Fig. 20. Thermal efficiency as a function of steam temperature in various systems. Power conversion efficiency determines material choice and bulk exit temperature.

Fig. 21. The DPA rate in the backing solid wall.

Fig. 22. The helium production rate in the backing solid wall.
ture compatible with the plasma operations, but also to maintain a mean bulk temperature of greater than 600°C for high thermal efficiency (see Fig. 20). This temperature can be higher than the maximum allowable free surface temperature. One approach to overcome this difficulty in a thick liquid wall design is to use two different coolant streams: one for surface heat removal and the other for neutronics heat deposition in order to simultaneously achieve these two conflicting temperature requirements. A power conversion system would then include two cycles: one for the conversion of the thermal power fast plasma-facing stream, and the other for conversion of the thermal power in the thick liquid behind it, which has a much higher thermal conversion potential. Since thick liquid walls will reduce both the neutron damage rate and helium transmutation rate, the choice of the structural material should be determined by the high temperature capability and liquid/structure compatibility. It appears that the use of tungsten alloys would achieve the highest thermal efficiency because of its high temperature operation capability. The oxide-dispersed ferritic steel (ODFS) can operate up to a temperature of 650°C, which provides a thermal efficiency of about 41.2%.

4.6. Effects of liquid walls on reducing activation and radiation damage

Reducing activation and radiation damage to structural materials are among the important advantages of liquid walls, particularly the ‘thick’ liquid wall concepts. The magnitude of these advantages is design dependent. Calculations were performed to quantify the benefits as a function of key design parameters. The results are briefly summarized in this subsection. The results were utilized to guide the choices for concept exploration discussed earlier in this section. More detailed analysis is required in the future to address flow support structures other than the backing wall that may be also needed in liquid wall designs (for example, inlet nozzles and flow dividers). The thickness of the liquid in front of these elements may be less than that protecting the backing wall. Note also that these elements are more accessible than the backing wall (or first wall in traditional concepts) and therefore faster maintenance may be possible. In particular, the key factor is the extent to which liquid walls attenuate the neutrons before they reach the structural materials. The main structural element in liquid wall designs is the backing wall. So, the thickness of the liquid wall is important in determining the reduction in activation and radiation damage in the back wall relative to the solid first wall in traditional concepts.

4.6.1. Radiation damage parameters

The effects of the liquid layer thickness on radiation damage parameters such as atomic displacement and helium production rates were studied for lithium, Flibe, Sn–Li, and Pb–Li. Figs. 21 and 22 show the rates (per full power year, FPY) of atomic displacement (DPA) and helium production (appm), respectively, in the back wall structural material as a function of the liquid layer thickness (L) protecting the back wall. Without the liquid layer, which corresponds to a ‘bare wall’ case, the DPA and helium production rates in the backing solid wall are comparable in the four breeders. However, because Pb–Li exhibits larger reflection, the low-energy neutron flux is larger at the solid wall which results in larger DPA rate (occurs at all energies). This also gives smaller He-4/DPA ratio in the case of Pb–Li breeder ( ~ 8.7) as compared to the value with the other breeders (10–11).

As the thickness of the liquid layer increases, the reduction in these damage parameters varies among the four breeders. Lithium is the weakest material in moderating neutrons as compared to the other breeders. The reduction in DPA rate is less than an order of magnitude for \( L = 42 \) cm, while the reduction in helium and hydrogen production is about an order of magnitude. The attenuation characteristic of the Pb–Li breeder for the DPA rate is similar to lithium. However, the Pb–Li is superior to the other breeders in attenuating the helium and hydrogen production rate in the solid wall. This is due to its larger attenuation power to high-energy neutrons (through \((n, 2n)\) and \((n, \text{inelastic})\) reactions) which is basically the main contributor to the high-
threshold helium and hydrogen reactions in the solid wall. Because of the smallest He-4 production and the largest DPA rate with the Pb–Li breeder, the ratio He-4/DPA is the smallest (≈ 0.3) at L = 42 cm as compared to the values with the other breeders (Li: ≈ 7, Flibe: ≈ 6, Sn–Li: 2). The attenuation characteristics of the Flibe and Sn–Li are similar for the helium and hydrogen production. However, the Flibe gives the best attenuation to the DPA rate since it is capable of attenuating both the high- and low-energy component of the neutrons reaching the backing solid wall.

Using the damage values at the bare wall (L = 0 cm) and at the wall with various L thicknesses, one can estimate the tenfold thickness, L_{10}, for each breeder defined as the required thickness of the layer to reduce a particular response, R (damage parameter), by an order of magnitude. This thickness is given in Table 7 for the various damage parameters and breeders. For helium and hydrogen production rate, ≈ 22 cm is required to achieve an order of magnitude reduction with Flibe and Sn–Li and smaller thickness (≈ 18 cm) is required in the Pb–Li liquid layer. Twice as much thickness is required in the Li case because of its poor attenuation characteristics for helium and hydrogen production. As for the DPA rate, larger thickness is required. It is ≈ 26 and ≈ 36 cm for the Flibe and Sn–Li, respectively, but much larger thickness (≈ 58 cm) is required in the Li and Pb–Li to reduce the DPA rate by an order of magnitude.

The DPA rate in backing solid wall is ≈ 26 DPA/FPY, 3.6 DPA/FPY, 9.5 DPA/FPY, and 30 DPA/FPY, with the Li, Flibe, Sn–Li, and Pb–Li liquid layer, respectively. If the 200 DPA is considered as the limit at which the wall and shield zone require replacement, the lifetime of these components would be 7.7, 56, 21, and 7 yr, respectively. Clearly the presence of the liquid layer made these components last the lifetime of the plant (30 years) when Flibe is used as the breeder. In the case of Sn–Li, one replacement may be required after ≈ 20 years. But three to four replacements may be needed in the case of Li and Pb–Li breeders.

4.6.2. Hazard and volume of radioactive waste

Another clear advantage of deploying a thick liquid wall/blanket concept is the substantial reduction in the hazard and volume of the waste generated from the activation of solid materials (including solid walls, vacuum vessels, shield and magnets themselves). It has been found [1] in comparing the liquid FW/blanket (42 cm-thick) to a conventional blanket (2 cm-thick ferritic steel, FS, FW — 40 cm-thick blanket made of 10% FS and 90% Flibe), while keeping the radial build the same (ARIES-RS radial build and materials are assumed), the specific activity (curies/cc) at shutdown in the bare FW of the conventional blanket is two orders of magnitude higher than the specific activity in the back wall of the liquid FW/blanket case. The specific biological hazard potential (BHP) has the same features. The two orders of magnitude difference continues during the first year and starts to narrow down after the first year following shutdown. The next step is to find out how this may translate into advantages from both the waste generation (dominated by long-lived nuclides) and safety hazard (dominated by short-lived and intermediate-lived nuclides) viewpoints.

An analysis comparing the waste disposal ratings and volume of waste generated in a power plant based on the two concepts was conducted.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Li/FS</th>
<th>Flibe/FS</th>
<th>Sn–Li/FS</th>
<th>Pb–Li/FS</th>
</tr>
</thead>
<tbody>
<tr>
<td>DPA (dpa/FPY)</td>
<td>~58</td>
<td>~26</td>
<td>~36</td>
<td>~56</td>
</tr>
<tr>
<td>Helium production (appm/FPY)</td>
<td>~46</td>
<td>~22</td>
<td>~21</td>
<td>~18</td>
</tr>
<tr>
<td>Hydrogen production (appm/FPY)</td>
<td>~44</td>
<td>~22</td>
<td>~22</td>
<td>~19</td>
</tr>
</tbody>
</table>

a The thickness required to reduce a response by an order of magnitude.
Table 8
Comparison of class C waste disposal ratings using Fetter limits

<table>
<thead>
<tr>
<th>Zone</th>
<th>FPY</th>
<th>Liquid blanket concept</th>
<th>Conventional blanket concept</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inboard FW and blanket</td>
<td>3</td>
<td>–</td>
<td>1.37</td>
</tr>
<tr>
<td>Inboard shield</td>
<td>30</td>
<td>0.81</td>
<td>0.73</td>
</tr>
<tr>
<td>Inboard VV</td>
<td>30</td>
<td>0.141</td>
<td>0.1</td>
</tr>
<tr>
<td>Outboard FW and blanket</td>
<td>3</td>
<td>–</td>
<td>1.34</td>
</tr>
<tr>
<td>Outboard shield</td>
<td>30</td>
<td>0.795</td>
<td>0.71</td>
</tr>
<tr>
<td>Outboard VV</td>
<td>30</td>
<td>0.087</td>
<td>0.06</td>
</tr>
</tbody>
</table>

The waste disposal ratings for the Fetter [15] and 10CFR61 [16] limits are shown in Tables 8 and 9, respectively. Results in the tables are given for compacted wastes after 1 year following shutdown. As shown in Table 8, according to Fetter limits, all components of the liquid blanket concept would qualify for disposal as class C waste after 30 FPY. All components of the conventional blanket concept, except for the first wall and blanket, also would qualify for disposal as class C waste after 30 FPY. The first wall and blanket would not qualify for disposal as class C LLW unless they were replaced every 2 FPY instead of every 3 FPY. On the other hand, the 10% steel structure in the conventional blanket provided the shield and vacuum vessel behind it with better shielding, resulting in lower waste disposal ratings in comparison to the waste disposal ratings of the shield and vacuum vessel behind the liquid blanket. Results in Table 9 show that, according to the 10CFR61 limits, all components of both blanket concepts would qualify for disposal as class C waste. The absence of contribution from $^{192}\text{Ir}$ to the waste disposal ratings according to the 10CFR61 limits (10CFR61 has no limits for $^{192}\text{Ir}$) resulted in allowing for the disposal of the first wall and blanket of the conventional blanket concept as LLW after 3 FPY.

A power plant based on the conventional blanket concept will produce the equivalent of about ten blankets of additional waste during its lifetime. However, a power plant based on either the liquid or conventional blanket concepts will generate a comparable amount of waste from the shield, vacuum vessel, and magnets, whose volumes far exceed the volume of the blanket. As shown in Fig. 23, the volume of the waste generated during the lifetime of a power plant (30 FPY) based on the liquid blanket concept could be a factor of six lower than the volume of waste generated during the same lifetime if the plant was based on the conventional blanket concept. The factor of six is based on the assumption that the waste is non-compacted and the waste does not include the magnets. If the waste is compacted to 100% of its theoretical density, the reduction factor drops from six to two. If the waste is compacted and the magnet waste is included, a power plant based on the conventional blanket concept will generate about 35% more waste during its lifetime (30 FPY) than a similar power plant based on the liquid blanket concept.

4.7. Key issues and R&D

The present state of understanding of thick liquid wall concepts does not reveal any basic flaws in the underlying scientific and technical arguments for the concepts. Yet, there remain many issues for the implementation of this concept in any magnetic fusion configuration. Near term R&D should focus on continued concept exploration as well as modeling and experiments for key feasibility issues. These include:
1. Edge-plasma and core-plasma modeling and analysis as well as experimental research in various confinement devices for plasma–liquid wall interactions.
2. Experimental data on the achievable minimum liquid surface temperatures without MHD effects for turbulent Flibe and MHD laminarized 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 liquid metal flow hydrodynamics configurations applicable to MFE confinement schemes such as 1/R toroidal field variation, and effects of finite radial, poloidal, and vertical field components.

5. Computer simulation of MFE relevant three-dimensional free surface liquid wall thermal and hydrodynamics performance with MHD effects. In particular, hydrodynamics characteristics near the penetrations and supply and return lines.

6. Identification of the most promising hydrodynamics configurations with respect to different MFE confinement schemes.

In addition, engineering innovations and analyses are required for the following numerous mechanical design issues including:

- How to move mass quantities of liquid metal or salt in and out of the machine reliably.
- How to provide sufficient access for supply piping and return ducts.
- How to design the piping and nozzles for reliable operation at high fluid velocity.
- How to start and stop the system safely.
- How to keep the stream attached to the inboard wall (must prevent toroidal rotation of inboard stream).
- How to provide sufficient penetrations for heating and diagnostics.
- How to account for image current effects from moving plasma.

Issues related to effects of liquid metals on the plasma core and edge-plasma liquid–surface interactions are discussed in Sections 7 and 8.

5. Thin liquid wall concepts

The thin liquid wall concept was explored in APEX for liquid metals and for Flibe. Initial designs of thin liquid walls were developed and the associated advantages and disadvantages were analyzed. Thin liquid wall concepts are called Cliff.

The idea behind Cliff (the Convective Liquid Flow First-Wall concept) is to eliminate the presence of a solid FW facing the plasma through which the surface heat load must conduct. This goal is accomplished by means of a fast moving (convective), thin liquid layer flowing on the FW surface (see Fig. 24). This thin layer is easier to control than a thick liquid FW/blanket, but still provides a renewable liquid surface immune to radiation damage and sputtering concerns, and largely eliminates thermal stresses and their associated problems in the first structural wall. The attractiveness potential and key issues for the Cliff design are summarized in Table 10. The Clifff class of liquid wall concepts is viewed as a more near-term application of liquid walls.

Details of the preliminary design, heat transfer, power balance, thermal-hydraulics, neutronics, activation and safety are included in this section. It is noted that the first several centimeters of various thick liquid FW/blanket concepts discussed in the preceding section will behave in a similar fashion to the Clifff concept discussed

<table>
<thead>
<tr>
<th>Zone</th>
<th>FPY</th>
<th>Liquid blanket concept</th>
<th>Conventional blanket concept</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inboard FW and blanket</td>
<td>3</td>
<td>0.25</td>
<td>0.495</td>
</tr>
<tr>
<td>Inboard shield</td>
<td>30</td>
<td>4.22 × 10^{-3}</td>
<td>2.82 × 10^{-3}</td>
</tr>
<tr>
<td>Inboard VV</td>
<td>30</td>
<td>2.54 × 10^{-3}</td>
<td>1.69 × 10^{-3}</td>
</tr>
<tr>
<td>Outboard FW and blanket</td>
<td>3</td>
<td>0.25</td>
<td>0.473</td>
</tr>
<tr>
<td>Outboard shield</td>
<td>30</td>
<td>0.25</td>
<td>0.21</td>
</tr>
<tr>
<td>Outboard VV</td>
<td>30</td>
<td>2.54 × 10^{-3}</td>
<td>1.69 × 10^{-3}</td>
</tr>
</tbody>
</table>
here, and significant overlap with those analyses is seen in what follows.

5.1. Design description

The majority of the work reported here was carried out for the tokamak. Specifically, the ARIES-RS geometry was utilized whenever possible, with modifications for the unique structures and high flowrates required for CLiFF. This means, however, that the ARIES-RS fusion power needs to be scaled-up to 4500 MW to give the 10 MW/m² peak neutron wall load and 2 MW/m² peak surface heat flux goals of the APEX study. Tokamaks present a difficult challenge for liquid walls due to the fact that the plasma cham-
ber is relatively closed with short scrape-off lengths, and so, vaporized liquid wall material must be screened by the edge plasma to keep it from penetrating to the core.

The general CLiFF design, as seen in Fig. 24, is conceptually simple in its implementation. A thin fast liquid layer is injected near the top of the plasma chamber. The layer flows down the reactor walls without excessive slowing or thinning, and is removed in some fashion from the bottom of the chamber. Layer thickness \( h \) on the order of \( 0.5–2 \) cm, and velocity \( U \) on the order \( 10 \) m/s, are considered. The curved back wall fits the plasma shape and provides an adhesion force due to the liquid’s centrifugal acceleration. The criterion for continuous attachment of the liquid layer is simply \( U^2/R_c > g \cos \alpha \), where \( g \) is the acceleration of gravity, \( R_c \) is the radius of curvature of the first wall section and \( \alpha \) refers to the angle of the outward surface normal to gravity vector (so \( 0^\circ \) is completely inverted).

Table 10

<table>
<thead>
<tr>
<th>Potential</th>
<th>Issue</th>
</tr>
</thead>
<tbody>
<tr>
<td>Removal of surface heat loads (greater than 2 MW/m² possible). Local peaking and transients can be tolerated</td>
<td>Hydrodynamics and heat transfer involve complicated MHD interaction between flow, geometry, and the magnetic field: Suppression of turbulence and waves LM-MHD drag thickens flow and inhibits drainage from chamber Effects of varying fields on LM surface stability and drag</td>
</tr>
<tr>
<td>FW surface protected from sputtering erosion and possibly disruption damage</td>
<td>Evaporating liquid can pollute plasma, surface temperature limits unknown</td>
</tr>
<tr>
<td>Beneficial effects on confinement and stability from conducting shell and DT gettering effects</td>
<td>High flowrate requirement can result in low coolant ( \Delta T ) or two coolant streams</td>
</tr>
<tr>
<td>Elimination of high thermal stresses in solid FW components, having a positive impact on failure rates</td>
<td>Effect of liquid choice on edge plasma gettering, tritium through-put, and tritium breeding</td>
</tr>
<tr>
<td>Possible reduction of structure-to-breeder ratio in FW area, with breeder material facing virgin neutron flux</td>
<td>Neutron damage in structure is only slightly reduced compared to standard blankets, blanket change-out required for high power density operation</td>
</tr>
</tbody>
</table>
| Integrated divertor surface possible where CLiFF flow removes all \( \alpha \) heat | **Note:** The velocity range is chosen quite high both to ensure adhesion to the back-wall, but also to keep the exposure time to the plasma short, and thus keep the surface temperature low. If one desires an inlet temperature that is \( > 300^\circ C \) (for power conversion reasons), it turns out that it is this second restriction that is the more severe, based on the maximum surface temperature estimates provided by the preliminary plasma edge analysis. The high velocity requirements and the large coverage area result in volumetric plasma flow rates in excess of \( 10 \) m³/s compared to ARIES-RS in the \( 3 \) m³/s range.

The conceptual CLiFF design shown in Fig. 24 has an integrated droplet-type divertor. Some means (mechanical or electrical) is used to stimulate the breakup of the FW flow into a droplet screen. It is hoped that the droplet screen will have a higher heat removal capability due to the rapid rotation and internal circulation in the droplets, but this fact remains to be proven. In
addition, for LMs, the droplet screen will be electrically isolated from the main FW flow and plasma currents will not be able to close. The liquid film can be removed from the vacuum chamber by gravity drainage or by an EM pump if the working liquid is an electrical conductor.

Supply nozzles will form the desired liquid flow at the top of the reactor. These nozzles can be designed to be protected from surface heat flux by the flowing liquids.

Note also that since these nozzles are at the top of the reactor chamber, the surface heat load and nuclear heat will be lower than the peak mid-plane values. Liquid removal from the plasma chamber is accomplished through a combined vacuum pumping and liquid drain port. It is envisioned that the liquid flow itself will pump a portion of the implanted plasma particles into the pumping ducts by convection, thus aiding in impurity removal.

The working liquid should be a tritium breeding material like lithium, tin–lithium or Flibe. Thus the liquid removed from the reactor can be recirculated to the blanket as the main tritium breeder and coolant. The bulk nuclear heat is added on top of the FW: divertor heat before the liquid is sent to the power conversion system. In this manner, the FW and divertor power is converted at relatively high thermal efficiency.

Penetrations for various heating, fueling and diagnostics functions will be provided as much as possible in the lower half of the outboard FW. Flow can be guided by means of submerged grooves around the penetration, and close again downstream to form a continuous surface protection as discussed in the previous section. Cooling of the penetration structures themselves will be aided by the CLiFF flow. It is likely that for LMs, the penetrations will have to be electrically isolated from the flow by means of an insulator coating. This will be true in supply lines and nozzles as well.

Off-normal plasma events like disruption can possibly induce large currents in LM CLiFF flows and cause the layer to be splashed or torn off the wall altogether. For poorly conducting Flibe, the effect of the disruption is not as clear. It is hoped that, in any case, splashing will turn out to be an allowable response, and that the liquid wall will just be restarted following the disruption. For an all-liquid wall system, this seems a reasonable assumption, except for possible damage to antennae and sensitive diagnostics. It is hoped that ‘liquid tolerant’ antennae could be designed that could accept the occasional splashing of liquid metal, but this certainly remains to be demonstrated.

5.2. Hydrodynamic and heat transfer analysis

Aside from plasma compatibility, one of the key issues for CLiFF is related to finding a feasible hydrodynamic configuration. A significant amount of design analysis has been done so far on CLiFF in order to answer the three basic questions: How do you form it? How do you drain it? How do you maintain it? It is noted that liquid metals and Flibe behave very differently in the magnetic environment of a tokamak. The low thermal and electrical conductivity of Flibe leads to a FW flow that will still be turbulent, and have heat transport at the free surface and flow drag at the back-wall that depend heavily upon this turbulence. For LMs the converse case occurs, where it is expected that the MHD effects will dominate the drag, and the thermal conduction dominates heat transfer.

5.2.1. Turbulent Flibe flow

Several models have been applied to predicting the flow profiles for Flibe, ranging from simple hydraulic models for the steady state equilibrium flow profile, to more complex two- and three-dimensional non-steady codes for studying phenomena like surface waves and penetrations. The 1.5 D hydraulic calculations indicate that flow depth equilibria in the range of 2 cm can be achieved for Flibe flows in the 10 m/s range (see Fig. 25). A more sophisticated, low-Reynolds number $\kappa-\varepsilon$ model of turbulence was also applied to the CLiFF flow in order to study the effect of MHD turbulence on the flow profile. In comparison to the ordinary $\kappa-\varepsilon$ model, the present one was extended to the MHD case by means of additional terms in the closure equations. Due to turbulent viscous friction, the layer thickness increases rapidly over the initial flow section (see
again, Fig. 25). This is in contrast to the results presented earlier where the simple friction factor formulation predicts nearly constant flow height and velocity profiles for CLiFF. This contradictory result is cause for concern because if the layer slows down significantly, the transit time through the plasma chamber will go up, as well as the surface temperature. Attempts to benchmark the $\kappa-\varepsilon$ and friction factor against available data from the UCLA Mega-Loop Experiment [17] are inconclusive — the data splits the difference between the $\kappa-\varepsilon$ and friction factor model.

The effect of the magnetic field on the flow parameters is negligible if the Hartmann number is less than about 1000, and hence for CLiFF with $Ha = 500$, we conclude that there is no strong impact of MHD on the Flibe flow hydraulics.

Heat transfer calculations using this same model indicate that depending on surface turbulence assumptions, the temperature rise at the surface can be quite low. For a 10 m/s, 2 cm thick Flibe flow, the surface temperature rise is in the range of 30–160°C depending on whether optimistic or pessimistic assumptions are used. The effect of the magnetic field again appears to be small. When considering the thermal hydraulics, it is seen that the temperature window for Flibe is limited (see Flibe system diagram in Fig. 26), and so the surface heat transfer is critical for feasibility. There are, however, no experimental data, and this issue needs closer study and experimental validation.

The surface stability for Flibe CLiFF flows was also analyzed using a linear stability analysis technique for infinitesimal disturbances. For CLiFF, the results show that whenever the flow is adhered, it should be stable as well. The effect of finite size perturbations may alter this picture. The primary source of large disturbances comes from the turbulence of the flow itself. The fluid dynamic behavior of the first-wall flow system may be affected due to these eddy generating mechanisms including boundary layer relaxation near nozzles, Gortler-type instabilities, structural vibrations, etc.
Penetrations have also been analyzed for the Flibe case using a three-dimensional free surface code that allows the introduction of arbitrarily formed structures. The penetrations considered are elongated into ellipses in order to be more hydrodynamically streamlined. The specific case

Fig. 26. CLiFF — flow/temperature schematic-Flibe option.

Fig. 27. Influence of the wall conductance ratio on the layer thickness increase ($2b = 1$ m). Line 1 — $c_w = 0$; 2 — $c_w = 1.0 \times 10^{-6}$; 3 — $c_w = 2.0 \times 10^{-6}$ (Lithium).
considered has dimensions 20 cm wide and 90 cm long (in the flow direction). The back-wall in the vicinity of the penetration is tailored to guide the liquid around the penetration itself, and to aid in closing the liquid again downstream of the penetration. Results presented in the previous section of thick liquid walls show that such a design solution can successfully guide the flow around penetrations, but additional work and optimization is needed for their design.

5.2.2. Magnetohydrodynamics for lithium and Sn–Li flows

Mathematically these types of flows can be described by a set of Navier–Stokes equations for incompressible fluids and Maxwell’s equations for electromagnetic phenomena. The numerical tools used to analyze this system of equations are based on two-dimensional, simplified magnetohydrodynamic equations and can be performed in practice for any values of governing parameters for ducts of various geometries. This is an extreme simplification of the physics and assumes that all currents close in their own cross-sectional plane. This type of calculation is accurate for well-behaved, nearly fully developed flows with simple geometries, but ignores significant effects near field gradients and developing regions.

It is well known that the presence of electrically conducting walls can lead to larger electrical currents in the flow domain and, as a result, to a significant increase in the MHD drag effect. In the case of free surface MHD flows, this effect manifests itself in the increase of the layer thickness with the accompanying reduction in the velocity. Ideally, if the liquid layer is assumed to be completely axi-symmetric in the toroidal direction, flow along poloidal flux surfaces with no field gradients, no MHD drag will occur. This ideal case, though, is not possible in practice and we look at two variants to gauge the relative effects of the MHD. One case is the presence of fins, side-walls, or penetrations breaking up the flow toroidally, and the other is a slight deviation of the flow path from the flux surfaces resulting in a small surface-normal field component. Figs. 27 and 28 illustrate the results for these two cases for lithium, where we assume that a doubling of the initial height represents an unacceptable result. Note that in Figs. 27 and 28, the thicknesses on the vertical axes are scaled by the initial thickness.

For the case of side-walls, it was found that electrically insulated side-walls are acceptable only if they are no closer than 1 m toroidally, and that low conductivity walls like SiC (thickness = 1 cm, assumed $\sigma = 10^3 \, \Omega^{-1} \cdot m^{-1}$) are acceptable provided they are no closer than 8 m. Bare metal walls (thickness = 2 mm, $\sigma = 10^6 \, \Omega^{-1} \cdot m^{-1}$), even if very thin, can be no closer than 110 m, and so are not feasible for CLiFF. For the case of a small radial field it was found that if the back-wall is bare metal the allowable field is only $B_r < 0.1 \, T$. This value goes up to $B_r < 0.5 \, T$ if the backing wall is insulated. These calculations assume that there are insulated side-walls present at some distance to break up the toroidal electric path (but they are separated by enough distance that they do not add appreciable drag). If complete axi-symmetry is assumed, where induced currents close on themselves, the allowable radial field is $B_r < 0.015 \, T$! These calculations indicate that serious work is needed in the area of LM-MHD analysis and experiments to prove that passive flow schemes like CLiFF are possible.

Heat transfer at the surface is calculated for Li and Sn–Li using only conduction, but assuming some penetration of X-ray photons in the case of lithium. The conclusion is that at 10 m/s the temperature rise will be on the order of 150°C for
Li, and 300°C for Sn–Li. The thermal-hydraulic calculations utilizing these numbers result in a blanket outlet temperature around 650°C for the Sn–Li, but much lower for the lithium, possibly necessitating a two-stream approach, where only part of the Li flow is sent to the blanket.

The results of stability computations are in a good agreement with the linear stability analysis conclusions. Long wavelength initial disturbances grow very rapidly on the inverted surface under the effect of gravity and centrifugal acceleration and then propagate down with slowly decreasing amplitude. The growth rate and the maximum amplitude depend on the wavelength. The short waves (\( < 20 \text{ cm} \)) are suppressed rapidly by the surface tension, while the long wave disturbances (1.5–2 m) are not suppressed over the whole flow length. The most dangerous disturbances are those having the long wavelength of about 2 m, for which the amplitude can reach 40–50% of the initial flow depth, however, layer disintegration, flow separation, and/or excessive increase in the thickness do not accompany the wave propagation. Therefore, special means to suppress surface instability are not needed provided inlet fluctuations are at a level \( < 5–10\% \).

Due to the complexity of the problem, no detailed work has yet been done in the area of accommodation of penetrations with liquid metals. Such penetrations represent in MHD flow both a disturbance to the hydrodynamic flow field via the physical diversion of liquid from its initial course, and also, and more significantly, a disturbance to electrical current paths that potentially can overwhelm the flow with local and global MHD drag. Preliminary conclusions, gleaned from the discussion of side-walls above, is that any penetration will require an insulator coating to isolate the structure from the free surface flow.

5.3. Nuclear heat, tritium breeding, and activation

The thin layer of liquid does not significantly alter the radial build of ARIES-RS, however, the choice of working liquid plays a big role in the neutronics. Analyses of the nuclear heating and activation have been carried out using the ARIES-RS radial build at higher power density and with different coolants. The conclusions are that waste and damage issues in the vacuum vessel, the shield and magnets are lower when Flibe and Sn–Li are used, as compared to lithium. Solid walls damage parameters are reduced by \( \sim 10–15\% \) with the 2 cm Li-layer and \( \sim 20–30\% \) with 2 cm Flibe or Sn–Li layers. Lithium coolant offers the best tritium breeding potential at natural Li-6 enrichment. Lithium and Flibe coolants have maximum tritium breeding ratio (TBR) at 25% Li-6 enrichment (local TBR \( \sim 1.5 \) for Li and \( \sim 1.2 \) for Flibe) whereas it keeps increasing with Li-6 enrichment in the Sn–Li coolant (\( \sim \text{TBR} \sim 1.3 \) at 90% Li-6). The inclusion of beryllium drastically enhances TBR in the Flibe and Sn–Li cases (local TBR \( \sim 1.7 \) in Flibe at 25% Li-6 and \( \sim 1.4 \) in Sn–Li at 90% Li-6 enrichment) which indicates that the tritium self-sufficiency condition could be met with Flibe or Sn–Li breeder. With regard to power deposition however, the Sn–Li offers the largest power multiplication (PM) among the several breeders. PM is \( \sim 1.4 \) for Sn–Li, \( \sim 1.14 \) for Li and \( \sim 1.02 \) for Flibe. The Sn–Li breeder therefore could offer better plant thermal output for the same fusion power.

5.4. Key issues and R&D

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 addition to the plasma compatibility, the issue of establishing a viable hydrodynamic configuration threatens feasibility. 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
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). 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 and nature of the 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 forces governing the flow are gravity and friction. Without toroidal axi-symmetry of the flow and field, reliable insulator coatings will be required on all surfaces in contact with the LM layer. MHD forces from surface normal components of magnetic field can upset this balance, even when complete axi-symmetry is assumed in the toroidal direction. Additionally, gradients in toroidal field can exert a significant drag on the free surface flow. 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. 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 devices.

Apart from the free surface flow itself, MHD issues exist in the LM 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. A clever design of inlet piping may help reduce this effect, as would a reduction in the LM flowrate as well.

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 and Sn–Li database issues must be addressed for all liquid wall and blanket options as well.

6. Electromagnetically restrained lithium blanket

This section focuses on another type of thick liquid in which electromagnetic forces are utilized to restrain, or ‘confine’, the fluid. In this concept, called the electromagnetically restrained (EMR) lithium blanket, an approximately one meter thick shell of liquid lithium metal almost completely surrounds the tokamak’s toroidal plasma discharge, absorbing plasma particles, neutrons and other radiation while breeding tritium and collecting high temperature heat for power generation. The ~1 m thickness is chosen based on considerations of tritium breeding, of absorbing most of the fusion power, and of minimizing activation and damage to the solid chamber walls located behind the liquid. Of the candidate liquid materials, pure lithium metal is chosen due to its high abundance, superior tritium breeding, low chemical toxicity, almost zero neutron activation, and its high conductivity resulting in low power consumption for the EMR action.

The EMR concept converts MHD difficulties introduced by the liquid metal’s electrical conductivity into MHD advantages by deliberately injecting controlled electrical currents to influence liquid flow dynamics. As depicted in Fig. 29, two axi-symmetric liquid lithium streams enter the toroidal chamber’s top. The two streams are electrically separated there, either by an electrical
insulator or by a non-insulating structure in which some electrical dissipation is wasted via leakage. At the top, the two streams are biased to different voltages via electrodes connected to an external power supply. Poloidal current injected via these electrodes is conducted through the streams which meet and join at the bottom of the chamber. The resulting $J \times B$ electromagnetic forces push the streams against the chamber walls and thus help hold them away from the plasma. The EMR lithium blanket concept makes use of these electromagnetic forces in conjunction with the other natural forces that exist, including centrifugal (inertial) forces, contact forces, viscosity, and surface tension. The liquid’s transit time from the top to the bottom of the chamber is determined by gravity, frictional losses and chamber geometry. Since centrifugal force does not act alone in producing the liquid blanket structure, slower liquid velocities may be tolerated for the bulk liquid. Optional non-axi-symmetric solid structures could be mounted on the chamber walls to slow the lithium’s rate of descent via induced eddy currents.

Conducting liquids flowing through magnetic fields can generate large MHD forces opposing their motion, if a closed path exists for electric current to flow in response to the motion-induced electric field. For flow through pipes, these MHD forces can be overcome by using high pumping pressure, but for free-surface liquid blankets, which inherently have a low pressure gradient, external pumping is not effective. The use of injected electric currents provides the possibility of compensating for some of the MHD effects in free-surface systems. However, the flow described above will need to be highly axi-symmetric to avoid large drag forces from Hartmann layers forming on non-axi-symmetric structures. In addition, the flow must conform to the shape of the poloidal flux surfaces to a large degree so that surface normal field components are avoided as well.

In a variation on the EMR concept, a two-pass design using hot and cold liquid sublayers may be desirable to simultaneously achieve high exit temperature of the heated lithium while keeping the maximum vapor pressure of the colder plasma-facing liquid lithium surface low. That the flow is highly laminarized by the magnetic field may be an advantage here, suppressing the mixing between the two streams and allowing them to flow directly on top of one another. Detailed analysis of this problem is being carried out in conjunction with the two-stream GMD research.

### 6.1. Flow phenomena with injected electric current

Significant forces can be generated in liquid lithium metal without excessive electrical power. The threshold of significance is levitation. Lithium’s mass density is about half of water’s, so its gravitational weight density on earth is about 5000 N/m$^3$. With the approximately 5 T toroidal field typical of many tokamak reactor designs, to generate a force field matching lithium’s weight density requires a current density of $J = \rho g / B = 1 \text{kA/m}^2$ in the lithium, which implies an electric field of 350 $\mu\text{V/m}$ and an electric power dissipation of 0.35 W/m$^3$. These are modest parameters. At this ‘one-gee’ force-field level, a lithium EMR blanket surrounding an ITER-sized plasma would require a total current of 50 kA, implying a loop voltage of 0.01 V, and a power of 500 W. Increasing power to 1 MW would increase the lithium force field to the equivalent of 45 times gravity!

Fig. 29. Electromagnetic restraint (EMR) lithium blanket concept.
These calculations show that a relatively small current can easily overcome the effect of gravity. However, there will also be stray currents produced during operation (due to plasma motion) that could very well exceed the purposely generated currents. This fact demonstrates the importance of coupled analysis of the liquid wall and plasma MHD and the potential need for active control of the applied wall current.

6.2. Axi-symmetric LMMHD analyses

If highly conductive liquid metal were flowing in non-axi-symmetric patterns beside a tokamak plasma, MHD effects would produce non-axi-symmetric currents in the liquid. In addition to the potential to induce significant MHD drag, this could produce non-axi-symmetric magnetic fields which would perturb the plasma. Tokamaks and several other plasma confinement schemes require precisely axi-symmetric magnetic fields to maintain nested internal flux surfaces. They have very little tolerance for departures from axi-symmetry and develop ‘magnetic islands’ which deteriorate plasma confinement at very small levels of non-axi-symmetric magnetic field ‘ripple’. A reactor blanket must therefore avoid doing harm to the plasma equilibrium, so strict axi-symmetry is an important requirement for the portions close to the plasma of a highly conductive, fast moving, liquid blanket.

Although exact three-dimensional MHD equations for an incompressible liquid are complicated, they can be simplified without any approximation for the EMR lithium blanket concept by this requirement for axi-symmetry. In deriving exact axi-symmetric LMMHD equations with independent variables \((r,z,t)\), it is convenient to express magnetic field via the poloidal magnetic flux stream function, \(\Psi\), and the total poloidal threading current stream function, \(I\) (including any toroidal field coil system current). Formulated in primitive hydrodynamic variables, the result is six time-dependent scalar PDEs, and an ODE-integral equation describing the effect of the power supply voltage. The full derivation of this system and associated boundary conditions are given in ref. [1].

It is important to note that boundary conditions on the surfaces of the liquid and solid metallic conductors will be required for \(\Psi\). These time-varying boundary conditions depend on the plasma and poloidal field coil currents, which depend on the plasma scenario. For the case of no plasma or PF coil currents, the above equations are closed and are ready to be solved for specific cases. For cases including a plasma and/or PF coil current histories, additional data is needed to conduct an analysis. This magnetic coupling of the plasma/liquid wall/magnet coils set is an important feature of this formulation, and in the end will be required even for passive schemes like the GMD or CLiFF to fully described the liquid wall reaction to electromagnetic plasma events and to control field variations. It should be noted though that galvanic halo currents flowing between the plasma and the liquid conducting surface are not modeled in this system.

Although greatly simplified from the three-dimensional case, the above-described equations are not amenable to direct analytical solutions unless many approximating assumptions are made. That has not been done, but might perhaps be useful. The equations are amenable to numerical solution. No commercially available simulation code was identified capable of such a simulation, so the development of one has been undertaken. However, the code is not complete at the time of this report, so no detailed numerical studies of the EMR concept are yet available. Some important observations, though, can be made can be made directly about the equations.

- The toroidal swirl motion should remain identically zero as long as the poloidal current in the liquid metal is aligned to follow poloidal flux surfaces.
- If liquid velocity and injected current were both aligned to poloidal flux surfaces the velocity along streamlines should be unaffected by the variables magentic \((I\) and \(\Psi\)).

6.3. Necessary departures from axi-symmetry and key issues

It is not possible to design an entirely axi-symmetric blanket system since the flowing liquid
must cross between structural supports at some location, and in most versions of the concept need to exit and reenter the TF coil region. Analyses of these non-axi-symmetric regions will be more complex. There may be significant MHD pressure losses and pumping problems in the non-axi-symmetric regions.

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 although several efforts have been initiated and continue this year.

7. Effects of liquid metal walls on plasma performance

The interaction of liquid walls with the plasma core is a complex topic that requires future studies. In this section, we address some potentially favorable effects of flowing liquid metal walls on tokamak plasma performance and reactor attractiveness.

Liquid metal walls have been considered in tokamaks primarily for heat flux and radiation protection, and to modify particle recycling. In addition, it is clear a priori that liquid metal walls could in principle act as a close fitting conducting shell, but the advantages of this have not been examined. Here, we describe how this can lead to higher plasma $\beta$ values and improved confinement.

The stabilizing effects of the liquid metal can be either passive (merely due to the presence of a nearby conductor), or active (due to the flow of the liquid metal). The passive effects are significant because liquid metals such as lithium can be closer to a reactor plasma, as well as thicker, and thus more stabilizing. The active effects are important because they can prevent flux penetration in steady state, preventing resistive wall modes by naturally converting liquid metal kinetic energy into magnetic flux to compensate for resistive losses. It is widely recognized that resistive wall modes strongly limit performance in advanced tokamak operation, and also seriously affect reversed field pinches (RFPs) and other toroidal confinement devices. We consider both passive and active effects here.

7.1. Passive stabilization by LM walls

In reactor studies such as ARIES-RS [14], passive stabilizing conductors are placed behind the blanket. These conductors must maintain a toroidally continuous conduction path to stabilize the vertical instability. They are placed behind the blanket because structural metals compatible with the fusion environment are poor conductors, and a thickness to provide substantial conductivity would negatively affect tritium breeding if placed in front of the blanket. Also, the degradation of the conductivity of such metals due to radiation damage is a problem. In particular, radiation degradation of joints and welds may jeopardize the required toroidal continuity. The removal of the stabilizing plates by this significant distance from the plasma, substantially reduces their stabilizing effect, limiting tokamak reactor designs to an elongation $\kappa$ of approximately two or less. It is well known that the maximum plasma current is a strong function of elongation, and thus, so is the attainable MHD $\beta$ as well as the confinement predicted by scaling laws.

In contrast, molten lithium metal can be placed close to the plasma since it does not degrade breeding (in fact it improves it). Furthermore, the conductivity of a liquid is unaffected by the radiation environment. Liquid plasma facing designs considered by APEX have lithium much closer to the plasma. Alternatively, a liquid lithium vessel could be placed just behind a solid first wall (maintaining a toroidally continuous conduction path). Below we consider the effects of this on $\kappa$ and thus on $\beta$.

An $n = 0$ vertical resistive stability code has been written. It solves the perturbed Grad–Shafranov equation $\Delta \Delta \Psi = (\mathcal{F} \mathcal{F}^\prime) \Delta \Psi$ as an initial value code including inductive fields and resistive elements. The elliptic operator is inverted with vacuum boundary conditions, and includes the effects of external resistive coils, a resistive wall, and active feedback coils with voltages determined from the signals of sensor coils. Presently pressure is not included in the equi-
librium, and the toroidal current profile $FF'$ is taken to be a constant. The voluminous literature on vertical stability [18] shows that plasma pressure is not a major effect (and is usually stabilizing), and hollow current profiles (as expected with high bootstrap fraction operation) are expected to be more stable than a flat current profile. Thus, the results below are more pessimistic (and thus conservative) than expected from more realistic profiles.

As examples, we consider the vertical stability of a $\kappa = 3$ plasma with aspect ratio $A = 3$ and 4, with 4 cm of lithium (a typical number for thin liquid plasma facing concepts), and the liquid at a distance $b/a = 1.2$ (i.e. a distance from the plasma of 20% of the horizontal minor radius, or about 30 cm for ARIES-RS). Liquid facing concepts usually have liquid closer than this, and would be more stabilizing. The resistive wall time is roughly 0.5 s, and the resistive vertical instability growth time is about 0.66 s. This time scale is easily within the reach of existing vertical feedback technology (which can have response times of the order of a millisecond, or slightly less). With a standard feedback geometry with the active coil above the plasma a distance which would place it behind a 1 m shield, and a sensor coil on the outboard side (though a distance which would place it inside the shield but behind the first wall), vertical stability is achieved with feedback gain about an order of magnitude larger than in the case $\kappa = 2$, and with feedback response times $\lesssim 50$ ms. This appears to be within the range of present technology. Little effort has been spent optimizing the parameters of the feedback system, and considerable improvement might be possible.

We find the consequences of this to the attainable $\beta$ in advanced technology (AT) modes are large. The MHD equilibrium code TOQ (used routinely by the general atomic (GA) MHD group) has been used, to obtain high bootstrap fraction equilibria for $A = 3$ and 4. Broad pressure profiles are used which have been used by the GA group in $\beta$ optimization studies for $A = 1.4$ tokamaks. The maximum $\beta$ for ballooning stability for $A = 4$ and 3 is:

$$
\begin{align*}
\kappa = 2 & \quad \beta = 5–7\% & \beta_N = 4.5 & S = 7.3 \\
\kappa = 3 & \quad \beta = 20–22\% & \beta_N = 5.7 & S = 13.9
\end{align*}
$$

As can be seen, the stable $\beta$ is increased by about a factor of 3. Note $\beta_N$ does not increase much, so the increased $\beta$ is mainly due to increased current. The $\beta$ and $\beta_N$ values for $\kappa = 2$ are quite similar to those found in the ARIES-RS study ($\kappa = 1.9$, $\beta_{\text{max}} = 5.4\%$, $\beta_{N,\text{max}} = 4.8$). We do not have capabilities to examine $n = 1$ stability, so we estimate stability based on the shape factor $S = (1/aB)q_{\text{edge}}$. With wall stabilization, the maximum stable $\beta_N$ is an increasing function of $S$ and profile flatness $p(0)/\langle p \rangle$. If we extrapolate published results by Turnbull et al. [20], we infer that the much higher shape factor for $\kappa = 3$ should enable $n = 1$ stability for the modestly higher $\beta_N$ value.

This has large consequences for a reactor. For a 1 GW reactor with 1 m of inboard blanket/shield and 13 T superconductors (and the same $\beta$ as ARIES-RS for $\kappa = 2$):

$$
\begin{align*}
\kappa & \quad \beta & \quad \text{Major MW/m}^2 & \quad \rho^* & \quad H-\text{factor} \\
R & \quad (\text{ITER89P})
\end{align*}
$$

<table>
<thead>
<tr>
<th>$\kappa$</th>
<th>$\beta$</th>
<th>$\rho^*$</th>
<th>$H$-factor</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.9</td>
<td>4.8%</td>
<td>5.5</td>
<td>1/500</td>
</tr>
<tr>
<td>3</td>
<td>18%</td>
<td>3.15</td>
<td>9.5</td>
</tr>
</tbody>
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As can be seen, there is a large reduction in size and therefore mass and cost. For example, the length of superconducting wire needed is reduced by about a factor of 2.5. The wall loading is in the range considered as the nominal case for APEX design evaluations of advanced wall concepts (8 MW/m$^2$).

Note that the $\rho^*$ of the $\kappa = 3$ reactor is the same as JET and JT-60. Thus, this reactor is not an extrapolation in $\rho^*$, but rather in geometry. Since geometry is not a fundamental physics variable, we expect that extrapolations in $\kappa$ from existing machines can be made with much less uncertainty than extrapolations in $\rho^*$. 

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7.2. Active stabilization of resistive wall modes

We now consider the effects of a flowing wall on the \( n = 1 \) resistive wall instability. We employ a self-consistent limit of the MHD equations to obtain an analytically solvable model of the \( \beta \) driven external kink mode. The model uses high \( A \), reduced MHD, simplified with flat current and pressure profiles and circular plasma cross-section. We note that independently, a similar model was investigated by Betti et al. with similar results.

We obtain a \( \beta \) driven kink mode, which requires coupling between adjacent poloidal mode numbers \( m \) for instability. The mode can be stabilized with an ideal (perfecting conducting) wall. Finite resistivity and rotation are added numerically, using the analytic plasma response. As expected, with no rotation there is a resistive wall mode with \( \gamma_{RWM} \sim \) the resistive wall time. For a poloidally rotating wall (which adds a current \( \eta \delta j = \nu_0 \times \delta B = \nu_0 \delta B_r \) in addition to the inductively driven current), we find stabilization when the poloidal transit time for the flow to go from the top to the bottom \( 1/\tau_p = \nu_0/\pi r \) is fast enough that:

\[
\frac{1}{\tau_p} \geq \gamma_{RWM}.
\]

This result has also been found independently by Betti. Here, we note that for 4 cm Li, this corresponds to velocity levels considered by the APEX design for liquid metal walls. Note that it is not necessary for the flow to be facing the plasma, but rather the flow could be in a cavity behind a solid first wall (but close to the plasma).

This stabilization can be interpreted as an inability of the \( n = 1 \) flux to penetrate if the metal flows from the top to the bottom more rapidly than the growth rate, since then the metal is always being replaced by fresh metal. Alternatively, the result can be interpreted as the dephasing of a toroidal instability which requires a particular phase relationship between different poloidal harmonics. Since each \( m \) number is Doppler shifted by a different amount, there is not rotating frame where the relative phases needed for instability can be maintained.

Note that stability requires that the conducting wall be placed somewhat closer for stability than is the case for a perfectly conducting wall. This is due to the fact that the mode can rotate with a frequency to remove wall stabilization for a single poloidal harmonic. Since only two of the three harmonics are wall stabilized, the stabilization is not as effective. However, in more realistic shaped equilibria, there is a much broader spectrum of \( m \) numbers required in the eigenfunction than in this circular model. Since only one among the large number of harmonics can escape wall stabilization, we anticipate that shaped equilibria will have rotational stabilization effectiveness more nearly equivalent to that of a perfectly conducting wall.

Stabilization of resistive wall modes would lead to several benefits. Higher \( \beta \) steady state equilibria could be obtained, with very hollow current profiles. Steady state operation with such profiles enables high bootstrap fractions and thus low recirculating power. Also, hollow current profiles are theoretically predicted to give \( E \times B \) shearing rates larger than instability growth rates for conventional drift instabilities, leading to transport barriers and high confinement. Hollow current profiles are well correlated experimentally with such good confinement. Thus, flowing liquid walls may enable the conditions needed for high steady state confinement.

We note that the codes used to obtain the above results are being developed further to handle arbitrary equilibria output from an equilibrium MHD code. We also note that flowing liquid metals can behave differently under perturbations since they can be pushed out of the way. Analysis indicates that liquids flowing at the speeds indicated above are not greatly affected by this (though a stationary liquid would be), but inclusion of this effect is also in progress. This last point especially indicates that there is a synergism between both liquid metal walls and tokamak physics performance, both in \( \beta \) and confinement, as well as in the analysis of dynamics of plasma discharges and flowing wall behavior.

We recommend that this synergism be pursued vigorously through cooperation between the fusion physics and the fusion engineering communities.
8. Plasma–liquid surface interactions and edge modeling

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. 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 [21] and to fuel dilution are shown in Fig. 30 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 resulting in increased evaporation. The maximum tolerable evaporation rate determines the maximum allowable surface temperature of the liquid, and the sputtering analysis determines the required edge-plasma.

A multi-faceted, self-consistent model is required to make a complete evaluation of the 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. The progress is summarized below and presented in more detail in Ref. [1] for the following areas: two-dimensional fluid transport simulations of properties of the hydrogenic edge plasma; two-dimensional fluid transport simulations of impurity penetration to the core region arising from evaporating Flibe and Li-based walls; one-and-a-half-dimensional kinetic and two-dimensional fluid transport calculations of evaporated and sputtered impurities from liquid divertor plates; two-dimensional simulations of intense power deposition to a lithium divertor plate during a disruption; one-and-a-half-dimensional plasma core transport modeling, beginning simulations of the behavior of small liquid samples in the PISCES plasma divertor simulator and the DIII-D tokamak.

8.1. Edge fluid transport simulations

We have used the two-dimensional UEDGE code [22] to obtain profiles of hydrogen ion density, parallel ion velocity, and separate ion and electron temperatures. The base-case is an ITER-like tokamak where the transport simulation sets boundary conditions of power and density a small distance inside the magnetic separatrix and calculates the resulting scrape-off layer (SOL) profiles. We have characterized two-dimensional plasmas profiles for both high-recycling regimes (Flibe or other non-recycling divertor) and low-recycling (lithium divertor which retains incident hydrogen). The low-recycling case results in high electron temperature at the divertor and low density, with the opposite being true for high recycling. An important consideration for the low-recycling case is the large particle flux out of the core that must be maintained by an edge particle-fueling source such as pellets.

To assess the effectiveness of the edge plasma for shielding the core from impurities, the UEDGE calculations are extended to include im-
purity gas evaporating from the liquid wall. A number of processes are included in this modeling. The impurity gas is emitted from the wall in the form of atoms at typically 1 eV, although a range of energies have been used to assess the energy of the atoms after molecular dissociation which is not yet modeled in any detail. These neutrals diffuse by elastic collisions with ions until they are ionized by the electrons of the edge plasma. Once an ion, the impurity diffuses across the magnetic field with anomalous diffusion coefficients estimated from present experimental devices. Thus, the ions can diffuse radially into the core or back to the liquid wall where they are assumed to be absorbed. In addition, the ions can flow along the magnetic field and out of the system. The electron energy lost by ionizing the impurities through all of their charge states is included, so that the impinging impurities decrease the electron temperature, especially near the liquid surface. A typical set of charge-state profiles from fluorine from a Flibe wall are shown in Fig. 31.

Similar calculations have begun for Sn–Li walls where only Li is evolved from the surface; it is assumed that evaporation of Sn is negligible. Lithium penetrates less easily to the core due, in part, to its lower first-ionization potential of 5.4 V compared to 17.3 V for fluorine from Flibe. Secondly, if one considers a Sn–Li, its evaporation rate is less than that of Flibe at a given temperature.

The comparison between the fluorine (Flibe) cases and the lithium (Li, Sn–Li) cases with respect to impurity concentration is shown in Fig. 32. This figure quantifies what core impurity core density should be expected for a given gas flux, which can be determined from known data of the evaporation rate at a given liquid surface temperature.

From Figs. 30 and 32, one can deduce that for an ITER-like tokamak with 150 MW of plasma power flowing into the scrape-off layer, impurity penetration to the core may be kept to an acceptable level if the liquid surface temperature for Flibe is 540°C or less, while for Sn–Li it is 740°C or less. However, these results are quite preliminary with one of the most important uncertainties being the fact that the transport simulations have not yet found steady state solutions at the larger gas flux regions of Fig. 32 shown by the dotted lines. These dotted line portions of the curves are just those being used to make the estimates of the maximum surface temperature quoted above.