as first evidenced in T-10 ECH experiments. Use of tunable FEL's would greatly enhance these experiments.

6.1.4.9 Amplification of Current Drive by the Bootstrap Effect

The bootstrap current is expected to occur in all high beta poloidal tokamaks. If τ_E scales with plasma current the bootstrap current turns out to be approximately a fixed number of amperes in a given machine for a fixed heating power. DIII-D, TFTR, JT-60 and ASDEX should all observe significant bootstrap current. Theoretically, the bootstrap amplification factor is limited by MHD-stable profiles, so that techniques for current profile control can raise the Troyon beta limit coefficient C will raise β_p as well as β_t , and improve the bootstrap effect. Also, peaked density profiles with deep-penetration fueling techniques ($\gtrsim 100$ km/s plasmoid injection) should also help.

6.1.4.10 Current Drive at High Beta

Lower hybrid alone has produced very significant current drive but only in low density and low beta plasmas. The operating density for current driven plasmas must be increased in order for the total beta to be compatible with the ETR requirement ($\beta_t \gtrsim 5\%$, $\omega_{pe}^2(a)/\omega_{ce}^2(a) \gtrsim 0.5$). Higher energy negative ion beams capable of penetrating 10^{20} m^{-3} ETR-size plasmas would be an answer, but tests await negative ion beam development. High frequency ECH is another

possibility, at 250 GHz in MTX, and plans for 120 GHz in DIII-D. All of these approaches could be advanced in TORE-Supra and DCT by the mid 1990's.

6.1.4.11 Fueling and Divertor Heat Loads with Steady-State Current Drive

One of the most important ETR issues to get data on would be understanding how one would integrate steady-state fueling (probably 10 Hz rep rate plasmoid injectors), steady-state impurity control with an H-mode plasma (probably using a divertor), plasma-wall and plasma-divertor interaction in steady-state with high heat and particle fluxes, maintenance of high beta at low q_a with sustained optimized current profiles, and with disruption stabilization. Not only is this combined regime the key to achieving high fluence and high reliability for nuclear testing in the ETR, but also is the key to attractive, practical Tokamak reactors. Although individual elements of various kinds of current drive, divertors and fueling methods could be tested on separate existing devices, the interaction of current drive, beta limits, and diverted H-mode plasmas could only be explored in a new machine having all these elements - DCT. Together with CIT and T-14 providing alpha physics (but not with current profile control, and for very short pulses), a steady-state non DT experiment in DCT would complete the physics database for the ETR to the largest extent that seems reasonable to be assured of a successful ETR, short of building the full scale ETR itself.

6.1.5 OPERATIONAL SCENARIOS

Since the ETR is assumed to operate under a variety of scenarios, the operational integration of heating, current drive, plasma position control, H-mode control, disruption control and fueling control must be developed and demonstrated. Each type of discharge will have at least three distinct phases (1) startup, (2) controlled burn (both ignited and driven in different phases), and (3) shut down. Sections 6.1.1 through 6.1.4 dealt with R&D on individual physics issues from the point of view of understanding them from theory and experiments. This section deals more with R&D on the implementation of this knowledge on improving the operation of the ETR. Once the physics issues are understood, there would need to be further physics R&D developing experimental techniques to optimize operation of the plasma under the different scenarios, which in a sense integrates all the issues simultaneously. To the extent we are successful in understanding the

individual physics issues, the operational scenarios could then succeed in approaching the ideal of operating a modern Tokamak in a very deterministic and reliable way, rather than by purely empirical recipes. Many modern Tokamaks are already operated by a programmed computer, with a few feedback elements such as plasma position control with selected PF coils. However, operation of an ETR very close to beta limits and at minimum safety factor will require a much more sophisticated real time computer control on the density and current profiles as well as the PF coils, and this also requires R&D.

The four basic operational scenarios listed in Table 6.1-5 (items 1 to 4) are given in order of increasing reliance on use of current drive. For TIBER, we have assumed that lower hybrid would be used for ramp-up and assisting an inductively-driven burn in the first phase of operation (scenario 3), followed by a fully non-inductive steady-state phase (scenario 4). For ITER, full steady-state is proposed for the second phase but the use of any lower hybrid current drive at all in the first phase of operation is still under debate, even though the US position would consider the use of lower hybrid to assist with some inductive capability to be a sufficiently-well qualified technology by 1991.

6.1.5.1 Pure Inductive Ramp-up, Ignition, and 200-s Burn

The key issues for this type of ETR scenario are:

- (1) The maximum rate of rise of current (≥ 1 MA/s may be necessary but difficult to achieve in a large ETR)
- (2) A close calibration of scaling of neoclassical resistivity corrections, including any bootstrap effects (or else the sustainable pulse length will be uncertain to within a factor greater than 2).
- (3) An understanding of the time evolution of the ohmic current profile and the associated beta limits for pulses as long or longer than the skin time $L/R > 100$ s. Here the questions are how rapidly would the beta limits degrade with the slowly relaxing current profiles, and at what point would disruptions be expected to terminate the pulse, if the safety factor were kept low?
- (4) Keeping the fusion power relatively constant during the ~200-s burn with only PF coil and fueling control (see Sec. 6.1.5.8 below).

6.1.5.2 LH-assisted Ramp-up, (~800 s) Ignited Burn

This scenario relies on lower hybrid only for current ramp-up at low density ($\langle n_e \rangle \lesssim 10^{13} \text{ cm}^{-3}$), so that the ultimate LH current drive efficiency is not so much an issue. The flat top would be sustained only by ohmic and alpha heating, equivalent to $Q \gtrsim 100$, since the ohmic heating is very small at ignition temperatures. This scenario encounters the same issues (2), (3) and (4) identified in 6.1.5.1 above [but of course avoids issue (1)]. The advantages of this scenario are that, with respect to issue (3), the lower hybrid current ramp-up before densification might "freeze-in" a more favorable initial current profile, prolonging the subsequent stability of the ignited burn, and, on the practical side, greatly reduces PF coil requirements for startup, prolonging the cyclic fatigue life of the PF coils. A new issue arises here, which is how fast one can densify a low density ramp-up plasma. Normal pellets may not penetrate even the low density LH ramp-up plasma because of the presence of an intense first electron tail. However, plasmoid injection may work.

6.1.5.3 LH-assisted Ramp-up and High Q, Inductive + LH-assisted Burn

This scenario extends the performance of scenario 6.1.5.2 in that the same LH system used for low density current ramp-up is also left on (but perhaps with some adjustment of the N spectrum by LH phase control), to assist the flat top burn in both power balance and partial current drive in the outer edge regime of the profile. Here the core would be dominantly alpha heated, with the edge heated by the LH current drive. Here the R&D issues are:

- (1) Can the LH N spectrum be controlled so that both heating and current drive are restricted primarily to the outer edge of the profile which is still accessible at the operating density of $\langle n_e \rangle \gtrsim 10^{20} \text{ m}^{-3}$?
- (2) How much can the loop voltage be reduced during flat top due to the LH edge current drive?
- (3) Is alpha heating physics for long pulses ($\tau \gtrsim 800 \text{ s}$) still dominant enough in the core to allow study of pure alpha heating for the long pulses?
- (4) Can the inductive + LH current profile be maintained in a way to avoid sawteeth and disruptions for larger pulse?

6.1.5.4 Full Non-Inductive Steady-State, $Q > 5$

This scenario, of course, depends on the successful development of one or more current drive techniques discussed in Section 6.1.4, capable of minimum efficiencies $n_{20} I_{MA} R_g / P_{MW} \gtrsim 0.4$, and penetration to the axis with alphas and $n_e(a) > 1.5 \times 10^{20} \text{ m}^{-3}$. This section is concerned with how we learn to employ one or more of these current drive techniques, assuming they are successful, to not only sustain the current but keep the current profile optimized for stability, and also to use the current drive power and radial disposition to control the burn temperature and fusion power. The primary purpose of the steady-state ETR phase of operation is nuclear testing of blankets under reactor relevant conditions of steady-state, high reliability, availability, and neutron fluence. Thus, a key issue with this scenario is the control be driven burn to

- (a) keep the fusion power and neutron wall loading as steady as possible (lest the advantages of steady-state be lost),
- (b) reduce cyclic fatigue limits on the divertor plates and walls,
- (c) keep the burn duration going as long as possible by optimizing plasma edge conditions to minimize long term divertor and wall erosion, and
- (d) keep the plasma virtually disruption free, by tight control of the plasma beta and current profile.

This is an extremely important R&D issue, not only for a successful, high availability ETR but also for practical Tokamak reactors. Aside from the ETR itself, there is only one experimental facility in the world which comes close to addressing some of these needs--TORE-Supra, which does not have a divertor, and one proposed--DCT, which would have a divertor.

6.1.5.5 Plasma Startup and Turnoff

This item deals with a collection of machine control issues for startup and turnoff of the various operational scenarios. For the pulsed inductive scenarios, there is a premium on minimizing the time required to startup and turnoff, in order to maximize the machine duty cycle. We have already mentioned the current ramp rate limit issue in 6.1.5.1. Here one of the key issues is the maintenance of the x-point position with divertors during

startup and turnoff, to mitigate peak heat loads and erosion rates, which are likely to be worse when the x-point is absent or moved significantly from its desired position during the current ramp-up or ramp-down. Also, the maximum PF coil limits in $B_{\max}$, $JB_{\max}$, or stress may occur during startup or turnoff, when the plasma current is high but the plasma beta is low, for highly shaped plasmas. Also, when current drive is used in the steady-state scenarios, the ratio of power input to the current driven (inverse efficiency) that is optimum for the steady-state, may need to be higher during the current ramp-up and ramp-down, when confinement may be worse. In such a case ways to control the current drive efficiency downwards during ramp-up or ramp-down may be needed. This may also be needed to control the fusion power level to compensate for any fluctuations in H-mode confinement time, when the machine is running at the plasma current limit (see Sec. 6.1.5.6 next).

6.1.5.6 H-mode τ_E Control

In general, the problem of keeping the fusion power constant (Sec. 6.1.5.10) may require some method of controlling the confinement time. Although the fusion program has been correctly pursuing "up button's" for τ_E , it is quite conceivable, even likely, that once sufficient τ_E is reached in the ETR to either ignite (in the first phase) or reach say, $Q = 5$ (in the second phase), then plasmas burning at or near a beta limit may suffer from even a small fluctuation in τ_E above an initial balance point, if it turned out the beta limit was a "hard" (disruptive) limit at low q_a .

In such a case, we need to find externally-controllable machine "knobs" that could fine-tune τ_E . For example, TIBER optimizes Q versus wall loading for a given current drive efficiency $nIR/P = 0.4$, with an H-mode multiplier $\times \tau_E$ (Kaye-Goldston) where H is desirably less than 2, typically 1.2-1.4. Higher $H\tau_E$ would imply less current drive power, which leads to less current, lower beta limit, and undesirably lower wall loading for nuclear testing, for example. At present, what is needed most here is an understanding of the H-mode mechanism (see Sec. 6.1.1.2 above), but failing that, we could try exercising more the key ingredient to H-modes: an x-point just inside the wall (+ minimum power). Perhaps jiggling the x-point location a few centimeters closer to the divertor with a couple of selected PF coil currents would suffice to control H.

6.1.5.7 Plasma Position Control with Strong Shaping

It has been generally seen that active feedback control of plasma position becomes more difficult for highly elongated plasmas (see Sec. 6.1.2.5). For highly elongated plasmas, the shape of the current profile is important -- broader profiles with current drive couple more readily to available PF equilibrium and trim control coils. Here the R&D issue is the coupling of x-point location with the edge plasma recycling, and thus indirectly, with plasma fueling and H-mode confinement. To handle peak heat loads, most divertor designs are curved, with flux layers making a small angle of attack to the divertor surface. Under such conditions, the position of the x-mode either needs to be held with high precision, or deliberately jiggled back and forth with enough amplitude to spread the scrape-off layer over a sufficiently large zone.

6.1.5.8 Variable Fueling Control

The primary control for long pulse ignited plasmas (after ignition occurs) is either pellet (plasmoid) fuel mixture, speed and rep rate, or beta-limit control (or both), assuming the beta-limit is soft (τ_E - degrading rather than disruptive). Considerable R&D is needed if all the above fueling parameters were to be controllable in real time, for 3-10 Hz rep rates, not just for the injectors themselves, but on learning how to use such control knobs once that fueling capability is available. More than one type of injector may be needed to give the full dynamic range of control, and tests would be needed on machines with sufficient pulse length ($> 10 \tau_{\text{particle}}$, effective, which may be $> 100 \tau_E$).

6.1.5.9 Current Profile Control

We have already mentioned current drive and its role in current profile control for beta and disruptions (see Secs. 6.1.2.2, 6.1.2.3 and 6.1.2.10). Here we need only to add the point that this current drive control must be integrated with position control, τ_E control, and density (fueling) control, in order to achieve an overall optimum in machine performance.

6.1.5.10 MHD Activity Monitoring and Disruption Prevention

Experimentally, precursor magnetic fluctuations can often be measured prior to disruptions, and this motivates consideration of how to use this data

in real time to adjust machine controls away from an impending disruption condition. Key issues here are: how much time to precursors give to effect disruption prevention measures, and which measures would be most effective. Also, some types of disruptions do not exhibit precursors or have very short precursors, and so what other signatures of danger could be monitored for these types of disruptions (e.g., a current profile slowly evolving in the "wrong" directions).

Also, for optimum burn control and keeping central ash concentrations down to manageable levels, some level of sawtooth activity may actually be beneficial. Keeping an optimum sawtooth activity level presumes some means of sawtooth control (e.g., current drive).

6.1.5.11 Alpha Heating Effects on Operating Point Control and Burn Control

The fact that alpha-slowng down times are finite (a few seconds in an ETR) introduces a non linear response to changes in density (fueling control) or temperature (heating control) on time scales comparable to or longer than the alpha slowing down time. This is particularly of concern to burn control of an ignited or very high Q burn as in phase I of the ETR. Data on the time-dependent α -effects important to burn control can be studied mainly on ignited plasmas in CIT and T-14, and possibly in the core of JET, assuming favorable H-mode confinement.

6.2 TECHNOLOGY DEVELOPMENT FOR TIBER II/ETR

The development program necessary to build the TIBER/ETR and extend it to the ITER is more extensive than the entire OFE Development and Technology Program. Every program element is beyond the scope and available time of this report. Accordingly, focus is placed on high-leverage design issues for which a proof-of-principle or other significant milestone can be achieved prior to an assumed construction decision in the early 1990's. The briefest summary of these items is given in Table 6.2-1. These tasks are subdivided into four major categories:

- (1) Technologies that reduce costs.
- (2) Technologies for steady-state.
- (3) Reliability.
- (4) Environment.

6.2.1 TECHNOLOGIES THAT REDUCE COSTS

Smaller size can be directly related to smaller costs. That is because most Tokamak component costs are related to the surface area or the square of the plasma major radius. This is obvious for the plasma vessel, first wall, shield, etc. However, it is also true for the magnets, since the required ampere meters of conductor increase as the radius squared for superconducting magnets (as opposed to a cubic relationship for pulsed copper magnets). A major driver in the tokamak size is the radial build of the center post region. Accordingly, it is the magnet current density and neutron shield thickness that most greatly influence the reactor size. A thin efficient shield is as important as magnet current densities, along with a more radiation tolerant magnet winding, superconductor and insulators.

The critical R&D requirements for the TIBER magnet design effort are epitomized by requirements for the central solenoid. These are the high fields and high current densities as well as the ac losses and heat removal requirements when ramping these coils to drive plasma current. The TF coils themselves must achieve relatively high magnetic fields while withstanding high nuclear heat loads. The outer PF coils, although not trivial from a structural standpoint, are less demanding in the sense that they are generally

Table 6.2-1. Validating R&D for TIBER II/ETR.

Task	3-Year Milestone	Design Goal
I. Technologies that Reduce Cost		
A. Radiation-tolerant, high current density magnets	4 kA/cm ² , 14 T, 40 cm bore coil	4 kA/cm ² , 14 T, 10 ¹⁹ n/cm ² , 4 meter major radius TF coils
B. Radiation-tolerant insulators	Low temperature data base on polyimides and advanced epoxies	10 ¹¹ rads, 400 MPa bearing strength
C. Pulsed ohmic heating coils	1 meter, 10 T pulsed coils	2-3 meter, 14 T, 4 kA/cm ² , O.H.
D. Shielding effectiveness	Shielding experiments correlated with design	10 ¹⁹ n/cm ² at magnets in 10 ⁸ sec, and acceptable magnet heating
II. Technologies for Steady-State		
A. ECH for disruption control	245 GHz, 2 MW, 10% tunable oscillator for FEL, ETA II Alternate: 245 GHz tunable FEM, d.c.	170-500 GHz, 10-30 MW and 10% tunable FEL
B. Negative ion sources	1 Amp at 400 keV D.C. accelerator	100 Amps at 500 keV
C. Negative ion RF quadropole accelerator	1 Amp at 1 MeV	10 Amps at 2-3 MeV
D. Photodetachment neutralizer	Highly reflective laser cell for 1 Amp	90% efficiency D ⁻ D ⁰
E. Pellet fueling	5-10 km/sec	50 km/sec
F. Alternate plasmoid fueling	750 km/sec with 1 milligram	750 km/sec, 5 mg and 1 Hz

Task	3-Year Milestone	Design Goal
III. Reliability		
A. Carbon/carbon fiber composites for divertor	10^{21} n/cm ² with less than 4% swelling and low erosion	10^{22} n/cm ² with low swelling and low tritium retention
B. Alternate divertor materials	Beryllium/tungsten data base and tokamak experimental evidence of H-mode	low sputtering and erosion with redeposition
C. Stress corrosion of steels in water	200 MPa and 10^8 second life	600 MPa and 10^9 second life
D. RF components	Antenna and insulator tests in ionizing radiation	20 MW LH antenna in plasma environment
E. Properties of redeposited divertor materials	Physical/mechanical properties of C, W, Be neutron radiation effects	Net erosion rates of 1 cm/year
F. Copper alloys for divertor cooling	Data base for joining of strengthened alloys, low temp irradiation effects, tritium permeation and retention	3 MW/m ² heat flux with less than 10 g of tritium retention and 10 curries/day permeation
G. Beryllium properties	Compatibility with breeder/coolants, mechanical integrity, and multiplication	10^8 second life at 2 MW-yr/m ² with negligible corrosion
H. Remote vacuum seals	Demonstrate 1 meter diameter valve and remote vacuum seal	Remote sealing of inner plasma chamber
IV. Environment		
A. Low activation structural alloy	Metallurgical modification of austenitic stainless steel to remove niobium and moly	Meets 10 CFR61 class C, withstands 60 dpa

subjected to lower fields and slower ramp rates than coils in the central solenoid. Specific R&D Issues Associated with ETR Magnet Design are as follows:

- Develop high field (~15 T) and high current density conductors compatible with design aspects required of the central solenoid.
- Develop conductors with acceptably low ac losses at moderate fields and ramp rates for the ohmic current drive aspects of the central solenoid.
- Develop reliable insulation systems that will tolerate the high radiation environment of the TF coils while providing adequate strength and allowing reasonable terminal voltages.
- Develop methods for quench protection and heat removal, and confirm adequate conductor stability for each of the ETR magnet subsystems.
- Develop high pack current density winding for ETR-TF coils subjected to high heat loads.
- Develop appropriate winding techniques (e.g., winding before or after reaction of superconductor).
- Design low heat loss joints and terminations for superconductors.
- Develop appropriate sheathing alloys for conductor.
- Structural design analyses for PF and TF magnets from LCP data base.

Assuming that the superconducting magnets for ETR will be constructed with internally-cooled cabled superconductors (ICCS), the most effective and efficient way to address the magnet design needs is to focus on scaling parameters for the conductor. Several materials issues will have to be resolved prior to materials procurement and construction of TIBER II. These investigations will include characterization of the performance of available materials in a "TIBER II environment" and development of fabrication technology for component assembly. The research and materials development can be divided into two broad areas; magnet structural materials and pack materials.

6.2.1.1 Structural Materials

Structural materials include both the case external to the winding pack and the reinforcing channel and conduit material that are part of the

conductor system. Due to the strain effects in the superconductor, most of the magnets will likely be wound and then reacted. This may not be necessary on the larger PF coils. This will require that this class of materials be able to survive the Nb₃Sn heat treatment and simultaneously meet stringent mechanical performance requirements. Case materials are considered independently of the winding pack.

The cryogenic materials properties of greatest importance for magnet design are yield and tensile strengths, thermal expansion, magnetic permeability, thermal conductivity, fracture toughness, fatigue, weld cracking susceptibility, aging kinetics, microstructural stability, etc. Since different conductors will be specified for both the PF and TF coils two different conductor heat treatments are likely. This will require that mechanical tests and characterization be performed separately for both the PF and TF coils systems. In addition to mechanical properties measurements, welding parameters and non-destructive weld evaluation methods must be developed to increase magnet reliability, particularly for pulsed operation scenarios.

6.2.1.2 Magnet Pack Materials

Magnet electrical insulation will be a composite consisting of ceramic/glass fibers in an organic binder such as epoxy resin or polyimide. The operation temperature of the insulator will be 4.2 K, while in the TF coils it will have to support average compressive loads of 400 to 600 MPa with high peaking factors. Insulation is also subject to alternating shear loads due to varying electromagnetic forces. Current magnet design does not call for the insulator to support or transmit these shear loads. However, the conductor insulation must be capable of surviving the abrasion generated by shear without a failure that results in loss of electrical insulation.

Selection of an insulating composite for the TF coils will be driven by the need to accept a high fluence of incident neutrons and gamma (10^{11} rads total). The insulator must maintain electrical isolation between turns at the end-of-life and prevent excessive mechanical motion of the conductor. The need for survivability of the TF coil insulation in a high neutron fluence may result in selection of an impregnate that is different from that used in the PF coils.

Uncertainty in shield effectiveness (both bulk shielding as well as penetration shielding) must be ascertained and reduced as needed by both analytical and experimental methods.

6.2.2 TECHNOLOGIES FOR STEADY STATE

Steady-state current drive and continuous plasma fueling are essential for steady-state operation of TIBER. The primary current drive method has evolved to be based on negative ion neutral beams and lower hybrid radio frequency waves. Lower hybrid has been demonstrated in JT-60 and other tokamaks. Unfortunately, it does not penetrate to the center, such that an auxiliary method is needed. Negative ion neutral beams with voltages about 500 kV can penetrate to the center of TIBER and drive current. The physics is reasonably understood and current drive has been demonstrated in TFTR.

6.2.2.1 Negative Ion Sources

The LBL group is developing a small negative ion source of less than 1 A. This source would need to be expanded and linked to a DC or RF accelerator to produce at least 500 keV. An early step shown in Fig. 6.2-1 is to produce 170 kV at LBL and then 400 kV in the RTNS II facility. Further development would require a multiampere 500 kV test stand. Alternately, it may be feasible to pursue R.F. acceleration by one of several means to a few MeV.

6.2.2.2 Electron Cyclotron Heating - ECH

While ECH is a back-up option for current drive, 10 MW is included in the TIBER II baseline for auxiliary heating and disruption control. A tunable oscillator needs to be developed to drive a free electron laser or maser. It should be slowly adjustable between 250 and 600 GHz, and electronically tunable in milliseconds by about $\pm 5\%$ for disruption control. As an alternate amplifier a free electron maser (steady-state) can be developed with efficient energy recovery.

A primary candidate for the tunable ECH oscillator is a Backward Wave Oscillator (BWO). It is tunable by changing the electron beam voltage quickly as a tool for disruption control.

M-T-30

EFCR

N.B.

Figure 6.2-1. Engineering Test Reactor logic diagram.

6.2.2.3 Deep Fueling

Continuous deep fueling of the plasma core will be needed for steady-state. High speed pellets may be achieved with an electron beam driven rocket to 10 km/s. Laser beam drive pellets may be driven even faster to 50 km/s. Alternately, injection of plasmoid may be possible, if impurities can be controlled, and the density raised.

In the present RACE (Ring ACcelerator Experiment) configuration of Hartman and Hammer the toroids are formed by a magnetized coaxial Marshall gun. The gun is fired following the establishment of an initial magnetic field. A resulting plasma, with embedded toroidal accelerating field, emerges from the gun and stretches the initial field within a flux conserving region. Reconnection of this initial field produces a dipole-like, poloidally-directed field and results in an isolated trapped plasma ring. In typical rings, the internal poloidal and toroidal fields are of comparable magnitude. Masses of 0.5 mg have been accelerated to velocities of ~400 km/s, while velocities of over 2500 km/s have been achieved. At a velocity of 1400 km/s and a ring mass of 0.018 mg, approximately 18 kJ of the capacitor-bank energy of 50 kJ was converted into kinetic energy, resulting in a wall-plug to injected efficiency of ~30%. The total capacitor bank energy available for these experiments is 260 kJ although full use has not been made of this to date.

6.2.2.4 Photodetachment

Photo-detachment was proposed earlier as a future neutralizer of high energy neutral beams. Now mirrors with absorptivities of 10^{-4} have been developed and projected advances in solid-state laser-pumped Nd:YAG lasers bring photodetachment to the brink of creditability. To be practical, the lasers must be scaled from their present 60 to 80 mW output up to 300 W.

To neutralize 95% of a 500 keV D- beam, studies show two optical cavities would be needed, mounted in series, each roughly 3/4 m long, 1/2 m wide and about 1 m high. The cavities consist of two mirrors of 99.99% reflectivity, mounted face to face and excited by 4 rows of YAG lasers. The mirrors are irradiated by a maximum of 40 kW/cm^2 of light and each row of lasers, consisting of 50, 1 cm^2 YAG laser rods, emit 11 kW. Assuming the lasers operate at 4% wall-plug efficiency, the neutralizer requires 2.2 MW. The efficiency of a beamline with photodetachment would exceed that of a beamline

with a gas cell, that neutralizes only 60% of the ions, when forming only 7.2 A of neutrals.

To neutralize 95% of a 3 MeV beam, 4 cavities are needed, each 1 m long, 1/2 m wide and about 1 m high. As before, the cavities consist of two mirrors of 99.99% reflectivity, exposed to a maximum 40 kW/cm^2 of light and illuminated by 4 rows of 50 YAG lasers. For this application, each row of lasers must emit 13.5 kW, making the needed wall-plug power equal to 5.4 MW. A photodetachment cell would be more efficient than a gas cell to form a 3 MeV neutral beam of 2.9 or more.

6.2.3 TECHNOLOGIES FOR RELIABILITY

In the context of TIBER II/ETR reliability can mean that the experiment has high availability or it can mean that it is reliable enough to work at all. The plasma divertor and first wall armor fall into both of these categories. There are immediate issues to choose the final design, plus lifetime issues to survive a reasonable time: longer term issues address the extension of the technology to a demonstration reactor.

6.2.3.1 Divertor Engineering

Immediate Issues - R&D needed to choose final design and verify viability of that design.

- a. Fabrication of W tiles brazed to Cu tubes -- Selection of materials and fabrication techniques.
- b. Testing of target in high heat flux environment -- Verification testing under simulated high heat flux (6.6 MW/m^2) conditions, plus simulated disruption heat loads.
- c. Measurement of sputtering and impurity transport -- Experiments in a high-recycle tokamak divertor plasma to measure impurity transport into core plasma and to resolve materials redeposition issues in order to assist in materials selection.

Lifetime Issues - R&D needed to verify that the design will survive the required 1 full-power-year lifetime.

- a. Measurement of tritium inventory and permeation -- Experiments with simulated divertor components in a plasma tritium source to provide data for tritium inventory and permeation calculations.

- b. Measurement of erosion -- Experiments in a high recycle tokamak divertor plasma to resolve divertor erosion under normal and disruption conditions.
- c. Low fluence irradiation tests -- Tests of irradiated samples ($\sim 1 \text{ MW-y/m}^2$) to investigate radiation damage and possible synergisms with tritium permeation and inventory.
- d. Develop instrumentation -- Develop techniques (e.g., acoustic and tritium signatures) to predict cooling tube burnout or failure. Develop radiation-hardened diagnostics for reliable in-situ thermal and stress measurements.

Long Term Issues - R&D needed to carry divertor designs forward to DEMO and other high power, higher fluence machines.

- a. High fluence irradiation tests -- Tests of materials and components at moderate fluences ($> 5 \text{ MW-y/m}^2$) to verify reactor-level lifetimes.

6.2.3.2 First Wall Armor Engineering

Immediate Issues - R&D needed to choose final design and verify viability of that design.

- a. Tile materials development -- Measurement of thermal and mechanical properties of selected C/C composite tile materials.
- b. Fabrication, mounting and replacement -- Verification of design and remote replacement viability.
- c. Testing in a high heat flux environment -- Verification testing of designs under simulated high heat flux (0.5 MW/m^2) conditions, plus simulated disruption heat loads.

Lifetime Issues - R&D needed to verify that the armor will survive the desired 10 year (5 MW-y/m^2) lifetime.

- a. Development of radiation-resistant C/C composites -- Development and testing of C/C materials for service up to 5 MW/m^2 . Measurement of 14 MeV neutron damage needed.
- b. Testing of carbon-coated wall -- Develop and test techniques for carbonizing the outboard first wall, verify adherence.

- c. Measurement of tritium inventory -- Experiments in a plasma tritium source to provide data for tritium inventory calculations.
- d. Measurement of erosion -- Experiments in a tokamak plasma to resolve first wall erosion under normal, transient and disruption conditions.
- e. Develop instrumentation -- Develop radiation-hardened diagnostics for reliable in-situ thermal and stress measurements.

Long Term Issues -- R&D needed to develop first wall armor for DEMO and other high power, high fluence machines.

- a. Disruption control -- Development of techniques to control disruptions so that thick armor is not needed.
- b. Design development -- The first wall for DEMO and commercial reactors will be much different than that for ITER. Design must be developed and tested.

6.2.3.3 Stress Corrosion

Another area of reliability is stress corrosion of steels in water and lithium nitrate solutions. The TIBER/ETR activity has identified the aqueous self cooled blanket (ASCB) concept as a leading candidate for a low-technology breeding-shield. The aqueous solution adopted consists of ~16 grams of dissolved LiNO_3 (lithium nitrate) per 100 cc of water; and the structural material is PCA (primary candidate alloy, a stainless steel). On the outboard the breeding-shield would consist of beryllium (for neutron multiplication) and the aqueous solution as the coolant/breeder. Inboard, the breeding-shield would consist of tungsten alloy (for enhanced shielding) and the aqueous solution as the coolant/breeder. Preliminary thermal-hydraulic calculations for both the inboard and outboard regions indicate that the aqueous solution temperature would range between ~40-80°C, the flow velocity would be ~0.2 m/s, and the pressure would be ~0.2 MPa.

There are no available corrosion data on tungsten, or ferritic stainless steels in LiNO_3 solutions. There is almost no information in the literature relating to the corrosion behavior of austenitic stainless steels in LiNO_3 solutions. The only references which are available relate to the addition of

the nitrates as inhibitors to localized corrosion in chloride containing solution.

It has been proposed that both austenitic and ferritic stainless steels be exposed to flowing solutions of LiNO_3 in water as a function of solution concentration, at 40°C and at 80°C . By increasing the concentrations of the reference solution of 16 g LiNO_3 /100g H_2O , we can examine the possibility of potential solution concentration through such processes as evaporation or occluded cell formation. These problems have proven to be endemic in other nuclear applications such as in power generation units.

6.2.3.4 R.F. Components

Antennas and diagnostic insulators will need to work in a highly ionizing radiator flux. Initial operation must accommodate to this new environment and withstand the full neutron fluence of the first wall.

6.2.3.5 Redeposited Materials

The first wall and divertor will suffer erosion. Especially during a disruption significant quantities of materials will be vaporized and redeposited. The properties of such materials need to be investigated for mechanical integrity, thermal properties and tritium retention.

6.2.3.6 Copper Alloys

Since copper will be necessary to cool divertors and other high heat flux components, more complete neutron damage data will be needed. The behavior would be a function of the copper alloy, heat treatments and environmental conditions.

6.2.3.7 Vacuum Seals

Large remotely demountable vacuum seals and valves will be needed to assemble the machine and close off ports. The development of expandable metals seals appears promising to allow simplified maintenance.

6.2.3.8 Beryllium Properties

Corrosion of the beryllium in the aqueous breeding solution is not very well studied and is worth studying in the presence of neutron irradiation to see if there are synergetic effects. Irradiation damage is a primary

consideration. Beryllium, already a brittle material, becomes more brittle under irradiation and is expected to undergo cracking with irradiation. Considerable amount of cracking is tolerable before the beryllium must be changed out based on experience in water cooled ($<100^{\circ}\text{C}$) fission reactors especially at INEL. The first wall fluence of 4.7 MW/m^2 in TIBER corresponds to $2.6 \times 10^{22}\text{ n/cm}$ ($E > 1\text{ MeV}$) and 18,000 appm He in Beryllium. There is experience to these fluence levels in fission reactor irradiation and on this basis we can predict the lifetime of the pebbles may be adequate for meeting TIBER's goals (4.7 MW/m^2), however, more study should be carried out to determine the failure mechanism for spheres and determine its dependence on fabrication methods. The usual way of fabricating beryllium is to hot press a powder and then machine to final size. Much cost savings (factor of two or more) is possible by cold pressing a powder and self or pressureless sintering. The lifetime of this much lower cost product under neutron irradiation is entirely unknown.

6.2.4 ENVIRONMENT: LOW ACTIVATION STRUCTURE

Since TIBER II would be representative of a future fusion reactor, it would be desirable to build it of low activation structural materials for safety and decommissioning considerations. It may be possible to metallurgically modify austenitic stainless steels by substituting alternates for niobium and moly. The remaining nickel may be satisfactory for near surface disposal per 10 CFR 61 Class C regulations. If so, the fusion image can be preserved and safety enhanced.

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