Thus the highest priority of our present research is to better resolve and understand the non-steady solutions. Such solutions correspond to where the electron temperature near the surface abruptly drops below a few eV owing to impurity radiation and particle energy losses to the wall. This is a ‘detached’ type of plasma, but here the detachment is from the side wall rather than the divertor plate.

The calculations for impurity influx from side walls have been mostly performed in a tokamak geometry. More simulations are needed for alternate confinement geometries such as the field-reversed configuration (FRC), spheromak, spherical torus, and others.

8.2. Kinetic simulations of the sheath and presheath

Kinetic simulations are performed for the region near liquid divertor plates using the test-particle codes BPHI and WBC codes [23] with Monte Carlo collisions. BPHI focuses on the sheath region, including ionization within the sheath, whereas WBC uses a reduced sheath model and includes the presheath region \( \sim 10 \text{ cm} \) in front of the plate. Both codes begin with a hydrogen plasma from a two-dimensional fluid transport code, but then trace sputtered and evaporated impurities from the plates made of Flibe or lithium until they escape upstream or are redeposited on the plates.

For the WBC code lithium analysis, the following is observed: (1) very high near-surface lithium redeposition rate (\( \sim 100\% \)), (2) high redeposited average energy with highly oblique Li ion impingement. Result (1) is favorable showing low potential for plasma contamination by sputtered lithium, even for the low-collisionality, low-recycle regime. Result (2) gives rise to concerns about runaway self-sputtering although preliminary estimates using initial ALPS/APEX project data show that this will probably not occur.

WBC calculations for Flibe assessed the near-surface transport of the individual sputtered Flibe constituents of F, Li, and Be. As with the lithium surface calculations, a highly preliminary sputtering model was used. Results using a hydrogen plasma in the high-recycle regime (\( T_e = 30 \text{ eV}, n_e = 3 \times 10^{20} \text{ m}^{-3} \)) show a high redeposition fraction for each element. There is a lower potential for self-sputtering runaway due to lower redeposition energies and less oblique incidence.

BPHI sheath code calculations were performed for a low-recycle plasma divertor regime with a lithium surface. Preliminary results, for one particular low-recycle regime, show that a majority of slow-moving, evaporated lithium atoms will be ionized in the sheath and will be returned to the surface due to strong sheath electric field. On the other hand, the sheath heat transmission factor will increase due to reduced sheath potential resulting from the extra electrons and ions produced by in-sheath ionization. The resulting increase in heat flux is of concern in terms of a runaway effect but this may be mitigated by the transient nature of the overheating and the fact that the lithium is flowing.

8.3. Additional on-going edge plasma simulation work

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 [24] 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 for 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 splash droplets. 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, affects 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; and 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 fusion core plasma is being studied with the one-and-a-half-dimensional core transport code ONETWO [25] which 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 performed for the ARIES-RS design.

Finally, it is important to benchmark models predicting 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 University of Illinois 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 to impurity transport modeling for the DiMES probe is that the probe is localized to one toroidal location, so three-dimensional effects do enter which can only be estimated by the present codes. Nevertheless, these calculations begin the vital process of comparing modeling results with experimental data. Larger-scale liquid samples in experiments will improve this benchmarking. There is on-going work to use liquid divertor surfaces in other devices such as CDX-U. This type of activity is important to provide the experimental data base to validate models predicting the influence of such walls in fusion-related devices.

9. High-temperature solid wall with lithium evaporation (EVOLVE)

This section discusses a novel method to extend the capabilities of a solid wall by using a high-temperature refractory alloy with heat extraction achieved by lithium evaporation.

The desire to achieve both high power density and high power conversion efficiency leads to several required features of a first wall and blanket concept. Achieving high power density means that the coolant heat removal capability must be high and the first wall material should have attractive thermophysical properties (high thermal conductivity, low thermal expansion, etc.). Achieving high power conversion efficiency means that the first wall and blanket should operate at very high temperatures. Materials operating at very high temperatures generally have limited strength and, therefore, such a concept should operate at low primary stresses. This means that the coolant pressure should be as low as possible, and the temperatures throughout the blanket should be as uniform as possible to reduce thermal stresses.

One system that has this potential is the EVOLVE (evaporation of lithium and vapor extraction) concept. The key feature of the EVOLVE concept is the use of the heat of vaporization of lithium (about ten times higher than water) as the primary means for capturing and
removing the fusion power. A reasonable range of boiling temperatures of this alkali metal is 1200–1400°C, corresponding with a saturation pressure of 0.035–0.2 MPa. Calculations indicate that an evaporative system with Li at ~1200°C can remove a first wall surface heat flux of > 2 MW/m² with an accompanying neutron wall load of > 10 MW/m². The system has the following characteristics:

1. The high operating temperature translates naturally to a high power conversion efficiency.
2. The choices for structural materials are limited to high temperature refractory alloys. A tungsten alloy, e.g. W–5%Re, is the primary candidate as a structural material, with tantalum alloys as the back-up.
3. The vapor operating pressure is very low (sub-atmospheric), resulting in a very low primary stress in the structure.
4. The temperature variation throughout the first wall and blanket is low, resulting in low structural distortion and thermal stresses.
5. The lithium flow rate is approximately a factor of ten slower than that required for self-cooled first wall and blanket. The low velocity means that an insulator coating is not required to avoid an excessive MHD pressure drop.

The areas addressed are first wall and blanket design, tritium breeding, activation and waste, power conversion, first wall thermo-mechanical behavior, tritium extraction, and critical issues. The key features of the design are summarized in Table 11.

### Table 11

<table>
<thead>
<tr>
<th>Feature</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Heat capture and removal</td>
<td>Li vapor</td>
</tr>
<tr>
<td>Li vapor pressure</td>
<td>0.035 MPa</td>
</tr>
<tr>
<td>Li vapor velocity</td>
<td>~500 m/s</td>
</tr>
<tr>
<td>Structural material</td>
<td>W–5%Re</td>
</tr>
<tr>
<td>Operating temperature</td>
<td>~1200°C</td>
</tr>
<tr>
<td>First wall heat flux</td>
<td>2 MW/m²</td>
</tr>
<tr>
<td>Neutron wall load</td>
<td>10 MW/m²</td>
</tr>
<tr>
<td>Tritium breeding ratio (local two-dimensional)</td>
<td>1.37</td>
</tr>
<tr>
<td>Power conversion efficiency</td>
<td>~ 57%</td>
</tr>
</tbody>
</table>

The cross-section design of the EVOLVE concept is illustrated in Fig. 33. In the EVOLVE concept, the first wall and primary breeding zone are combined into one unit. Behind this unit, there is as a separate component, a high temperature shield at the inboard region and a secondary breeding blanket at the outboard region. Behind the secondary breeding zone there is, as a separate component, an additional high temperature shield, required in order to meet the shielding requirements of vacuum vessel and magnets.

The first wall consists of a tube bank arranged in the toroidal direction as shown in Fig. 34. Within each tube is another tube that supplies the liquid lithium to the first wall. There are two different methods under consideration for the distribution of the liquid metal at the surface. One of them employs a large number of jets generated by nozzles in the supply tube by which the LM is distributed to the backside of the first wall. With the other one, capillary forces in a wick structure, arranged at the backside of the first wall, are employed to transport the liquid lithium from the supply tube to the entire surface of the first wall. This wick is connected to the supply tube via longitudinal slots in this supply tube. For a surface heat flux of 2 MW/m², a toroidal segment width of 3 m, and the tube dimensions given above, a boiling temperature of 1200°C (saturation pressure 0.035 MPa) results in a liquid metal velocity in the feed tube of about 1 m/s and a vapor velocity of about 500 m/s. This is about one-third of the sonic velocity and results in a tolerable pressure drop.

The blanket consists of a number of trays, stacked poloidally, containing liquid lithium. A space is left between trays to allow the Li vapor to be removed from the blanket. Each tray contains a lithium pool with a height of 10–20 cm, which is maintained constant by a system of overflow tubes. The large volume heating of the lithium leads to boiling. The vapor bubbles have to rise in the pool and separate from the liquid metal at the surface. From here the vapor flows a short distance in parallel to the surface before it enters the vertical vapor manifold. Entrained liquid metal will be separated there. Behind the trays is a manifold, approx. 20 cm thick, for collecting the
Fig. 33. Cross-sectional view of the EVOLVE first wall/blanket concept.

Fig. 34. Schematic of EVOLVE first wall tubes and blanket trays containing Li.
Li vapor. The total radial thickness of the first wall and blanket is approx. 70 cm.

Two-dimensional neutronics modeling of the front evaporation cooled blanket of EVOLVE is needed to properly account for the poloidal heterogeniety and gaps between trays. The $R-Z$ geometrical two-dimensional model used in the calculation includes the FW, trays with Li vapor manifold, secondary breeding blanket, shield, VV, and magnet in both the IB and OB regions. Both the IB and OB regions are modeled simultaneously to account for the toroidal effects. The TWODANT module of the DANTSYS 3.0 discrete ordinates particle transport code system was utilized. The overall TBR calculated for the reference design using the two-dimensional model is 1.37. It is based on the conservative assumption of no breeding in the divertor region. Tritium breeding (69.8%) occurs in the trays (57.3% OB and 12.5% IB). The OB secondary blanket contributes 27.7% of the total overall TBR (20.2% behind trays and 7.5% between trays). The contribution of the shield is only 2.5% (1% OB and 1.5% IB). Tritium breeding has a comfortable margin that allows for design flexibility.

There are two coolant streams exiting from the blanket. The front part of the blanket, including the first wall and the primary breeding zone, is cooled by boiling lithium, which carries approximately two-thirds of the total thermal power. The back part of the blanket, composed of secondary breeding zone and the high temperature (HT) shield at the outboard zone and the HT shield at the inboard zone, is a conventional self-cooled liquid lithium blanket with an exit temperature of also 1200°C, which carries the other one-third of the thermal power. The two blanket coolant streams will be fed to two heat exchangers to transfer the thermal energy to a helium loop. The reason that He is used for the secondary coolant is that a closed cycle gas turbine can be used for very efficient power conversion. The two lithium streams exit from the blanket operates in series, with the liquid lithium stream to heat up the secondary He from 700 to 800°C, while the high temperature lithium vapor super heat the same He stream from 800 to 1000°C. The He at 1000°C will enter a He turbine for power conversion.

With a very high He temperature, and very high recuperator, compressor and turbine efficiencies, a very high cycle efficiency of 57.7% is calculated. This thermal efficiency includes the pumping power of the secondary He stream, but does not include the pumping power of either of the lithium streams, which is very small in any case.

Finite element thermal and stress analyses have been performed for the first wall subjected to surface heat fluxes of 1.5 and 2 MW/m², a coolant temperature of 1200°C, and a coolant pressure of 0.05 MPa. A single tungsten tube of radius 2 cm and wall thickness of 3 mm deforming under generalized plane strain condition is considered. The primary membrane stress in the EVOLVE first wall is so low (< 1 MPa) that neither low-temperature nor high-temperature ratcheting should be a limiting criterion for the surface heat flux. The peak surface heat flux will be controlled either by creep-fatigue (which is not considered here) or possibly by brittle fracture (due to helium-embrittlement). The temperature distribution for a peak surface heat flux of 2 MW/m² and a heat transfer coefficient of 40 000 W/m²°C shows a peak temperature of 1317°C. The peak stress intensity is 158 MPa, which easily satisfies the ratcheting limits. Very little ductility is needed to maintain the allowable stress limit at a high value. For example, if the uniform elongation remains higher than 2% or the reduction in area at failure is > 1%, then the allowable stress is > 300 MPa. A stress of 150 MPa would be allowable even for completely embrittled tungsten at 1200°C.

The EVOLVE concept is at an early stage of evaluation. At this stage, it is important to assess the potential of the concept, identify crucial issues, and to define needed R&D work to resolve those issues. The critical issues to be addressed in the near future are:

1. Will the backside of the first wall remain wetted under all conditions?
2. Will the vapor generated in the stagnant boiling pools of the primary breeding region separate fast enough from the liquid metal?
3. Will the liquid metal overflow system work and lead to equal liquid metal pressure in each tray?
4. Is it possible to fabricate entire blanket segments of tungsten or tungsten-alloys in spite of their low ductility and their limited weldability?
5. How will the structural material behave under intense neutron irradiation?
6. Will the high after heat in tungsten cause a safety problem in case of a LOCA?

10. High-temperature solid wall with helium cooling

This section explores extending the capabilities of a solid wall using high temperature refractory alloy cooled with high-pressure helium. A primary motivation is to explore the possibility of using a high-temperature helium for high-efficiency energy conversion in a gas turbine cycle.

10.1. Material selection and compatibility

The material selection and compatibility issues are discussed in Section 12. Pure tungsten or tungsten alloyed with \(~5\%\) Re (to improve fabricability) appear to be suitable candidates. The unirradiated mechanical properties of tungsten are strongly dependent on thermomechanical processing conditions. The best tensile and fracture toughness properties are obtained in stress-relieved material. In order to be conservative, since data are not available on the possibility of radiation-enhanced recrystallization of W, and also to account for the presence of welds in the structure, the preliminary design is based on recrystallized mechanical properties. There are no known mechanical properties data on tungsten or tungsten alloys at irradiation and test temperatures above \(~800\)°C. There are no known fracture toughness or Charpy impact data on tungsten irradiated at any temperature. Pronounced radiation hardening is observed in W and W-Re alloys irradiated at temperatures of 300–500°C to doses of \(~1–2\) dpa, which produces significant embrittlement in tensile tested specimens (\(~0\%\) total elongation). Simple scaling from existing data on irradiated Mo alloys suggests that the operating temperature for W should be maintained above \(~800–900\)°C in order to avoid a significant increase in the ductile-to-brittle transition temperature (DBTT). The upper operating temperature limit for tungsten will be determined by thermal creep, helium embrittlement, or oxide formation issues. The thermal creep of W becomes significant at temperatures above \(~1400\)°C. Helium embrittlement data are not available for tungsten; however, based on results obtained on other alloys, helium embrittlement would be expected to become significant at temperatures above \(~1600\)°C (\(~0.5\) melting temperature, \(T_m\)). The formation of volatile oxides is another potential problem in tungsten at temperatures above \(~800\)°C especially during an air ingress event. However, if the oxygen partial pressure in the helium coolant can be maintained at or below 1 appm, then the rate of corrosion is calculated to be less than 2 \(\mu\)m/year for temperatures up to \(~1400\)°C. In summary, the selected upper temperature limit for tungsten in the structure of the preliminary design He-cooled system is \(1400\)°C depending on the applied stress.

10.2. He coolant impurity control

Refractory metals like W, Mo, and V are sensitive to grain boundary oxidation and embrittlement. However, if the oxygen (including \(\text{H}_2\text{O}, \text{CO}_2, \text{CO}, \ldots\) etc.) partial pressure in the helium coolant can be maintained at or below 1 appm, then the rate of corrosion may be acceptable. With the use of Brayton cycle as the power conversion system (PCS), without the need of using high temperature water as the secondary coolant, the ingress of oxygen impurities should be much lower than the system that uses a high-temperature intermediate heat exchanger. For impurity extraction, several powder metal solid getters have been developed. Most are based on zirconium metal (ZrAl, ZrVFe, \ldots etc.). With these materials, hydrogen can be pumped reversibly by temperature control. These solid getters will pump active gases (oxygen, oxides, N, and \(\text{C}_x\text{H}_y\)) reversibly and have been used on the tokamak experiment TFTR. In the semiconductor industry, getters have recently achieved the control of impurities to a level lower than 1 appb. These are
commercial modular units with no moving parts and are self-monitoring in design.

10.3. Mechanical design and reliability

Several first wall and blanket system configurations were evaluated. The mechanical design is shown in Fig. 35. The helium-cooled refractory alloy design includes a high temperature helium-cooled first wall and a lithium bath that is also cooled with high temperature helium.

The first wall is made up of separate units which, in this case, are connected to separate cooling manifolds at the back of each module. The first wall units consist of multiple parallel passages connected through an integral manifold to round inlet and outlet connections. The large modules contain the lithium in a single volume, with pure lithium in the breeding zone and a combination of lithium and steel balls in the shielding zone. The temperature is relatively uniform, although there will be some gradients, albeit transient, between the front and back structural walls. There are two inboard and three outboard modules to each of the 16 sectors arranged in the toroidal direction. The piping is routed in two circuits. The first circuit includes the first wall and part of the interior heat exchange tubing. Helium at 800°C enters the first wall through the supply manifold and exits into the first wall outlet mani-
fold at 950°C. The helium is then routed inside the lithium can to the first supply manifold for the heat exchange tubes. The first tube circuit exits into a return manifold at 1100°C. The second tube circuit is fed at 800°C and exits at 1100°C.

One of the primary goals of the APEX study is to increase the availability of fusion reactors by increasing the mean time between failures and by decreasing the mean time to repair. To this end, we recommended the approach of sector maintenance, modular maintenance for everything and pretested modules for all components.

10.4. First wall blanket thermal-hydraulics design and analysis

10.4.1. Design inputs

With the mechanical design concept described earlier, we determined the material volume fractions and power generation from different FW/blanket zones. We performed iteration calculations between thermal hydraulics and nuclear analysis. The normalized volumetric power density for W-alloy as a function of distance $x$ from the first wall is approximated by $PW(x) = 9e^{-3x}$ W/cc per neutron wall loading in MW/m$^2$. The normalized volumetric power density for Li-breeder is approximated by $PLi(x) = 4e^{-3x}$ W/cc. Other input parameters are:

- Reactor power output: 2005 Mwe
- Helium pressure: 12 Mpa
- Helium mass flow-rate: 2.528 kg/s
- Helium $T_{in}/T_{out}$: 800°C/1 100°C
- Structural material: W–5Re
- Max. neutron wall loading: 7.49 MW/m$^2$
- Max. surface heat flux: 2.16 MW/m$^2$

10.4.2. First wall design

The use of helium as a FW/blanket, divertor coolant has been proposed in various fusion design studies. To handle the high surface heat load, extended heat transfer enhancements by porous medium and swirl tape were evaluated.

10.4.3. Porous medium

A porous medium enhances heat transfer from the wall to the helium thereby reducing the film temperature drop and the absolute temperatures of the first wall. The design activity reported here was based, in part, upon development activities by two small US businesses. One of the companies, Thermacore, Inc., uses a porous medium to enhance heat transfer. Thermacore designed and built a series of helium-cooled modules that were tested at Sandia and elsewhere [26–29]. One advance in their development of a helium-cooled heat sink was the development of designs that connected open axial inlet and exhaust passages to circumferential flow passages that contained the porous medium, as shown in Fig. 36. The other company, Ultramet, Inc., has experience in fabrication of refractory materials. Ultramet has designed and built commercial products made of refractory metals for rocket nozzles and other applications in which they use a metallized foam that is integrally bonded to fully dense material [29] as shown in Fig. 37. Their experience demonstrates that a tungsten channel with integrated porous medium structure can be fabricated.

10.4.4. Swirl tape first wall design

Another method for extended surface heat transfer is to use a swirl tape insert. Swirl tape increases the heat transfer coefficient by increasing the effective flow velocity of the coolant and increasing mixing. There is a large amount of reliable data available on this method. However, the corresponding increase of coolant flow friction factor has to be accounted for.

For this calculation, the enhancement in heat transfer coefficient is given by, $h_{en} = 2.18/Y^{0.09}$, and the increase in friction factor is given by $f_{en} = 2.2/Y^{0.406}$, where $Y$ is the twist ratio defined by pitch/2*diameter of the tube. Therefore the equivalent $h_{eq} = h_{en}^*h$ and equivalent friction factor $f_{eq} = f_{en}^*f$, where $h$ and $f$ are heat transfer coefficient and friction factor for a simple circular tube, respectively. In the following calculation, we used $Y = 2$.

Using a maximum neutron wall loading of 7.11 MW/m$^2$, and maximum surface heat flux of 2.06 MW/m$^2$, and the swirl-tube first wall coolant ve-
locity range of 54–62 m/s, the W-alloy maximum
temperature was found to be in the range of 1073–1242°C. With simple tubes in the
blanket, the W-alloy maximum temperature is
1199°C, and the lithium maximum temperature is
1228°C.

The first wall and blanket system pressure drop
was also estimated. Including frictional losses,
turns, contractions, expansions, and main helium
inlet and outlet pipes, the total pressure drop was
estimated to be 0.61 MPa, which gives a ΔP/P of
5.1%.

10.5. Thermal stress analysis of APEX first wall
design

A ‘ground rule’ of the APEX study was that
structures should be robust, and specifically, 3
mm was taken as a minimum first wall thickness
(with some scientists recommending 5 mm). A

Fig. 36. Thermacore circumferential flow design.
Fig. 37. Porous Ta implant, diameter is 0.75 in.
Fig. 38. Effect of FW/blanket inlet temperature on PCS gross efficiency.
central challenge in the design is to relieve the primary and secondary stresses that result from the high helium pressure, surface heat load and the related steep thermal gradient in the heated surface. The FW is permitted to flex to relieve the thermal strain (bending stresses) form the surface heat load.

A thermal analysis of a dual-channel FW structure (without the porous medium included) was performed using two-dimensional plane strain models (PATRAN: ABAQUS) for a surface heat load of 2 MW/m² and an internal pressure of 10 MPa; the FW was permitted to flex under the heat load. At 1000°C, the maximum von Mises stress is 80 MPa, this is well within the suggested stress limits stated below. Further iteration will be needed for the reference case of 12 MPa pressure but the result should not be significantly different. The double-tube wall design will then be incorporated into the porous medium design in the next design phase.

The thermal stress due to a prescribed temperature distribution along a single tube first wall of the APEX FW/blanket was also analyzed using the COSMOS finite element code. The structural model consisted of two-dimensional beam elements interconnected along with the defined temperature distribution. The first wall tube has an i.d. of 1.6 cm and an o.d. of 2.2 cm. The beam elements representing the lithium case are 0.2 × 2.2 cm for the inner case and 3.8 × 2.2 cm for the outer strong back case. The lithium case is supported by a guide structure attached to the vacuum vessel. It is assumed that the guide structure allows free thermal expansion of the lithium case in the vertical and radial directions. The following W–5Re alloy properties were taken at 1000°C: Young’s modulus = 392 Gpa, Poisson’s ratio = 0.267, and coefficient of thermal expansion = 3.96 × 10⁻⁶/°C.

The deformed shape and maximum stress due to the prescribed assigned temperature distribution and boundary conditions were calculated. The tangential thermal growth of the first wall tube of 2.0 mm requires that the blanket modules be installed with 4.0 mm gaps in the cold condition to prevent contact with one another during operation. The radial thermal growth of the plasma facing tube is 4.4 mm. Since we projected that the irradiated W-alloy should be treated more as a brittle than ductile structural material, we proposed that the stress criteria for evaluating calculated stress intensities for tungsten materials be taken as one-half the ultimate stress (133 MPa) at 1000°C for welded joints and two-thirds the ultimate stress (177 MPa) away from joints. Adopting these criteria, the allowable stress at the weld joint due to all load combinations is 152 MPa at 1000°C. Since the proposed support structure will allow free thermal expansion of the lithium case, only the temperature difference between the first wall tube and lithium case will induce thermal stresses. The maximum thermal stress occurs in the first wall tube at its junction to the lithium case and is only 6 MPa.

Although the proposed concept for supporting the blanket induces low thermal stress, details of how to implement the support concept will certainly result in higher thermal stresses. Also, the stresses due to dead weight, pressure, and disruption loads have yet to be calculated. This will be performed in the next phase of design.

10.6. Nuclear analysis

Based on the material volume fractions generated, the reference design was determined by iteration between the thermal hydraulics task and assessed the impact of W-alloy on the nuclear heating profiles across the blanket and power multiplication (PM), and on the tritium breeding profiles and the tritium breeding ratio (TBR). The impact of Li-6 enrichment on these profiles and on TBR and PM is also assessed. In addition, we assessed the damage indices, expressed in terms of DPA, helium, and hydrogen production rates at several key locations including the vacuum vessel (VV) and TF coil case. When compared to other refractory alloys like TZM and Nb–1Zr, the best local TBR performance is with W and Li breeder. Based on a one-dimensional cylindrical base on
the outboard blanket geometry, the TBR increases with Li-6 enrichment and starts to saturate at a value of \( \sim 1.43 \) when Li-6 enrichment is \( \sim 35\% \). The damage parameters, DPA rate, helium and hydrogen production rate at various locations were estimated in the W-alloy design. Compared to the liquid breeder Flibe, liquid lithium is the less effective material in attenuating the nuclear flux at the VV and TF coil by a factor of 6–10.

The radioactive waste characteristics of the different components of the machine were evaluated according to both the NRC 10CFR61 [16] and Fetter waste disposal concentration limits (WDR) [15]. According to Fetter limits, the first wall, module wall, blanket, and transitional zone would not qualify for disposal as class C waste. As a matter of fact, the W–5Re alloy produces such a high activity that the first wall would have a WDR that is more than an order of magnitude higher than the class C WDR limits. The high WDR is due to the \(^{186m}\text{Re}\), \(^{108m}\text{Ag}\), and \(^{94}\text{Nb}\) isotopes. Only \(^{186m}\text{Re}\) is a product of nuclear interactions with base elements in the W–5Re alloy.

10.7. Power conversion system

The major incentive for employing high-temperature refractory alloy FW/blanket with helium cooling in this design is to enable direct coupling with a CCGT (Brayton cycle) for high efficiency power conversion. This has the advantage of eliminating an intermediate high-temperature He/He heat exchanger (HX), which would be a significant technical challenge. However, the potential for tritium contamination in the power conversion system (PCS) must be addressed, and appropriate design measures must be taken to prevent further spread of contamination and to facilitate maintenance of PCS components. Fig. 38 shows the effect of FW/blanket inlet temperature variation on PCS performance for the selected outlet temperature of 1100°C. Based on this, the selected gross efficiency for the preliminary design is 57.5%.

10.8. Safety

The use of tungsten as the structural material in this concept poses some safety challenges. Tungsten is a radiologically hazardous material with high decay heat, so we must ensure that the design is such that long-term accident temperatures are low enough that unacceptably large amounts of tungsten are not mobilized during an accident. Our preliminary calculations show that design options exist that result in long-term temperatures below 800°C. Details can be found in the APEX interim report [1].

10.9. Key issues and R&D

We have completed the preliminary design of a helium-cooled refractory alloy FW/blanket design. Many development issues are identified in different areas of the design. The following is a list of key issues, grouped by areas, which will have to be addressed in order to become a viable design:

<table>
<thead>
<tr>
<th>Materials</th>
<th>Irradiated and engineering design material properties of W-alloy.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design</td>
<td>Design criteria for W-alloy.</td>
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<tr>
<td></td>
<td>Fabrication of W-alloy components.</td>
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<td></td>
<td>Minimum cost of W-alloy components including material and fabrication.</td>
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<td></td>
<td>Compatibility between helium impurities and W-alloy.</td>
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</table>

<table>
<thead>
<tr>
<th>Availability</th>
<th>Failure rate and maintenance.</th>
</tr>
</thead>
</table>

| Design          | External coolant piping routing.                                   |
|                 | Structure support to handle thermal expansion.                    |
|                 | High temperature piping.                                          |
|                 | Develop robust high performance fusion power core W-alloy components. |
11. Gravitational flowing Li$_2$O particulates

One of the concepts considered early in the APEX study attempts to eliminate the structural first wall by flowing Li$_2$O particulates directly exposed to the plasma. The concept is called APPLE. The Li$_2$O particulate flow system serves as the coolant and breeder. To be able to handle simultaneously a high neutron wall loading and high surface heat flux, the particulate material for the coolant/breeder must have good thermal conductivity and high temperature capability. The desirable material properties are:

1. Low vapor pressure at high temperature.
2. Low activation.
3. Good tritium breeding capability.
4. Low electrical conductivity.
5. High thermal conductivity.

After reviewing the potential candidates of the available coolant/breeding material, the solid breeder Li$_2$O was identified to have good potential to fulfill most of the requirements.

Since the coolant will be facing the plasma, the low vapor pressure requirement becomes very important. The total vapor pressure over Li$_2$O can be very low. At 1000°C, the combined vapor pressure of all the possible components is less than $10^{-5}$ torr. Therefore, the maximum allowable temperature of the Li$_2$O is set at 1000°C. This high allowable temperature leads to a design with high thermal conversion efficiency.

Fig. 39 shows the conceptual design of the system. The Li$_2$O particulate will be fed to the reactor system through a feed tube by gravitational force. After the particulate enters the reactor, it will be directed toward the inner (IB) and outer (OB) blanket by a solid baffle, made by SiC. Upon entering the IB and OB blanket module, the Li$_2$O will be divided into two separate streams. The stream facing the plasma will be freely dropped by gravitational force, while the flow of the stream inside the blanket will be restricted by an opening at the bottom of the blanket module to slow down the flow. It is important to reduce the flow velocity of the blanket coolant to achieve a high coolant temperature rise for optimum power conversion.

The thermal analysis of the blanket was performed, and the parameters are summarized in Table 12.

The tritium breeding and activation have been calculated. Li$_2$O has very high lithium density, and sufficient tritium breeding can be achieved. With the expected low structural fraction in the APEX design, the tritium breeding will not be a
serious issue. Both Li and O are low activation materials. The only significant activation product from pure lithium is the tritium, which is required for the fueling of the D-T plasma. The activation from oxygen is very low. The only other activation products are from the structural material inside the blanket, and from the shielding material behind the blanket. All the structural materials for this design have shown to be qualified for class C waste disposal. The summaries for the neutronics and activation analysis are summarized in Table 13.

12. Summary of materials considerations and database

12.1. Introduction

The list of structural materials originally considered for the APEX study includes conventional materials (e.g. austenitic stainless steel), low-activation structural materials (ferritic-martensitic steel, V–4Cr–4Ti, and SiC/SiC composites), oxide dispersion strengthened ferritic steel, conventional high temperature refractory alloys (Nb, Ta, Mo, W alloys), Ni-based super alloys, ordered intermetallics (TiAl, Fe3Al, etc.), various composite materials (C/C, Cu-graphite and other metal–matrix composites, Ti3SiC2, etc.), and porous–matrix metals and ceramics (foams). In order to provide maximum flexibility in the design (and to increase the possibility for significant improvements in reactor power density), long-term activation was not used as a defining ‘litmus test’ for the selection of candidate materials.

Due to limitations in resources and time, the materials analysis for APEX quickly focused on refractory alloys due to their higher thermal stress capacity and higher operating temperature capabilities compared to conventional structural materials. However, it should be emphasized that
Table 14
Costs for simple plate products (1996 prices)

<table>
<thead>
<tr>
<th>Material</th>
<th>Cost per kg</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fe–9Cr steels</td>
<td>≤$5.50 (plate form)</td>
</tr>
<tr>
<td>SiC/SiC composites</td>
<td>&gt;$1000 (CVI processing) ~$200 (CVR processing of CFCs)</td>
</tr>
<tr>
<td>V–4Cr–4Ti</td>
<td>$200 (plate form—average between 1994 and 1996 US fusion program large heats and Wah Chang 1993 ‘large volume’ cost estimate)</td>
</tr>
<tr>
<td>Nb–1Zr</td>
<td>~$100</td>
</tr>
<tr>
<td>Ta</td>
<td>$300 (sheet form)</td>
</tr>
<tr>
<td>Mo</td>
<td>~$80 (3 mm sheet); ~$100 for TZM</td>
</tr>
<tr>
<td>W</td>
<td>~$200 (2.3 mm sheet); higher cost for thin sheet</td>
</tr>
</tbody>
</table>

conventional materials may work satisfactorily in some of the APEX concepts (e.g. austenitic stainless steel located behind a thick wall of Flibe). Other promising advanced structural materials (e.g. ODS alloys, intermetallics) should be considered in future analyses.

Numerous factors must be considered in the selection of structural materials, including:

1. Unirradiated mechanical and thermophysical properties.
2. Chemical compatibility and corrosion.
3. Material availability, cost, fabricability, joining technology.
4. Radiation effects (degradation of properties).
5. Safety and waste disposal aspects (decay heat, etc.).

Work by the APEX team focused on the first four items in this list during the initial 18 months of the study, and the key findings are summarized below. More details are presented in Chap. 13 of Ref. [1].

12.1.1. Material costs and fabrication issues

The APEX materials team gathered information on the costs of many of the candidate structural materials. This raw material cost information is summarized in Table 14. The fabrication costs for producing finished products of refractory alloys (particularly W) is known to be much higher than for steels. The group V refractory metals (V, Nb, Ta) are relatively easy to fabricate into various shapes such as tubing, whereas the group VI refractory metals (Mo, W) are very difficult to fabricate. A further issue with all of the refractory metals is joining, particularly in-field repairs. Satisfactory full-penetration welds have not been developed for W, despite intensive efforts over a > 25 year time span (1960–1985). The main issue associated with fusion zone welding of the group V alloys is the pickup of embrittling interstitial impurities (O, C, N, H) from the atmosphere. Experimental studies are in progress in the US to develop satisfactory fusion welds for vanadium alloys.

12.1.2. Overview of thermal stress capabilities of various alloys

The key mechanical and physical properties of high-temperature refractory alloys and low-activation structural materials are summarized in Section 13.3 of the APEX interim report [1]. A thermal stress figure of merit convenient for qualitative ranking of candidate high heat flux structural materials is given by $M = \sigma_U k_{th}(1 - \nu)/\alpha_{th}E$, where $\sigma_U$ is the ultimate strength, $E$ is the elastic modulus, $\nu$ is Poisson’s ratio, $k_{th}$ is the thermal conductivity, and $\alpha_{th}$ is the mean linear coefficient of thermal expansion. In addition, temperature limits (usually determined by thermal creep considerations) can be used for additional qualitative ranking of materials. A rigorous quantitative analyses of candidate materials requires the use of advanced structural design criteria such as those outlined in Section 13.2 of Ref. [1].

The mechanical properties for recrystallized refractory alloys have been used as the reference case for purposes of APEX designs. Fig. 40 shows the ultimate tensile strength for several recrystallized refractory and high conductivity structural alloys as a function of temperature. The mechanical properties of stress-relieved (non-recrystallized) refractory alloys are superior to those of recrystallized specimens, with increases in strength of up to a factor of 2 being typical. However, the possibility of stress- or radiation-enhanced recrystallization of these alloys (along with the likely inclusion of welded joints in the structure) does not allow this strength advantage to be considered for conservative design analyses.
Fig. 40. Temperature-dependent ultimate tensile strengths of recrystallized refractory alloys and high-conductivity structural alloys. Data from Tietz and Wilson [31], Conway [32], Buckman [33], and Zinkle et al. [34].

Fig. 41. The allowable operating temperature range for structural materials based on unirradiated/irradiated mechanical properties, void swelling and thermal conductivity degradation is denoted by the black boxes (see text). Chemical compatibility issues may cause a further restriction in the operating temperature window.
The thermal stress figures of merit vary from ~57 kW/m for a high strength, high conductivity CuNiBe alloy at 200°C [30] to ~2.0 for SiC/SiC at 800°C. Copper alloys are not attractive choices for high thermal efficiency power plants due to their high thermal creep at temperatures above 400°C. The low thermal stress resistance of SiC/SiC is mainly due to the low thermal conductivity in currently available composites (primarily due to a combination of poor quality fibers and imprecise control of the CVI deposition chemistry). The two major classes of low-activation structural alloys, V–Cr–Ti and Fe–8–9Cr martensitic steel have figures of merit of ~6.4 (450–700°C) and 5.4 (400°C), respectively. The refractory alloys offer some advantage over vanadium alloys and ferritic-martensitic steel, even in the recrystallized condition. For example, pure recrystallized tungsten has a figure of merit of $M = 11.3$ at 1000°C, and TZM (Mo–0.5Ti–0.1Zr) has a value of $M = 9.6$ at 1000°C. The alloy T-111 (Ta–8W–2Hf) has the best thermal stress figure of merit among the (non-copper) alloys considered, with a value of $M = 12.3$ at 1000°C.

12.2. Structural design criteria

Most advanced blanket design concepts require the first wall to operate in temperature regimes where thermal creep effects may be important. Therefore, in addition to the usual low-temperature design rules, high-temperature design rules may also have to be applied. We have adopted the ITER structural design criteria (ISDC) as a basis for the design rules to be used in APEX.

Since the design studies under APEX are preliminary in nature, only elastic analysis design rules are included. The design rules are divided into a high temperature section and a low temperature section, depending on whether thermal creep effects are or are not important. The low temperature rules are always applicable. High temperature rules are also applied if thermal creep may be significant. The low temperature design rules include limits associated with necking and plastic instability, plastic flow localization, ductility exhaustion, brittle fracture, ratcheting (cyclic loading), and fatigue. The high temperature design rules include limits associated with creep damage, creep-ratcheting, and creep-fatigue.

12.3. Summary of thermophysical properties (unirradiated and irradiated)

Analytical expressions for the temperature-dependent mechanical and thermophysical properties for five of the structural materials considered for APEX have been derived from least-squares fits of experimental data (Fe–8–9Cr ferritic/martensitic steel, V–4Cr–4Ti, SiC/SiC, Ta–8W–2Hf, and W–10Re) and documented in Chap. 13 of ref. [1]. Radiation-induced void swelling is not anticipated to be a lifetime-limiting issue in the refractory metals due to their BCC structure, although there are insufficient experimental studies to fully establish the void swelling behavior. Radiation hardening and associated embrittlement can have a major impact on all of the refractory alloys. The amount of radiation hardening at low temperatures ($< 0.3 \, T_M$) is pronounced in all of the refractory alloys, even for damage levels as low as ~1 displacement per atom. The amount of radiation hardening typically decreases rapidly with irradiation temperature above $0.3 \, T_M$, and radiation-induced increases in the ductile to brittle transition temperature (DBTT) may be anticipated to be acceptable at temperatures above ~$0.3 \, T_M$ (although experimental verification is needed). Very little information is available on the fracture toughness of irradiated or unirradiated refractory alloys.

12.4. Coolant/structure chemical compatibility

In general, the refractory alloys have very good compatibility with the liquid metals and salts of interest for fusion applications (Li, Pb–Li, Sn–Li, Flibe). Impurity pickup (O, C, N, etc.) is the key engineering issue in most cases for refractory alloys in contact with these coolants as well as for He-cooled concepts.

Formation of volatile oxides can lead to pronounced surface erosion of group VI metals (Mo, W) at elevated temperatures. The evaporation rate increases rapidly up to ~2000 K in both Mo and W. The high-temperature oxidation of Mo and W...
Table 15
Maximum allowable temperatures of structural alloys (bare walls) in contact with high-purity liquid coolants, based on a 5 μm/yr corrosion limit (Sn–Li corrosion limits are based on experimental studies conducted with liquid Sn)

<table>
<thead>
<tr>
<th></th>
<th>Li</th>
<th>Pb–17 Li</th>
<th>Sn–Li (Sn)</th>
<th>Flibe</th>
</tr>
</thead>
<tbody>
<tr>
<td>F/M steel</td>
<td>550–600°C</td>
<td>450°C</td>
<td>400–500°C</td>
<td>700°C (304/316 stainless steel)</td>
</tr>
<tr>
<td>V alloy</td>
<td>~700°C</td>
<td>~650°C</td>
<td>?</td>
<td>?</td>
</tr>
<tr>
<td>Nb alloy</td>
<td>&gt;1 300°C</td>
<td>&gt;600°C(&gt;1 000°C in Pb)</td>
<td>600–800°C</td>
<td>&gt;800°C</td>
</tr>
<tr>
<td>Ta alloy</td>
<td>&gt;1 370°C</td>
<td>&gt;600°C(&gt;1 000°C in Pb)</td>
<td>600–800°C</td>
<td>?</td>
</tr>
<tr>
<td>Mo</td>
<td>&gt;1 370°C</td>
<td>&gt;600°C</td>
<td>&lt;800°C?</td>
<td>&gt;1 100°C?</td>
</tr>
<tr>
<td>W</td>
<td>&gt;1 370°C</td>
<td>&gt;600°C</td>
<td>~800°C</td>
<td>&gt;900°C?</td>
</tr>
<tr>
<td>SiC</td>
<td>~550°C?</td>
<td>&gt;800°C?</td>
<td>&gt;760°C?</td>
<td>?</td>
</tr>
</tbody>
</table>

was analyzed using a thermodynamic model. If boundary layer scattering effects are ignored, the evaporation rate exceeds 100 μm/year at ~1500 K in both materials for 1 ppm oxygen in He at a pressure of 10 MPa. Boundary layer effects may reduce the evaporation rate by several orders of magnitude. The calculations suggest that limitations on mass transport through the boundary layer may reduce the erosion rate to less than 10 μm/year at wall temperatures up to 2600 K in both Mo and W. Although the model does not take into account many of the physical features of real wall-coolant interactions, such as roughness, bends, and temperature variations along the flow, it is reasonable to assume that the evaporation rate of W and Mo will be below a few microns per year, when operated at temperatures as high as 1200–1300°C.

Oxygen pickup in the group V metals (V, Nb, Ta) causes matrix hardening, which in turn produces an increase in the ductile-to-brittle transition temperature (DBTT). The matrix oxygen content must be kept below ~1000 wt ppm in order to keep the Charpy V-notch DBTT below room temperature. Due to the high affinity of the group V metals for oxygen, it is not realistic to avoid oxygen pickup from non-lithium coolants on the basis of thermodynamics. However, the kinetics of the oxygen pickup can be kept acceptably low either by maintaining the temperature below ~0.4 $T_M$ or by keeping the oxygen partial pressure sufficiently low so as to prevent significant impingement of oxygen on the metal surface. A conservative analysis indicates that an oxygen partial pressure of ~10$^{-10}$ torr would be sufficient to keep oxygen pickup to acceptably low levels in group V metals for expected structural material lifetimes (10–50 years).

The experimental database on corrosion of structural alloys in contact with liquid metals and Flibe was reviewed. The refractory alloys have excellent compatibility with liquid lithium up to very high temperatures. The maximum operating temperatures of various alloys in Li, Pb–Li and Flibe is summarized in Table 15. There is a strong need for experimental data on the chemical compatibility of the various structural alloys with Sn–Li and Flibe although several materials appear to be compatible with these coolants at temperatures of interest for APEX. The refractory alloys do not appear to have good compatibility with Sn–Li.

12.5. Summary and conclusions

The estimated minimum and maximum temperatures for several of the structural materials considered for APEX are summarized in Fig. 41. The lower temperature limit is based on radiation hardening/fracture toughness embrittlement ($K_{IC} < 30$ MPa-m$^{0.5}$) due to low temperature irradiation. This embrittlement effect would be expected to occur for damage levels above ~1 dpa. There is a large uncertainty in the lower temperature limit for radiation embrittlement in W due to lack of mechanical properties data at irradiation temperatures above 700°C. The upper temperature limit is based on thermal creep considerations (1% creep in 1000 h for an applied stress of 150 MPa). Depending on the choice of coolant, this
upper temperature limit could be reduced due to corrosion issues. On the other hand, even higher temperatures might be conceivable for applications which have very low applied stress. The corresponding minimum and maximum temperature limits for Fe–8–9%Cr ferritic/martensitic steel are ~250 and ~550°C. The upper temperature limit could be increased by using oxide dispersion strengthened ferritic steel, which has good creep strength to temperatures in excess of 650°C. The recommended minimum and maximum temperature limits for SiC/SiC composites are ~600°C (due to radiation-induced thermal conductivity degradation effects) and ~900°C (due to void swelling concerns), although additional irradiation data are needed to firmly establish these temperature limits.

13. Safety and environment considerations and analysis

Safety and environmental issues are being considered up front in the APEX study as new ideas and designs evolve so that the goal of safety and environmental attractiveness is realized. A comprehensive safety analysis requires detailed designs [35]. Since the objective of APEX is to explore and evolve new ideas, rather than develop detailed designs, the role of safety analysis is somewhat different. Safety analysis is used in two ways: (1) for screening concepts by looking for safety issues that could be ‘show stoppers’, i.e. meeting safety guidelines does not look feasible, and (2) for providing guidance to the design idea developers on areas of improvements to enhance safety and environmental attractiveness.

13.1. LOCA calculations

The 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. A number of concepts were examined 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 [36] used in these calculations was developed to analyze decay heat driven thermal transients in fusion reactors.

LOCA calculations were carried out for four different APEX concepts:

1. He-cooled, refractory alloy first wall/blanket (slowly moving liquid lithium breeder with tungsten alloy structure).
2. APPLE concept (SiC structure with flowing LiO₂ particulate breeder; total blanket thickness of ~40 cm).
3. CLiFF concept (V structure with thin, ~2 cm, liquid breeder).
4. One of the thick liquid (Pocket) concepts (a thick, ~50 cm, layer of liquid breeder flows over a ferritic steel back wall).

The decay heat distributions for the four designs analyzed are shown in Fig. 42. Note that the decay heat is shown per unit volume of the zone, including structure, voids, coolant channels, etc.

The optimal result, from a safety point of view, is when long-term accidents temperatures are adequately low without relying on active (safety-grade) cooling systems. The initial calculations for each design assumed no active cooling. If the temperatures were unacceptably high, various cooling options were then examined. Peak temperatures and the amount of time above 800°C for the APPLE, CLiFF, thick liquid wall, and He-cooled refractory alloy designs are shown in Table 16 (EVOLVE has not yet been analyzed). Because of the large amount of tungsten used in the He-cooled refractory alloy design, active cooling was necessary to keep accident temperatures to an acceptable level. Similarly, it is primarily the Tenelon in the shield that is contributing to the high decay heat in the CLiFF design. Active cooling of the vacuum vessel reduces peak temperatures to 875°C, however temperatures are above 800°C for 3.5 days.

The choice of Tenelon (which is a high manganese steel; manganese has high decay heat) in the shield behind CLiFF is independent of the idea of thin liquid wall, and thus can be easily replaced by another shielding material. Although the peak temperature during the transient for the APPLE design is above 800°C, the duration is less
Fig. 42. Decay heat distribution per unit volume for the four concepts analyzed.

Fig. 43. Volume of the shield, vacuum vessel and magnet components as a function of wall load.
Table 16

<table>
<thead>
<tr>
<th>Concept</th>
<th>Peak temperature (°C)</th>
<th>Time above 800°C (h)</th>
</tr>
</thead>
<tbody>
<tr>
<td>APPLE</td>
<td>1275</td>
<td>1.2</td>
</tr>
<tr>
<td>CLiFF</td>
<td>875(^a)</td>
<td>84(^a)</td>
</tr>
<tr>
<td>He-cooled</td>
<td>800(^b)</td>
<td>&lt;1(^b)</td>
</tr>
<tr>
<td>Thick liquid</td>
<td>675</td>
<td>0</td>
</tr>
</tbody>
</table>

\(^a\) See text; this is due to the shielding material, Tenelon, which can be easily replaced.

\(^b\) With active cooling of the blanket region.

13. Summary

The objective of the APEX study has been to identify and explore novel, possibly revolutionary, concepts for fusion chamber technology that can substantially improve the attractiveness of fusion energy systems. The first phase of the study was carried out in 1998–1999 by a multi-disciplinary team.
<table>
<thead>
<tr>
<th>Concept type</th>
<th>Peak wall load (MW/m²)</th>
<th>FW/blanket structure fraction</th>
<th>Approximate structure replacement time</th>
<th>Reduction in waste volume of FW and blanket components for liquid wall high power density designs relative to ARIES-RS</th>
</tr>
</thead>
<tbody>
<tr>
<td>ARIES-RS</td>
<td>5.5</td>
<td>10%</td>
<td>2.5 FPY</td>
<td>infinite</td>
</tr>
<tr>
<td>Thick liquid walls with no structure</td>
<td>10</td>
<td>0%</td>
<td>n/a</td>
<td>70</td>
</tr>
<tr>
<td>Thick liquid Flibe walls with 4% structure behind walls</td>
<td>10</td>
<td>4%</td>
<td>40 FPY</td>
<td>50</td>
</tr>
<tr>
<td>Thick liquid walls with 1% structure to guide the flow</td>
<td>10</td>
<td>1%</td>
<td>~ 1.5 FPY</td>
<td>10</td>
</tr>
</tbody>
</table>
integrated team of scientists and engineers from 12 US organizations with participation of experts from Germany, Russia, and Japan. A set of goals for the Chamber Technology were adopted to calibrate new ideas and to measure progress. These goals include: (1) high power density capability with neutron wall load \( > 10 \text{ MW/m}^2 \) and surface heat flux \( > 2 \text{ MW/m}^2 \); (2) high power conversion efficiency (\( > 40\% \)); (3) high availability (i.e. low failure rate and fast maintenance); and (4) simple technological and material constraints.

A number of promising ideas for new innovative concepts have already emerged from the first phase of the APEX study. While these ideas need extensive research before they can be formulated into mature design concepts, some of them offer great promise for fundamental improvements in the vision for an attractive fusion energy system. These ideas fall into two categories. The first category seeks to totally eliminate the solid ‘bare’ first wall. The most promising ideas in this category are in the ‘concept rich’ class of flowing liquid wall variations. The second category of ideas focuses on extending the capabilities, particularly the power density and temperature limits, of solid first walls. A promising example is the use of high temperature refractory alloys (e.g. tungsten) in the first wall together with an innovative heat transfer and heat transport scheme based on vaporization of lithium.

14.1. Liquid walls

The liquid wall idea evolved during the APEX study into a number of concepts that have some common features but also have widely different issues and merits. These concepts can be classified according to: (a) the type of working liquid, (b) the thickness of the liquid flow, and (c) the type of restraining force used to control the liquid flow.

14.1.1. Basic principles and concepts

The practical candidates for the working liquid are the liquid metals lithium and Sn–Li (Sn–Li was introduced into APEX because it has very low vapor pressure), and the molten salt Flibe. Many different considerations must be taken into account when assessing the performance of various liquid wall ideas. The hydrodynamics and heat transfer characteristics of high conductivity, low Prandtl Number liquid metal flows will depend heavily on the interactions with the magnetic field. In contrast, low-conductivity, high Prandtl Number Flibe flows will be dominated by turbulence considerations. The \( Z \) number and ionization potential of any vapor generated from the liquid surface will affect significantly the plasma contamination levels. The relative hydrogen solubility in the working liquid will play a significant role in the structure of the edge and the stability of the plasma discharge.

In addition, high conductivity liquid metal (LM) flows have the potential to affect the local magnetic fields and the plasma stability in a potentially positive manner. LM walls appear capable of allowing stable tokamak operation with increased elongation under reactor conditions. Modeling results indicate that the magnitude of improvement can be large with up to a factor of three improvement in stable \( \beta \) (from 5–7\% to 20–22\%) at aspect ratio 4 and 3, respectively. Flowing liquid metals can potentially stabilize resistive wall modes as well allowing higher \( \beta \) steady state equilibria 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.

The selection of the thickness for the liquid wall layer flow (directly facing the plasma and in front of a solid ‘backing wall’) leads to different concepts that have some common issues but many unique advantages and challenges. Both thin and thick liquid walls can adequately remove high surface heat flux. A primary difference between thin and thick liquid walls is the magnitude of attenuation of neutrons in the liquid before they reach the backing wall. The ‘thin’ liquid wall concept is easier to attain, but ‘thick’ liquid wall concepts greatly reduce radiation damage and activation of the structure. Assuming a 200 DPA damage limit for structure replacement, the use of about 40 cm of Flibe or Sn–Li can make the
structure behind it a lifetime component. Furthermore, the volume of the radioactive waste from the FW blanket system is greatly reduced.

Widely different liquid wall concepts are also obtained by applying various forces to drive the liquid flow and restrain it against a backing wall. An example is the gravity–momentum driven (GMD) concept, where the liquid is injected at the top of the chamber at an angle tangential to the curved backing wall. The fluid adheres to the backing wall by means of centrifugal force and is collected and drained at the bottom of the chamber. The criterion for the continuous attachment of the liquid layer is simply that the centrifugal force pushing the liquid layer towards the wall is greater than any destabilizing gravitational force.

Using Flibe as the working fluid, the GMD concept has been modeled with a three-dimensional, time-dependent Navier–Stokes solver that uses the Reynolds Averaged Navier Stokes (RANS) equations for turbulence modeling and the volume of fluid (VOF) free surface tracking algorithm for free surface incompressible fluid flows. Example solutions at 8 m/s inlet velocity demonstrate that a stable, thick fluid configuration can be established and maintained throughout a tokamak reactor configuration. Nevertheless, gravitational acceleration and mass continuity lead to some amount of jet thinning as it proceeds from the top to the bottom of the reactor. Jet thinning can be overcome by increasing the initial jet velocity, and a fairly uniform thick liquid film can be obtained throughout the plasma chamber if the jet is injected at 15 m/s. The thinning can also be minimized by the MHD drag from the Hartmann velocity profile in a flow with conducting toroidal breaks. More analysis of this effect is needed for Flibe.

Numerical analyses were also performed for LM flows in GMD to determine whether or not an insulator is needed for free surface MHD flows, and to define lithium’s initial velocity that enables a uniform thickness to be maintained throughout the plasma chamber in the presence of the toroidal magnetic field. The preliminary analysis based on simplified magnetic field geometries with only toroidal or radial fields shows that the MHD drag effect significantly increases the layer thickness and causes the associated reduction in the velocity. Thus, there is a need of insulators for a free-surface LM flow if a toroidally segmented poloidal liquid metal flow configuration is considered (other clever options may be possible that do not need an insulator). For an insulated open channel, calculations indicate that a uniform 40 cm-thick lithium layer can be maintained along the poloidal path at a velocity of 10 m/s.

Heat transfer calculations indicate that poloidal flow options like the GMD will have a surface temperature rise in the range of 150°C for lithium, and from 25 to 150°C for Flibe (depending on turbulence assumptions) when flowing at 10 m/s. A better understanding of free surface heat transfer (including the hydrodynamics near the free surface/plasma interface) is needed to more concretely determine these values. Variations of the GMD for the low aspect ratio spherical torus (ST) and cylindrical FRC include adding an additional azimuthal (toroidal) velocity to produce rotation. The ‘swirl flow’ results in a substantial increase in the centrifugal acceleration and better adherence to the backing wall, when the wall curvature in the poloidal direction is large and the toroidal curvature is comparable to the poloidal curvature.

The thin wall analog of the GMD is the convective liquid flow first-wall, or CLiFF, concept, where the goal 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 plasma side of the FW. Such a thin layer is easier to control than a thick liquid system, 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 CLiFF class of liquid wall concepts is viewed as a more near-term application of liquid walls, and is suitable for some currently operating plasma devices.

MHD analysis for LM-CLiFF has shown that the MHD drag can be significant if there is a radial magnetic field component — one normal to the free surface. Analyses indicate that a metallic backplate is acceptable with insulated toroidal breaks if the radial magnetic field is no more than
0.1–0.15 T. The acceptable field magnitude would drop to 0.015 T for the case of toroidally continuous flow. Other important MHD issues such as flow across field gradients (1/R dependence of the toroidal field for example), temporal fluctuations during start-up and plasma control will be addressed in the next phase.

Penetrations will be required for plasma-support functions such as heating and fueling. Novel schemes for accommodating penetrations in liquid walls have been proposed. For example, modifications to the back wall topology to guide the flow around elongated penetrations are found to be effective. Computational three-dimensional fluid dynamic simulation results for the CLiFF concept with Flibe show significantly reduced liquid layer disturbance, no splash at the stagnation point, and no unwetted regions downstream the penetration. These results are encouraging and provide an excellent start for studying penetrations in thick liquid walls, where the volume of fluid is much larger.

The electromagnetically restrained (EMR), applicable only to liquid metals, is another example of liquid wall concepts. EMR utilizes a \( \mathbf{J} \times \mathbf{B} \) force field to push the liquid against the backing wall. An injected poloidal current interacts with the main toroidal magnetic field to generate this force, resulting in liquid layer adherence to the back wall at potentially lower velocities than required for the GMD.

Other active control schemes with injected currents have been proposed as well, and will continue to be investigated with new modeling tools being developed for the task.

14.1.2. Motivation for liquid wall research
There are many attractive features of liquid walls that have motivated this research:
- High power density capability:
  Eliminate thermal stress and wall erosion as limiting factors.
  Smaller and lower cost components (chambers, shield, vacuum vessel, magnets).
- Improvements in plasma stability and confinement:
  Enable high \( \beta \), stable physics regimes if liquid metals are used.
- Increased potential for disruption survivability.
- Reduced volume of radioactive waste.
- Reduced radiation damage in structural materials:
  Makes difficult structural material problems more tractable.
- Potential for higher availability.

It is not clear yet that all these advantages can be realized simultaneously in a single concept. However, the realization of only a subset of these advantages will result in remarkable progress toward attractive fusion energy systems.

14.1.3. Key issues for liquid walls
The scientific and engineering issues for liquid walls are many. Of all the potential issues, a number of them stand out as the highest priority for near-term liquid wall research.

1. Plasma–liquid interactions including both plasma–liquid surface and liquid wall–bulk plasma interactions. Plasma stability and transport may be seriously affected and potentially improved through various mechanisms including control field penetration, H/He pumping, passive stabilization, etc. More careful estimates for the allowable amount of liquid evaporation and sputtering need to be obtained and benchmarked.

2. Hydrodynamics flow feasibility in complex geometries including penetrations. The issue of establishing a viable hydrodynamic configuration threatens feasibility for all concepts, but it differs significantly for thick versus thin and for molten salts versus liquid metals. The main issue facing liquid metals is of course that of MHD interactions. Without toroidal axi-symmetry of the flow and field, reliable insulator coatings will be required on all surfaces in contact with the LM layer. Eddy current forces perpendicular to the surface can pull the LM off the surface, 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. For thick liquid walls, the main issues concern the formation and removal of the liquid flow in the plasma chamber, and the accommodation of penetrations.
3. Heat transfer at free surface and temperature control. Liquid surface temperature and vaporization are tightly coupled plasma edge and free surface hydrodynamic problems that require knowledge of the radiation spectrum, surface deformation, velocity and turbulence characteristics. Being a poor thermally conducting medium, the Flibe surface temperature highly depends on the turbulent convection. However, the normal velocity at the free surface as well as the turbulent eddies near the surface can be greatly suppressed. The issue of heat transfer at free surfaces is a serious concern, especially considering that the preliminary plasma-edge modeling predicts a relatively low limit on the surface temperature for Flibe.

14.2. High-temperature solid wall concepts

APEX has also explored ideas for extending the power density and operating temperature capabilities of solid walls. Achieving high power density means that the coolant heat removal capability must be high and the first wall material must have attractive thermophysical properties. Since materials operating at very high temperatures generally have limited strength, such concepts should operate at low primary stresses. This requires that the coolant pressure be as low as possible, and the temperatures throughout the first wall and blanket be as uniform as possible to reduce thermal stress.

Analysis of materials shows that the only structural materials suitable for high-power density, high-temperature operation are refractory alloys. A tungsten alloy, e.g. W–5% Re, was selected as the primary candidate material, with tantalum alloys as the back-up. The minimum and maximum operating temperature for W and other structural materials were estimated. For W, the lower and upper operating temperature limits are about 900°C and 1250°C, respectively, depending on the choice of the coolant and the applied stress. The lower temperature limit is based on radiation hardening/fracture toughness embrittlement due to low temperature irradiation. There is a large uncertainty in the lower temperature limit for radiation embrittlement in W due to lack of mechanical property data at irradiation temperatures above 700°C. The upper temperature limit is based on thermal creep considerations and, depending on the coolant, could be further reduced due to corrosion issues.

Two coolant schemes were evaluated. The first uses helium with the motivation to explore the possibility of using high temperature helium for high-efficiency energy conversion in a gas turbine cycle. The key difficulties with helium cooling are the very high pressure (~12 MPa) and large temperature rise, which push the requirements on the refractory alloy structural material to the range of uncertainty in available data.

A more promising idea is an innovative cooling scheme based on the use of the heat of vaporization of lithium (about 10 times higher than water) as the primary means for heat removal. This idea, called EVOLVE (evaporation of lithium and vapor extraction) was explored in APEX in some detail and will continue to be investigated.

Calculations indicate that an evaporative system with Li at ~1200°C can remove a first wall surface heat flux of >2 MW/m² with an accompanying neutron wall load of >10 MW/m². The system has the following characteristics:

1. The high operating temperature leads to a high power conversion efficiency.
2. The choices for structural materials are limited to high temperature refractory alloys.
3. The vapor operating pressure is very low (sub-atmospheric), resulting in a very low primary stress in the structure.
4. The temperature variation throughout the first wall and blanket is low, resulting in low structural distortion and thermal stresses.
5. The lithium flow rate is approximately a factor of ten slower than that required for self-cooled first wall and blanket. The low velocity means that an insulator coating is not required to avoid an excessive MHD pressure drop.

A preliminary conceptual design was developed and analyzed for EVOLVE. Key issues that need to be addressed in the future in order to assess the potential of the concept include: (1) three-dimensional heat transfer and transport modeling and analyses for the two-phase flow including MHD
effects; (2) feasibility of fabricating entire blanket segments of W alloys; (3) effect of neutron irradiation on W alloys; and (4) analysis of safety issues associated with the high afterheat in tungsten in case of a LOCA.

14.3. Future work

The APEX team has already initiated its efforts for the next phase, which will focus on more detailed exploration of liquid walls and EVOLVE. The effort will include modeling, analysis, laboratory experiments, as well as collaborating with the physics community to conduct liquid wall relevant experiments in existing plasma physics devices.

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