CHAPTER 18: KEY ISSUES AND R&D ISSUES

(Collection from Previous Chapters)

18. KEY ISSUES AND R&D ISSUES

18.0 Introduction

This chapter is provided for the convenience of the readers. It lists the key issues and the associated R&D issues pertaining to all the concepts explored in this report during the APEX study. Also listed are issues regarding available database for liquid breeders, plasma-interface issues and edge modeling, materials issues, and safety and environmental issues. Below excerpts taken from each chapter (and its summary) as provided by the lead authors.

18.1 Thick Liquid Blanket Concept (Chapter 5)

Among the challenges of liquid walls being addressed in APEX are: 1) determining limits on the amount of material allowed to evaporate or sputter from liquid surfaces based on sophisticated plasma edge modeling, 2) evaluating temperature profiles on fast flowing free-surface liquids in the fusion environment, 3) establishing hydrodynamic models and exploring various thick-liquid formation schemes in different MFE confinement configurations, 4) developing a 3-D free surface hydrodynamic and heat transfer simulation code that incorporates MHD effects, 5) assessing multidimensional effects (e.g. penetrations) and the impact of time-varying magnetic fields on the flow characteristics, and 6) identifying high-temperature structural materials that are compatible with Flibe and Sn-Li for use in nozzles, vacuum vessel, and other regions behind the liquids. The R&D issues are:

1. Identification of the most promising hydrodynamics configurations with respect to different MFE confinement schemes.
2. Experimental data on the achievable minimum liquid surface temperature w/o MHD effects for turbulent Flibe and MHD laminaralized lithium/tin-lithium flow under high power density conditions.
3. Identification of practical heat transfer enhancement schemes necessary for minimizing liquid surface temperatures.
4. Experimental characteristics of small-scale hydrodynamics configurations applicable to MFE confinement schemes w/o MHD effects.
5. Computer simulation results of MFE relevant 3-D free surface liquid wall thermal and hydrodynamics performance with MHD effects. In particular, hydrodynamics characteristics near the penetrations and supply and return lines.
6. Modeling and test results of the maximum allowable evaporation rate from the liquid surface with respect to different MFE confinement schemes.
7. Hydraulic component development for liquid walls. Implementation of the liquid wall idea relies on the successful development of inlet and exit nozzles that are drip and splash free and are efficient in head recovery at outlet.

18.2. The Electromagnetically Restrained Lithium Blanket Concept (Chapter 6)

The key issues with the EMR Lithium Blanket concept all are based on the difficulty of predicting its performance. At the present time, there are no computer tools or other methods to design such a system. At this time, the key issues are all associated with developing methods to predict how it and variations of it would behave, and experiments with the goal of benchmarking those experiments.

18.3. Thin Liquid Wall Concepts and the CLiFF Design (Chapter 7)

There are several dominant issues that go directly to the feasibility of this concept, and many more issues that weigh heavily on the ultimate attractiveness. The amount of allowable evaporation must be determined for all liquid candidates. This is both a feasibility issue and an attractiveness issue. We recognize that a fully consistent answer to this question will require a considerable amount of research in modeling and analysis of plasmas with liquid wall boundaries, as well as experimental research in various confinement devices. In this section, then, we look to the most serious issues associated with the hydrodynamic feasibility for implementation in the ARIES-RS type reactor. In addition, we pay some attention to serious system issues associated with tritium retention and permeation and material compatibility. These issues are summarized in Table 7.10-1 along with a rating of their relative critical need.

18.3.1 Hydrodynamic and Heat Transfer Feasibility Issues

The issues in this category differ significantly for molten salts versus liquid metals. For Flibe, the main issue concerns the penetration of heat at the free surface and the availability of a robust operating temperature window. Other issues as to the formation and removal of the liquid flow in the plasma chamber, and the accommodation of penetrations are also serious, but in our opinion solvable via numerical modeling and scaled experiments with Flibe simulants (such as water). These issues are pointed out at the end of Section 7.6. The heat transfer issue is a more serious unknown, as current limits on surface temperature for Flibe are estimated by the plasma interface group at about 560°C. Also a serious issue for Flibe, is the behavior in the divertor region, where direct plasma contact occurs. The amount of material sputtered and redeposited needs to be determined before accurate plasma modeling of the region can take place.

The main issue facing liquid metals is of course that of MHD interaction. The CLiFF flow itself is very sensitive to changes in drag since the only driving force is gravity. Without toroidal axi-symmetry of the flow and field, reliable insulator coatings will be required on all surfaces in contact with the LM layer. The MHD forces from any surface-normal components of magnetic field can upset the force balance, especially when complete axi-symmetry is assumed in the toroidal direction. Additionally, gradients in toroidal or surface normal fields can exert a significant drag on the free surface flow. All these effects need to be analyzed in greater detail, with both modeling and small-scale experimental efforts to see if a suitable flow is indeed possible in the real fields of a

Tokamak or other plasma confinement device including the accommodation of required penetrations.

LMs however, offer the potential for active control that is not present with the molten salt. By biasing and applying electric currents, the LM can be pumped or pushed against the back-wall *in-situ* – offering the chance to "confine" the liquid wall just as we confine the plasma. Indeed we may find that such measures are required in order to utilize LMs at all.

Apart from the free surface flow itself, MHD issues exist in the supply and drain lines and blanket flows as well. Insulator coatings are needed for these structures. Additionally, due to the large LM flowrates required for CLiFF, large pressure drops are expected in the entrance regions between toroidal field coil legs. These pressure drops can theoretically be overcome by *in-situ* LM pumping, but lead to very large pumping powers for the CLiFF designs with LMs. This issue will require a careful design work and analysis of the supply/exit channel geometry so that the pumping power is reduced to an acceptable level.

18.3.2 Other system Attractiveness Issues

Impact of liquid wall implementation on other reactor systems is another category of issues for the CLiFF concept. In particular, it will be likely that heating and diagnostic ports must be redesigned to allow flow to pass around the penetration. Pumping systems with a considerable amount of vapor from liquid evaporation will need to be modified. Tritium recovery (especially with hydrogen getters like lithium) will be even more challenging, and material selection and compatibility to help optimize liquid wall performance must be addressed. Flibe chemistry, decomposition and corrosion issues must be addressed for all liquid wall and blanket options.

Ultimately a system study to weigh the relative effect of liquid walls on the entire reactor design and operation will be needed.

Table 18.3-1: Key Issues and R&D for CLiFF Concepts
(x's indicate relative critical need, - indicates this is not an issue for this material)

Issue and/or R&D Area	**Flibe**	**Li**	**Sn-Li**
Plasma Compatibility	xxx	xxx	xxx
MHD Surface Heat Transfer	xxx	x	x
MHD Drag in FW	-	xxx	xxx
MHD Drag in Supply/Exit	-	xxx	xxx
MHD Effects due to Plasma Shifts	-	xxx	xxx
Active MHD Control	-	xxx	xxx
Tritium Recovery	x	xxx	x

Tritium Permeation	xxx	x	xxx
Effect on Plasma Stability	xx	xx	xx
Accommodation of Penetrations	xx	xxx	xxx
Improved Design	xxx	xxx	xxx
Chemistry and Corrosion	xxx	x	xxx
Material Database	xxx	x	xxx

18.4 Data Base for Liquid Breeders and Coolants (Chapter 8)

The material properties of lithium, Pb-17Li and $(LiF)_n(BeF_2)$ have been reviewed and summarized in this report. It will be useful for fusion scientists to use the same material properties in the fusion engineering work.

The new Sn-Li breeding material for D-T fusion has been proposed. The key advantage of this new breeding material is the very low vapor pressure, which is dominated by lithium. Therefore, this material is well suited for open channel applications such as APEX, ALPS, and inertial confined fusion. It is expected to have low chemical reactivity with air and water, and have moderately low tritium solubility. Recent experimental results show it has excellent thermal conductivity.

The tritium breeding capability is marginal for a conventional blanket. Detailed tritium breeding calculations are being done to resolve this issue. Also, safety assessment is being done to assess the activation due to the Sn, and its impact on safety.

Some preliminary experimental work has been planed in the US, which includes Sn-Li preparation, sputtering measurements, chemical reactivity with air and water, vapor pressure measurement, as well as small scale experiments on the material capability tests.

Most material properties of Sn-Li are not available. While work is on going to measure Sn-Li material properties, some suggestion of the properties to use are summarized here. Those properties are preliminary, and subject to change when more reliable information becomes available.

18.5 Li2O Particulate Flow Concept and the APPLE Design (Chapter 9)

Li2O has good tritium breeding potential. This is partially due to the high lithium density in the Li2O, which is higher than the lithium density in pure lithium. This is only one of the few materials that has the potential to sufficiently breed tritium without a neutron multiplier. With the low structural fraction in the APEX-type design, the tritium breeding will not be a serious issue. Neutronics calculations confirmed the tritium breeding capability.

Li2O has a very low electrical conductivity. Therefore, MHD will not have severe impact on the flow of the Li2O stream.

The thermal conductivity of Li_2O is rather low. This causes problem on removing surface heat, which will be the issue on the divertor and the first wall of the blanket. The heat transfer issue can be alleviated due to the high allowable surface temperature of the Li_2O. Also, an increase in the Li_2O velocity will lead to an increase in the heat removal capability of the Li_2O flow.

The tritium solubility in the Li_2O is rather low. The solubility reduces as the temperature of the Li_2O increases. Therefore, for a system like APEX, the total tritium inventory in the blanket is calculated to be less than 10 g.

Many issues remain to be resolved: The more critical issues are:

1. Cooling of the solid baffle.
2. Impact of oxygen contamination to the plasma.
3. Material erosion and attrition issues.
4. Solid material transport.
5. Solid to gas heat exchanger design with the solid in vacuum.

18.6 Evaporation Cooling Blanket Concept and the EVOLVE Design (Chapter 10)

It has been already mentioned that the work on the EVOLVE concept is at a very early stage. The main emphasis in the program is to assess the potential of the concept, identify crucial issues which may became killing issues, and to define most needed R&D work to remove the important question marks. No detailed design work, comprehensive analyses, or concept optimization is required at this stage. In this spirit, the following main questions should be addressed:

1 Will the backside of the first wall remain wetted under all conditions?

The liquid metal is supplied to the heated surface by an array of jets. Vapor will be generated in the liquid metal film and has to escape from the surface. Surface tension has to keep the wall wetted. This is the same principle as successfully employed in heat pipes where in addition the liquid metal has to be transported over a relatively large distance by capillary forces. This "pumping" is in heat pipes facilitated by grooves, porous walls or a suitable wick structure. The conditions with actively pumped arrays of jets are probably even more favorable for high heat flux removal but the required surface conditions and the heat flux limits of the EVOLVE concept have to be investigated.

2 Will the vapor generated in the stagnant boiling pools of the primary breeding region separate fast enough from the liquid metal?

The trays in the breeding zone contain a lithium pool with a height of 10 to 20 cm. The bottom of this pool is made of tungsten with about 5 mm thickness. The peak volumetric heat generation in the tungsten structure is about 100 W/cm3, in the lithium pool about 10 W/cm3. This results in a rather high rate of lithium vapor generation. The important question is, how the vapor will rise inside the pool to the surface and how it will separate there from the liquid metal. If this boiling process is similar to boiling water, the liquid metal density will remain reasonably high. However, if the lithium boils like milk, the liquid metal density could become too low for sufficient tritium breeding and shielding. The problem becomes even more complicated by the presence of the strong magnetic field, damping all fast liquid metal movements.

It has been suggested, that under these conditions discrete sites for initiating bubbles at the bottom of the tray should be triggered in order to generate a desired pattern of vapor flow channels with increasing diameters from the bottom to the surface. This mechanism is supported by the larger heat generation in the tungsten bottom plate. No liquid metal movement would be necessary for this pattern since the vapor channels are stationary. This model, however, is an assumption at the moment and suitable modeling and possible experiments are required to prove the feasibility of the vapor separation.

3 Will the liquid metal overflow system work and lead to equal liquid metal pressure in each tray?

The stack of trays in each blanket segment is about 3 m high. There is a supply tube located at the top tray and a drain tube at the bottom tray to extract excess liquid metal. The trays are all connected with standpipes (similar to the ones commonly used in showers) which maintain a constant level in each tray and supply the liquid metal to the tray below. The trays are connected directly in vertical direction by the vapor manifolds only. This arrangement should result in equal liquid metal height and pressure in every tray, provided the MHD pressure drops in the standpipes con be overcome by small static heads. Using thin-walled inserts in the standpipes can reduce the MHD pressure drop but it should be investigated if this is sufficient or if MHD flow coupling has to be employed to insure the desired liquid metal flow.

4 Is it possible to fabricate entire blanket segments of tungsten or tungsten- alloys in spite of their low ductility and their limited weldability?

The EVOLVE FW/blanket design allows for large fabrication tolerances and has minimum requirements on the strength of welds. Most of the welds have even not to be leak proof. Primary and secondary stresses in the blanket structure are minimized by the design. But nevertheless, the feasibility of fabricating these segments with such a material has to be investigated. It has been suggested to facilitate the fabrication by using a combination of tungsten and tantalum alloys but again this possibility would have to be investigated.

5 How will the structural material behave under intense neutron irradiation?

There is very little experience with tungsten under neutron irradiation. The general trend is that the material becomes even more brittle already at low fluence. However, all the irradiation experiments had been performed at relatively low temperatures < 700 C and they are not relevant for temperatures > 800 C as envisaged in the entire EVOLVE blanket structure. Neutronics calculations show that the dpa-rates and the helium generation rate in tungsten are much lower than in steel or vanadium but which values are allowable with this material? Some estimates at least based on suitable models about these issues are required.

6 Will the high after heat in tungsten cause a safety problem in case of a LOCA ?

The after heat in tungsten at shutdown amounts to about 2 % of the full power value and is considerably higher than the one in steel. Without any cooling in case of a LOCA , temperatures > 1500 C could be reached for longer periods of time. These estimates are, however, based on over-conservative assumptions for the following reasons :

a) The front zone including FW and primary breeding zone is cooled by a loop separated from the cooling loop for secondary breeding zone/shield. If one of the two systems remain operational, the maximum tungsten temperature will probably fall much below 1500 C.
b) The FW/blankets in each torus sector have their individual cooling system. The after heat in the failed sector can be radiated at relatively low temperatures to the other sectors.
c) If the cooling of the vacuum vessel remains operational, the EVOLVE FW/blanket can probably radiate away the afterheat to the vacuum vessel at maximum temperatures below 1500 C.

More analyses are required to find out if there is a significant safety problem caused by a LOCA in the EVOLVE system.

18.7 Helium-Cooled Refractory Ally FW and Blanket Concept (Chapter 11)

The preliminary evaluation of a helium-cooled refractory alloy first wall and blanket design has been completed. Many development issues are identified in different areas. The following is a preliminary list of key issues, grouped by areas that will have to be addressed in order to make it to become a viable design.

- Materials:
  - — Irradiated and engineering design material properties of W-alloy
  - — Design criteria for W-alloy
  - — Fabrication of W-alloy components
  - — Minimum cost of W-alloy components including material and fabrication
  - — Compatibility between helium impurities and W-alloy
- Design:
  - — External coolant piping routing
  - — Structure support to handle thermal expansion
  - — High temperature piping
  - — Develop robust high performance fusion power core W-alloy components

• Thermalhydraulics:	— Helium flow control, distribution and stability
	— First wall and blanket temperature management and startup
• Nuclear analysis:	— 3–D assessment
• Safety:	— Removal of afterheat during LOCA and LOFA
• Plasma and surface interaction:	— W-surface compatibility with high performance plasma
• Tritium:	— Extraction, inventory and PCS contamination

In addition, the availability of fusion power core components will have to be demonstrated.

18.8 Plasma-Interface Issues and Edge Modeling (Chapter 12, Summary and Issues)

A crucial issue for the use of liquid walls in fusion systems is their impact on the performance of the fusing plasma core. The thin layer of edge plasma provides the interface between the hot-plasma core and the liquid first-walls and divertor plates. The edge-plasma properties must be accurately determined to predict the coupling between the core plasma and the wall, and the edge-plasma itself is affected by both the core plasma and the wall.

The liquid surfaces can impact the edge and core plasmas by releasing impurities through sputtering, recycling, and evaporation. Such impurities degrade fusion core performance through enhanced radiation loss and fuel dilution. The tolerable levels of core impurity concentration owing to radiative energy loss and to fuel dilution are shown in Fig. 12.1 for a Tokamak. Changes in the edge plasma temperature and gradient scale-lengths can also affect the stability of the core-edge plasma, *e.g.*, the L-H transitions, ELMs, and possibly disruptions.

The edge plasma, in turn, influences the liquid surfaces through particle bombardment and line radiation from excited ions. The bombardment leads to sputtering and recycling, and both bombardment and radiation heat the surface that results in increased evaporation. The maximum tolerable evaporation rate specifies the maximum allowable surface temperature of the liquid and the sputtering analysis specifies the required edge-plasma properties through the plasma-induced particle flux to the walls.

A multi-faceted, self-consistent model is required to make a complete evaluation of these interactions between the edge-plasma and the liquid walls. We have made substantial progress in developing components of this general model and in using these components for initial evaluation of some of the critical issues (see Chapter 12)

In the coming months, a self-consistent sputtering erosion/redeposition analysis of a lithium divertor surface is planned, using coupled UEDGE/WBC/VFTRIM (plasma SOL fluid code/Monte Carlo kinetic impurity code/vectorized fractal-TRIM sputtering code) codes. This will better compute plasma contamination potential, tritium codeposition, and self-sputtering runaway potential.

Another important question is the response of a liquid divertor plate to a Tokamak disruption. A number of physical processes have been included in the HEIGHTS package and simulations performed for a liquid lithium plate. The incoming power to the plate is taken as 100 GW/m^2 which is typical of what would be expected in a reactor-sized Tokamak. As this high particle energy strikes the plate, material is ablated in the form of a gas vapor, which is subsequently ionized by the incoming electrons. The energy required to ionization of the vapor can decrease the incoming energy to the plate by an order of magnitude to less than 10 GW/m^2 while this partially ionized vapor cloud becomes optically thick. An additional reduction of the power to the plate comes from the splashing of plate material into droplets due to Kelvin-Helmholtz or Rayleigh-Taylor instabilities in the vapor. The power loss in vaporizing these droplets can result in another factor of 5 reduction in power reaching the plate. The mass loss of the liquid lithium plate can likewise be reduced by about two orders of magnitude from the combined shielding of the vapor and the droplets from slashing. As a result, the effect of a disruption on the lithium plate is not thought to be limiting.

Further assessment is needed to determine how the incoming disruption power, which is initially absorbed by the vapor and droplets but then re-radiated, effects nearby structures. Also, the vapor and splashing that result from the disruption will migrate to other surfaces in the machine. If all surfaces are moving liquids, they will self-clean, but using the same liquid for the plate and the walls will eliminate the problem altogether.

The impact of different edge-plasma conditions on the performance of the fusing core plasma is being studied with the 1 1/2-D core transport code ONETWO that has been used extensively for analyzing DIII-D experimental results. As an initial case, an ITER-like Tokamak is being considered with a 20 keV operating point since a lot of previous analysis has been done on this configuration which provides a good simulation benchmark. The effect of the low-recycling edge conditions using lithium plates will be contrasted with the normal high-recycling edge (which would likely arise if Flibe were used). Given this background, a similar analysis will be done for the ARIES-RS design.

Finally, it is important to benchmark models of how liquid surfaces emit impurities in the presence of plasma discharges, and how the impurities transport in the plasma. At present, small samples of lithium and gallium have been used in the linear plasma device PISCES, and lithium has just been used on the DiMES probe for the DIII-D Tokamak. Sputtering data is also available from particle beam measures on the Univ. of Ill. experiment. The sputtering data from these various experiments are being tabulated and will be used as input for the fluid and Monte Carlo codes which follow the subsequent ionization and transport of the impurity ions. A challenge impurity transport modeling for the DiMES probe is that the probe is localized to one toroidal location, so 3-D effects do enter which can only be estimated by the present codes. Nevertheless, these calculations will begin to force reality checks on the models. Larger-scale samples in experiments will improve this benchmarking. There are discussions to use liquid divertor surfaces in other devices such as CDX-U. This type of activity is important to provide the experimental database to validate models predicting the influence of such walls in fusion-related devices, and close collaboration will be maintained.

18.9 Materials Considerations and Data Base (Chapter 13, Summary and Issues)

The determination of minimum and maximum allowable temperature limits for structural materials requires consideration of several processes. In BCC alloys, the minimum operating temperature limit will likely be determined by radiation hardening and embrittlement issues. The minimum temperature limit for SiC/SiC composites will likely be determined by thermal conductivity degradation effects (the amount of thermal conductivity degradation in SiC is particularly pronounced at lower irradiation temperatures). The upper temperature limit for BCC alloys will typically be determined by either thermal creep, helium embrittlement, or chemical compatibility issues. The upper temperature limit for SiC/SiC will likely be determined by either void swelling or chemical compatibility issues (helium embrittlement and thermal creep would be expected to become pronounced at higher temperatures than the void swelling limit is SiC, which is estimated to occur at ~1000°C).

Figure 18.1 summarizes the operating temperature windows (based on thermal creep and radiation damage considerations) for some of the structural materials considered for APEX. Additional temperature restrictions associated with coolant compatibility issues are summarized in section 13.4. The lower temperature limits in Fig. 18.1 for the refractory alloys and ferritic/martensitic steel are based on fracture toughness embrittlement associated with low temperature neutron irradiation. An arbitrary fracture toughness limit of ~30 MPa-m$^{1/2}$ was used as the criterion for radiation embrittlement. Further work is needed to determine the minimum operating temperature limit for oxide dispersion strengthened (ODS) ferritic steel. The assumed value of 300°C used in Fig. 18.1 for ODS ferritic steel was based on results for HT-9 (Fe-12Cr ferritic steel). The minimum operating temperature for SiC/SiC is based on radiation-induced thermal conductivity degradation, whereas the minimum temperature limit for CuNiBe was simply chosen to be near room temperature. Low temperature radiation embrittlement is not sufficiently severe to preclude using copper alloys near room temperature, although there will be a significant reduction in strain hardening capacity as measured by the uniform elongation in a tensile test. The high temperature limit was based on thermal creep for all of the materials except SiC and CuNiBe. A Stage II creep deformation limit of 1% in 1000 h (3×10^{-9} s^{-1} steady-state creep rate) for an applied stress of 150 MPa was used as an arbitrary criterion for determining the upper temperature limit associated with thermal creep. Further creep data are needed to establish the temperature limits for longer times and lower stresses in several of the candidate materials. Helium embrittlement of grain boundaries may cause a further reduction in the upper temperature limit, but sufficient data under fusion-relevant conditions are not available for any of the candidate materials. The high temperature limit for SiC was determined by void swelling considerations and the limit for CuNiBe was associated with its low unirradiated fracture toughness at elevated temperatures.

With the exception of CuNiBe, the temperature windows summarized in Fig. 18.1 are sufficiently wide (?T=300-400°C) to enable attractive blanket systems to be designed. The specific values of the operating temperatures need to be combined with compatibility data for the candidate coolants (cf. section 13.4) to determine if the temperature window is reduced due to corrosion issues. One disadvantage with the high minimum operating temperatures of the Ta, Mo and W alloys is that they may require the use of high-performance, high-cost materials (e.g., Ni-based superalloys) in the power conversion piping external to the reactor.

Additional important issues which have not yet been fully considered in the selection of the structural materials for APEX include transmutation effects (long term activation and burnup of alloy elements), afterheat/safety issues (including volatization), and availability/proven resources.

The list of recommended materials R&D activities for the next 1-3 years includes:

1. chemical compatibility of coolants and structural materials at high temperatures (oxygen, Flibe, Sn-Li, including oxidation resistant alloys such as Mo-Ti-Si, intermetallics);
2. measurement of thermophysical properties for Sn-20Li
3. effect of coolant velocity on the erosion of structural materials (determination of the upper limit for coolant velocity)
4. unirradiated and irradiated fracture toughness of refractory metals at APEX-relevant (high) temperatures;
5. joining methods for refractory and oxide dispersion-strengthened alloys (stir friction welding, etc.)

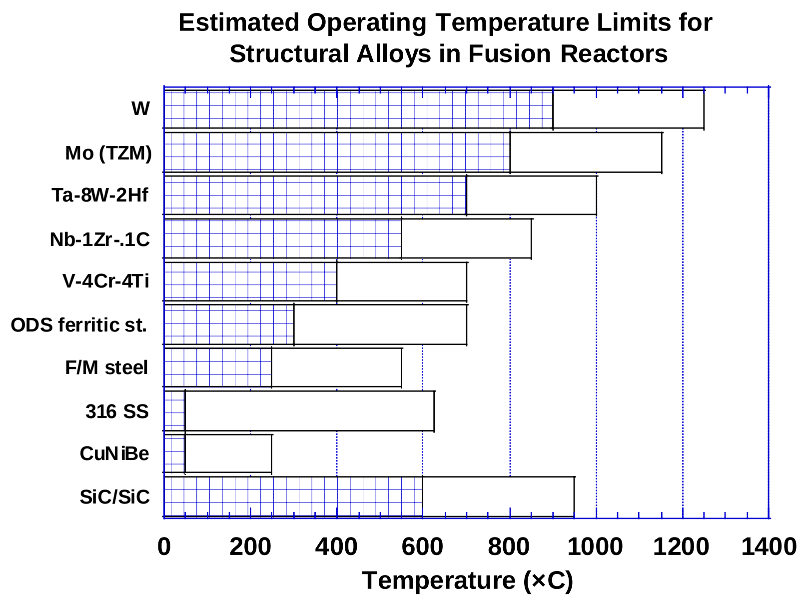

Fig. 18.1. Estimated operating temperature limits for structural alloys in fusion reactors based on thermal creep and irradiation considerations (see text). Chemical compatibility issues may cause a further restriction in the operating temperature window.

18.10 Safety and Environment Considerations and Analysis (Chapter 14, Summary and Issues)

Safety and environmental issues are being considered up front in the APEX project as designs evolve so that the goal of safety and environmental attractiveness is realized. Designing safety into the concepts as was done in the ITER project results in less complex systems than retrofitting the design to meet safety requirements.

The designs under development in the APEX project are at a pre-conceptual stage, lacking the detail needed for a comprehensive safety analysis. However based on safety screening criteria, we look for safety issues that could be "show-stoppers,", i.e., meeting safety guidelines does not look feasible. In particular, our initial focus has been on the ability of the designs to remove decay heat. The goal here is to ensure that temperatures remain below levels at which oxidation-driven mobilization becomes unacceptable. We have done some parametric studies to identify potential safety concerns, and improve designs to meet safety requirements. We have examined a number of concepts to determine the ability of the design to remove heat from the plasma-facing surface during an accident. If surface temperatures are low enough, mobilization of hazardous material is minimized.). The CHEMCON code used in these calculations was developed to analyze decay heat driven thermal transients in fusion reactors.

These preliminary scoping calculations are by no means sufficient for determining whether these designs will meet safety guidelines. They are meant as a starting point, and are used to make recommendations to designers so that safety is "built into" designs as they mature. As more design detail becomes available, further safety analyses will be done to ensure that safety requirements are met.

18.10.1 Reliability Issues

As part of this study, comparisons between the traditional solid wall plasma facing surface Tokamak designs and the self-renewing liquid wall Tokamak designs are undertaken to determine what features are attractive in each design approach. The maintenance times for in-vessel component replacements in solid wall Tokamak designs are an important feature since these extended downtimes can affect the operational availability of a power plant. By surveying reported remote maintenance times for existing experiments, we can estimate times for next-generation experiments.

Both operational experience at JET with divertor replacement and best estimates of remote maintenance times for solid wall replacements for ITER show times on the order of 26 weeks. While future innovations may refine these times, they are still lengthy. Downtimes of that duration can be used to advantage to perform other time-consuming activities, such as in-service inspections of piping or heat exchangers, cleaning heat exchangers, etc. Nonetheless, these downtimes for remote replacements will put limitations on reactor operational availability. Solid wall fusion reactor availability may experience growth similar to the early fission reactors, as the understanding of how to build a robust first wall and divertor increases. Then wall reliability would increase so that wall replacements are less frequent. Further advances in materials and their hardening to withstand radiation damage would allow longer-lived solid wall blanket/first wall designs. Some current estimates of the availability for the EU DEMO reactor blanket/first wall designs show availabilities in the range of 84.3% to 87.7%. These values are lower than required to obtain a reactor that competes with nuclear fission (i.e., a reactor plant that has 87% or more plant availability), as discussed by M. Abdou in the APEX design study. Radiation damage to the liquid wall components must be evaluated. The liquid walls would also need some periodic maintenance, such as the possibility of flow nozzle or guide vane changeouts. Like solid walls, the liquid wall system would also require visual inspections, such as the vessel walls, guide vanes, screens, and other passive items. Weld testing would be needed in both solid and liquid wall designs. Remote changeouts of the liquid wall passive components (nozzles, vanes, screens, etc.) are expected to be less complicated than changeouts of the large, heavy blanket/first wall modules used in solid wall designs.

As the APEX design progresses, it should be possible to calculate a scoping availability value for the liquid wall system. Then comparisons can be made to the solid wall availability values from the literature.

18.10.2 Waste Disposal Issues

Materials choice has long been recognized as a key factor in realizing the full safety and environmental potential of fusion power. Because the materials are de-coupled from the fusion energy source (the plasma), the long-term neutron-induced activation of components can be tailored by proper selection of materials to avoid generation of waste that would require deep geological disposal. Thus, the idea of "low activation" materials was conceived for the US fusion program with the hope that such material could be disposed of as low level waste (e.g., shallow land burial) and would not pose a burden to future generations.

The environmental impact of waste material is, however, determined not only by the level of activation, but also the total volume of active material. A Tokamak power plant is large, and there is a potential to generate a correspondingly large volume of activated material. The adoption of low activation materials, while important to reduce the radiotoxicity of the most active components, should be done as part of a strategy that also minimizes the volume of waste material that might be categorized as radioactive, even if low level. Waste management strategies have typically concentrated on minimizing the activity of first wall and blanket components where the level of specific activity (Bq/kg) is highest.

Some materials may become candidates for recycling, and others may be cleared from regulatory control by meeting prescribed criteria that have yet to be agreed upon internationally. Recently these concepts of recycling or clearance have been recognized as options for reducing the volume of radioactive waste from a fusion power plant. Determining if a material can be recycled or cleared from regulatory control depends largely on our ability to limit the induced activation of the component. (It should be noted that the criteria for clearance are more restrictive than for recycling.) Thus, there is a need to explore new and innovative concepts that can substantially reduce the activation of the large ex-vessel components that contribute significantly to the overall volume of activated material and to extend the capability of conventional conceptual fusion designs with proper optimization to achieve the same goal. The impact of these parameters on other aspects of plant performance must also be considered.

18.10.3 Summary/Future Direction

By ensuring that safety and environmental issues are considered early in the APEX design process, we are producing designs that help show the safety and environmental potential of fusion energy. As designs mature we will continue the safety analyses to guide designers toward better designs. Similarly, waste management issues will be considered, working towards clearing or recycling large volume components.

Reliability is a very important aspect of the APEX project, and an area in which liquid surface designs may have an advantage over solid wall designs. Nozzle reliability will be an important part of this, and is an area that will be studied in the APEX project.

Build a robust first wall and divertor, then wall reliability would increase so that wall replacements are less frequent. Further advances in materials and their hardening to withstand radiation damage would allow longer-lived solid

18.11 Synergy between Liquid Walls, Tokamak Physics Performance, and Reactor Attractiveness (Chapter 15, Summary and Issues)

It is well recognized that the presence of a conducting shell close to plasma improves the MHD stability. Liquid metal wall concepts being considered by APEX place a liquid metal quite close to the plasma. Investigations are being performed to determine the potential physics benefits of using the liquid wall as the conducting shell, and the engineering feasibility of this approach. Analysis to date indicates that the potential physics benefits are quite large. For example, reactor designs such as ARIES have placed the stabilizing shell for the vertical instabilities behind the blanket and shield, limiting the stable elongation to approximately 2, with a beta limit of about 5-6%. However, the stable elongation could be increased to above 3 by using ~2 cm of lithium at the first wall of the plasma, increasing the beta limit for reactor relevant cases to ~ 20% (for the same aspect ratio as ARIES). Though a full systems analysis has not been performed, this could potentially allow a large reduction in the size of a reactor (e.g., from 5.5 meter major radius to ~ 3 meter). In addition, the confinement appears to be significantly improved allowing ignition in such small systems.

These calculations indicate large potential benefits, but are incomplete in several respects. This will be addressed in the next phase of APEX. Published results are indicative that a shell close to the plasma can stabilize ideal kink modes for highly elongated geometries, but this will be verified. The vertical stability calculations and feedback control requirements will also be refined. Several potential strategies to stabilize the resistive wall kink mode appear conceptually compatible with liquid metal walls, but these require quantitative assessment. The liquid nature of the stabilizing shell must be included in the calculations (adding greatly to the complexity of the analysis, but preliminary indications do not reveal any show stopping difficulties to the concept itself). The simultaneous use of a liquid first wall as a stabilizing shell as well as a heat removal system presents considerable engineering challenges. However there are multiple potential solutions, and each must be assessed. In addition, it may turn out that design possibilities need to be considered which separate these functions, e.g., use of a liquid metal slightly behind the first wall solely as a stabilizing element.

18.12 Tritium (Chapter 16, Summary and Issues)

The tritium recovery systems for different breeding materials have been reviewed and the most promising concept for each breeding material has been recommended.
The recovery method has to limit the tritium inventory below the design limit imposed by safety consideration. ITER set the maximum allowable releasable tritium inventory in each component to less than 200 g. Also, the tritium recovery system, together with the system design, has to limit the tritium leakage rate to less than 10 Ci/d. The allowable tritium

inventory goal can be reached for most of the breeding materials. However, the goal to limit tritium leakage rate to lower than 10 Ci/d is much more difficult to reach, especially for high temperature systems such as the APEX options.

Many tritium recovery systems from lithium have been proposed. The most attractive one is the one based on cold trap(1). Cold trap has been demonstrated to recover tritium from Li, Na and NaK. The tritium solubility in lithium is 440 appm at the cold trap temperature of 200C, which is far above the design goal of 1 appm. This design concept is to add protium in the lithium to increase the total hydrogen concentration. The cold trap will reduce the total hydrogen concentration to 440 appm, while the 1 appm tritium concentration can be reached.

The tritium solubility in flibe is very low. A very efficient tritium recovery system will be required to reduce the tritium permeation rate to below 10 Ci/d. A two stage vacuum disengager was proposed by the HYLIFE design(2). This design forces hot flibe into small droplets and letting them fall through a vacuum chamber. The tritium will diffuse out of the flibe and be removed. The calculated tritium recovery efficiency is 99.7% for each stage, resulting in a combined efficiency of 99.999%. There has been no experimental verification of this process.

Many tritium recovery processes have been proposed to recover tritium from LiPb. The most attractive one is based on permeation to NaK and a cold trap to recover tritium from the NaK(3). The process involves the following three steps: (1) Tritium permeation into NaK gap of a double walled steam generator, (2) Tritium recovery from NaK by precipitation as potassium tritide in a cold trap, and (3) Tritium recovery by thermal decomposition of the tritide. The entire process has been demonstrated in a laboratory scale experiment.

There is no theoretical or experimental work on tritium recovery from Sn-Li. The tritium solubility in Sn-Li is not available.

References for Section 18.12

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18.13 Power Conversion (Chapter 17, Summary and Issues)

The parameter range and efficiency of different power conversion systems have been reviewed for the APEX applications. The power conversion systems reviewed are steam cycle, the close cycle gas turbine, and the binary cycle. Below are some of the summaries from this review:

1. For steam cycle, advanced cycle efficiency can reach 50%, based on super critical steam cycle. Multiple reheat will be required to increase the thermal efficiency. A more realistic, and maybe more reliable system will be based on double reheat, with steam and reheat temperature near 600C. A thermal efficiency of 47% can be achieved.

2. For more conventional design, a closed cycle gas turbine has to have an operating temperature of ~ 850C, to reach high cycle efficiency. However, recent proposal suggested that 46% thermal efficiency can be achieved with gas temperature as low as 650C. To achieve this high efficiency at a modest temperature, the efficiencies of the compressor, turbine, and especially the recuperator have to be between 92 to 96%. Also, the coolant pressure has to be high, to reduce the circulation power. The efficiency of the recuperator, assumed to be 96%, is particularly important. While there is no issue that a recuperator heat exchanger can be designed to 96%, the reliability of this heat exchanger has to be demonstrated due to the very large number of coolant channels.

3. Binary cycle: While binary cycle is attractive for open cycle design to convert the thermal energy in the low temperature regime, there is no reason to adopt a binary cycle for a closed cycle design, as the ones will be used by fusion. It is always more efficient to design a regenerator heat exchanger to recover the thermal energy, and convert the thermal energy to electricity at a higher temperature.