CHAPTER 14: SAFETY AND ENVIRONMENT CONSIDERATIONS AND ANALYSIS

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14. SAFETY AND ENVIRONMENT CONSIDERATIONS AND ANALYSIS

The ultimate goal of the APEX project is to develop designs that make future fusion energy systems more commercially competitive. These concepts must be able to efficiently extract heat from in-vessel systems with high neutron and surface heat loads while satisfying all fusion power technology requirements and maximizing reliability, maintainability, and safety and environmental attractiveness. Safety and environmental issues are being considered up front as designs evolve so that the goal of safety and environmental attractiveness is realized. Designing safety into the concepts as was done in the ITER project (1) results in less complex systems than retrofitting the design to meet safety requirements.

The designs under development in the APEX project are at a pre-conceptual stage, lacking the detail needed for a comprehensive safety analysis. However based on the safety screening criteria described in Chapter 4, we look for safety issues that could be "show-stoppers,", i.e., meeting safety guidelines does not look feasible. In particular, our initial focus has been on the ability of the designs to remove decay heat. The goal here is to ensure that temperatures remain below levels at which oxidation-driven mobilization becomes unacceptable.

14.1 Radiological Hazard of Materials

Materials choice has long been recognized as a method of minimizing safety hazard (2). By avoiding radiologically hazardous (and mobilizable) structural materials such as tungsten and molybdenum alloys, the burden on confinement is minimized (3,4,5). However, tungsten and molybdenum alloys are examples of potential first wall materials that may be able to function adequately under the higher neutron and heat loads in APEX.

14.1.1 Coolants

In most APEX designs, the plasma facing surfaces of these materials are covered with a liquid to protect the structure and/or remove heat. Liquids considered for APEX coolants include lithium, lithium-lead, Flibe, and tin-lithium. Liquid lithium and its activation products do not pose a radiological hazard (assuming the tritium content is kept to reasonable levels). There are safety hazards associated with the production of Po-210 from lead, however studies indicate that these hazards can be controlled (6). SnLi, a "new" coolant material that is under consideration as a liquid surface (7), can pose a radiological hazard because of the tin component. The production of F-18 in Flibe is of concern, and experiments are planned to measure the mobilization of flourine.

14.1.2 Stuctural Materials

The radiological hazard of a material depends on the material, neutron wall loading, and fluence. Figure 14.1-1 shows the radiological hazard of representative APEX materials (molybdenum, silicon, vanadium, tin, and tungsten) in Sieverts per kg (expressed as specific early dose) of irradiated material. These values are based on activation calculations done by Culham Laboratory (8-9), with a neutron wall loading of 4.15 MW/m^2 and an irradiation time of 2.5 years. The doses are based on a ground-level release at 1 km downwind distance using stability class D, 4 m/s wind speed, and no rain. This fluence is lower than that in APEX designs, therefore the relative values of the specific early dose are more relevant than the absolute values. This metric is useful when evaluating the relative hazard of tokamak dust, for example; the dust composition is generally assumed to be the same as the material from which it was produced (10).

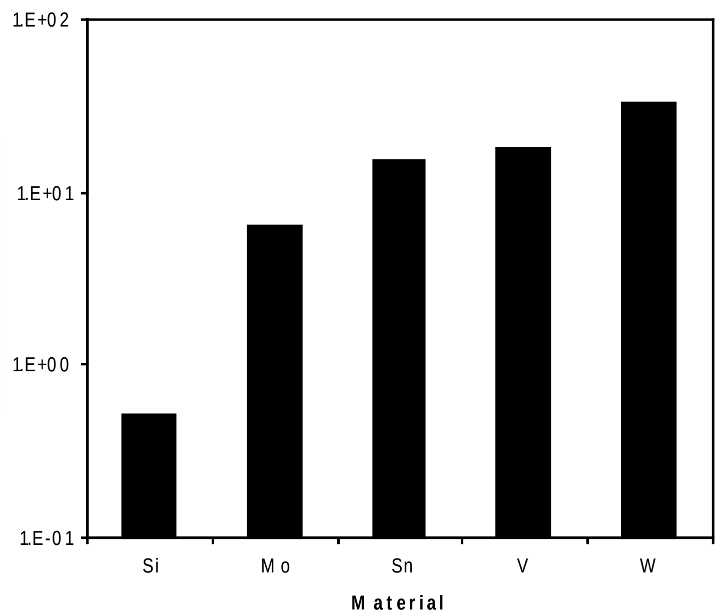

Figure 14.1-1 Radiological Hazard of Representative APEX Materials (Carbon does not contribute significantly to the dose from irradiated SiC)

Oxidation, however, can preferentially mobilize some elements over others. An example of this is that tungsten (with elements such as rhenium and tantalum added to simulate transmutation products) exposed to air, results in mobilization of more rhenium (by mass) than tungsten (see Figure 14.1-2), even though the composition of the material is primarily tungsten. Conversely, when exposed to steam, more tungsten than rhenium is mobilized (see Figure 14.1-3). Additionally, oxidation-driven mobilization is a strong function of temperature, so the temperature profile throughout the accident must be known to assess the hazard due to this source term.

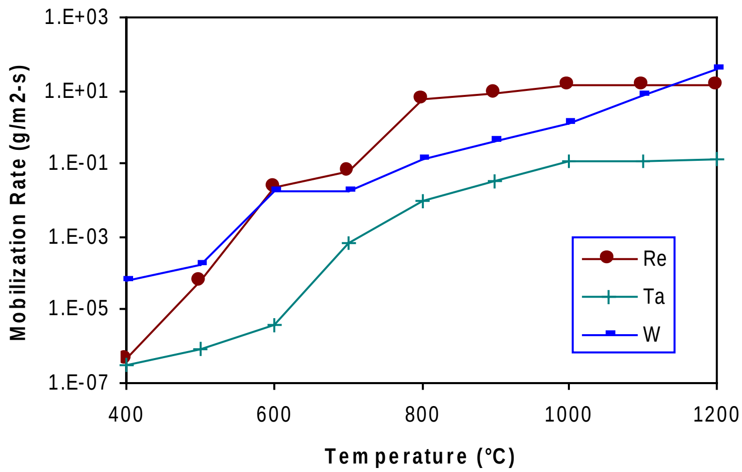

Figure 14.1-2 Mobilization of tungsten alloy in air

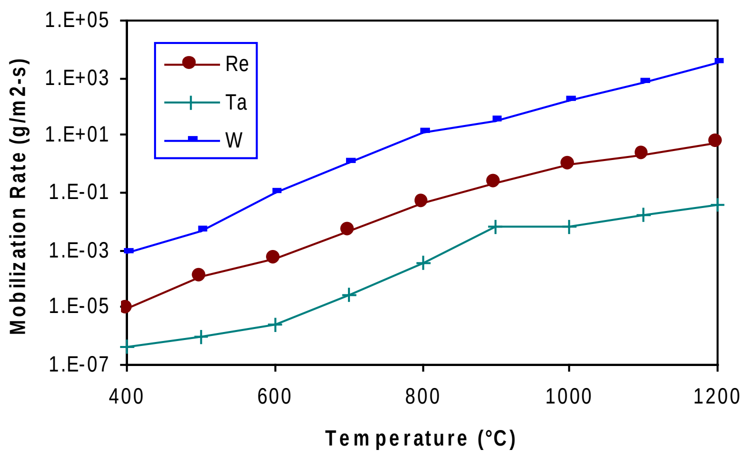

Figure 14.1-3 Mobilization of tungsten alloy in steam

14.2 Loss of Coolant Accidents

The designs under development in the APEX project are at a pre-conceptual stage, lacking the detail needed for a comprehensive safety analysis. However, we have done some parametric studies to identify potential safety concerns, and improve designs to meet safety requirements. We have examined a number of concepts to determine the ability of the design to remove heat from the plasma-facing surface during an accident. If surface temperatures are low enough, mobilization of hazardous material is minimized. The CHEMCON code (11) used in these calculations was developed to analyze decay heat driven thermal transients in fusion reactors. The LOCA calculations specific to each design are described in detail in the chapter on that particular design. We only summarize those results here. The radial decay heat distribution for the four designs analyzed is shown in Figure 14.2-1.

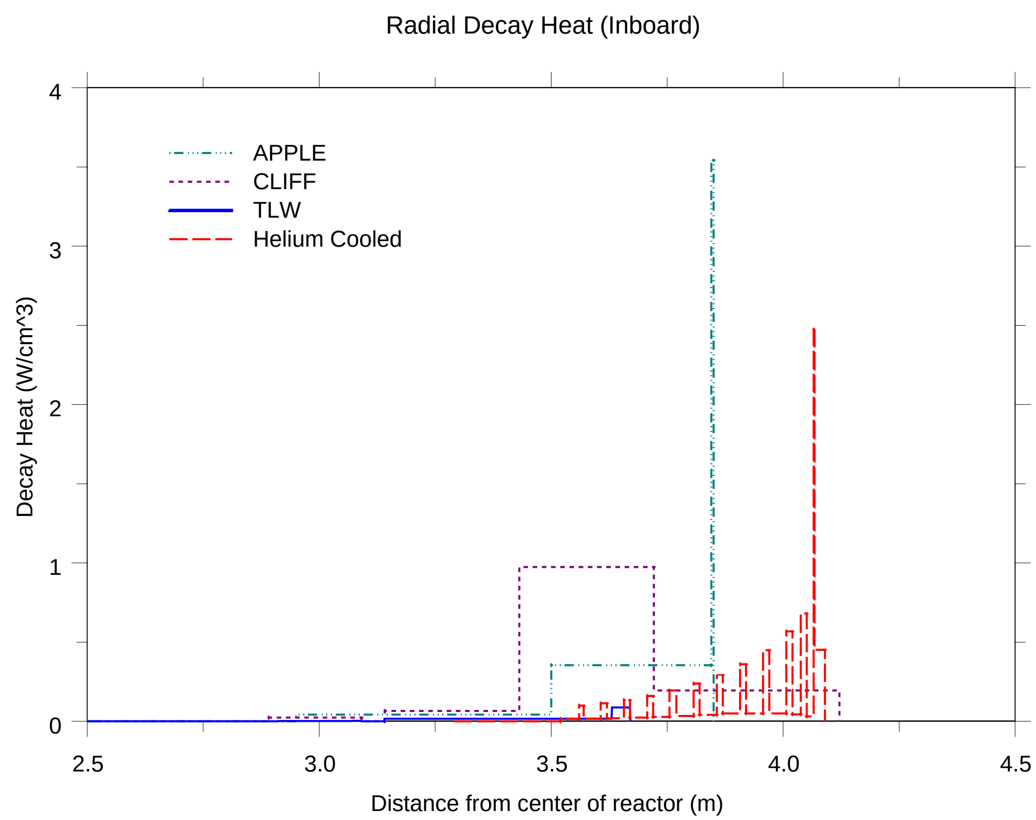

Figure 14.2-1. Radial decay heat distribution for the four designs analyzed.

The optimal result, from a safety point of view, is when long-term accidents temperatures are adequately low without relying on functioning (therefore safety-grade) cooling systems. Our initial calculations for each design assumed no functioning cooling system during the LOCA. If the temperatures were unacceptably high, we then looked at various cooling options. Peak temperatures and amount of time above 800°C for the APPLE, CLIFF, thick liquid wall, and He-cooled refractory alloy designs are shown in Table 14.2-1 (the EVOLVE design will be analyzed at a later date). Because of the large amount of tungsten used in the He-cooled refractory alloy design, 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. Cooling of the vacuum vessel reduces peak temperatures to 875°C, however temperatures are above 800°C for 3.5 days. It may be necessary to either cool the shield during the LOCA, or use a material other than Tenelon (which is a high manganese steel; manganese has high decay heat). Although the peak temperature during the transient for the APPLE design is above 800°C, the duration is less than 2 hours, and the relatively low radiological hazard of SiC makes this acceptable. The temperature in the thick liquid wall design never exceeded 675°C.

Table 14.2-1. Peak Temperature and Time Above 800°C for Apple, CLIFF, and Thick Liquid Designs

Concept	**Peak Temperature (°C)**	**Time Above 800°C (hours)**
APPLE	1275	1.2
CLIFF	875[a]	84[a]
He-cooled	800[b]	< 1[b]
Thick liquid	675	0

[a]With functional cooling of the vacuum vessel during the LOCA (see Section 7.7)
[b]With functional cooling of the blanket region during the LOCA (see Section 11.9)

Although the neutron and surface heat loads are higher in APEX designs than those in conventional fusion designs, these preliminary LOCA calculations indicate that safety criteria (and more specifically, no-evacuation guidelines) can likely be met. For some designs, such as the He-Cooled Refractory Alloy design, this will likely require the use of a safety-grade system to remove decay heat during accidents. It may be necessary to avoid the use of Tenelon in the shield in designs such as CLIFF; in that case, a functional cooling system may not be necessary. For others, such as the Thick Liquid concept, a safety-grade system is probably not necessary. It is desirable to make any such system passive to increase the reliability of the system.

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

14.3 Lithium Chemical Reactivity Issues

Lithium reacts exothermically with both air and water, however the reaction with water is of the most concern because of the subsequent production of hydrogen:

Reaction with water:

$$\text{Excess Li}: \quad 2Li + H_2O \rightarrow Li_2O + H_2 - 160\, kJ/mole - Li$$

$$\text{Excess } H_2O: \quad 2Li + 2H_2O \rightarrow 2LiOH + H_2 - 200 kJ/mole - Li$$

Reaction with air:

$$6Li + N_2 \rightarrow 2Li_3N - 69 kJ/mole - Li$$

$$4Li + O_2 \rightarrow 2Li_2O + 302\, kJ/mole - Li$$

Note that lithium reacts with both the oxygen and the nitrogen in the air. Because the hazards of a lithium-water reaction are too great, water should not be used in a design that includes liquid lithium.

Lithium-lead poses less of a chemical reactivity hazard because the lead "dampens" the chemical reaction (12). Use of water with lithium-lead is acceptable, if the system is low pressure so that a spraying contact mode (which provides a greater surface area for reaction) is avoided. Tin-lithium may behave similarly, however experiments are needed to confirm this, and a series of experiments will begin in FY-00.

There are certain features that should be included in designs with chemically reactive coolants. Examples of these feature include:

(a) Limit inventory of reactive liquid metal as much as possible
2. Design dump tank such that liquid metal spilled will drain quickly into the dump tank and reduce surface area available for reaction,
3. Inclusion of a fast-acting valve that allows the liquid metal loop to drain when a pressure loss is detected (to reduce the lithium spilled),
4. Lithium loop pressure should be as low as possible to reduce speed of lithium spill,
5. Inert atmosphere where spills are likely

14.4 Reliability Issues

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

The reason driving remote maintenance is radiological exposure (13). Tokamaks, such as the Tokamak Fusion Test Reactor, operating with deuterium and hydrogen can have in-vessel radiation fields on the order of 20 to 100 mr/hour (200 microSv/hour to 1 mSv/hour) (14,15). shortly after shut down. The Joint European Torus (JET) that burned tritium had an in-vessel radiation field of 4.3 mSv/hour (16). While the current US regulation for occupational radiation exposure is 50 mSv per year (17), DOE guidelines suggest that exposures should only be 20 mSv/year (18). Then site-specific guidelines typically reduce the exposure level to 10 mSv/year (19) to assure compliance with the DOE guideline. To satisfy the As Low As Reasonably Achievable (ALARA) guidance for radiological exposure, remote tools are used to reduce personnel exposure. Since there will be radiological exposure to workers in other areas of the plant that require hands-on maintenance, such as the ex-vessel parts of the cooling systems (20), keeping in-vessel maintenance doses low allows workers to maintain other plant equipment while keeping the facility collective dose acceptably small. The use of remote equipment rather than hands-on maintenance is needed to meet radiological exposure guidelines for solid wall tokamak designs.

A nonradiological benefit to remote maintenance is that personnel are not exposed to the vacuum vessel interior, where metallic dusts from plasma erosion have accumulated. Therefore, they are not exposed to that chemical hazard. While protective clothing can be worn to protect workers, the clothing – particularly gloves - reduces efficiency (21,22). One study showed that full protective clothing increased maintenance task time on missiles by an average of 45% (23). In-vessel maintenance with anticontamination and chemical protective "bubble suit" clothing, such as worn at JET to protect against beryllium dust exposure (24,25), would also reduce worker efficiency. Time increases from past in-vessel maintenance that required only respirators and coveralls for protection is expected at some value similar to the Waugh and Kilduff (23) study.

14.4.1 Existing Experiment In-vessel Maintenance Times

In 1998, the JET experiment replaced the divertor. The JET team used all remote handling for this activity. JET personnel estimated that the overall calendar time for remote handling is roughly equivalent to the calendar time needed for past hands-on handling (26). The former divertor was a water-cooled structure, and the new divertor is a gas box type (13). The planned time for the changeout was 26 weeks. The task was completed slightly ahead of schedule (27). A 20% time contingency was built into the remote tile exchange plan to account for any off-normal events (equipment breakdowns, stuck parts, etc.). The Mascot manipulator operator accounted for some of the 20% contingency, since over long operating times the concentration needed to operate the manipulator led to fatigue, eye strain, and hand fatigue on the controls (27). Fortunately, there were no failures of the remote handling equipment during the remote tile exchange program.

The JET pumped divertor was installed in 1993, after the preliminary tritium experiment in November 1991. Adding that divertor was a more complicated step than the gas box divertor installation because the pumped divertor required the additions of vacuum pumps, in-vessel magnet coils, and a water cooling system (28). 42,000 person-hours were estimated for that task, taking over a year. Decontamination of tritium and beryllium from the vessel interior required 3 weeks (25), using two-shift operation with three operators per shift.

14.4.2 In-vessel Maintenance Times from Design Studies

The International Thermonuclear Experimental Reactor (ITER) study had some time estimates for in-vessel maintenance. Replacing one divertor cassette was estimated to take 8 weeks, and 6 months for replacing all 60 divertor cassettes (29). The divertor was expected to be replaced three times in the basic performance phase (10 years). Blanket maintenance was assumed to be infrequent, and the maintenance intervention time was estimated to be 8 weeks for one module, 3 months for one toroidal array of modules, and 24 months for replacing all blanket modules. The cryopumps were assumed to need removal 2 or 3 times over the 20 year life of ITER, with a 2 to 3 week replacement time per removal session. The neutral beam injector ion source replacement frequency was estimated to be 1/year, but the time estimate for replacement was not given (29). Time estimates at JET for that process are about one operating shift (30). In-cryostat maintenance on ITER was expected to be infrequent. For in-cryostat maintenance, the warm-up to room temperature and then re-cooling to cryogenic temperature times for the magnets were estimated as one month each (29).

As shown in Table 14.4-1, the ITER reactor design was expected to have modest operational availability, with 100% availability during operating campaigns of several days to a weeks' duration per year (31).

Table 14.4-1. Illustrative ITER Operational Availability Values Typically Expected in the Basic Performance Phase (31)

	Year 1 to 3	Year 4 to 5	Year 6	Year 7 to 10
Availability (%)	---	4	7	10 (a)
Average burn length (s)	---	500	1000	1000
Number of pulses	3200	1600	1000	6000
Average pulse repetition time (s)	---	1700	2200	2200

(a) Including 100% availability for continuous test campaigns of 3 to 6 days with nominal pulse operation scenarios.

Past design studies comparable to the ITER project also had some time estimates for remote maintenance changeouts. The STARFIRE study desired to have a tokamak reactor with 75% operational availability, so the plant could have 91 days of outage per year (32). This outage time was allocated to the reactor and balance of plant as shown in Table 14.4-2.

Table 14.4-2. Outage times estimated for the STARFIRE design.

Balance of Plant Systems		Tokamak Reactor	
Scheduled Downtime	Unscheduled Downtime	Scheduled Downtime	Unscheduled Downtime
37 days	20 days	37 days	34 days

The balance of plant maintenance time was inferred from light water fission power plant performance during the time of the STARFIRE study. Downtime information for power plants operating in the 1990's has been ordered from the North American Electric Reliability Council, and will be added later. The maintenance plan for STARFIRE was to change out four of the 24 first wall/blanket segments each year during a scheduled shut down. The time to perform that task was estimated to be 10 days, adding two more days if beryllium recoating of the first wall was needed. If the replacement was unscheduled, an additional day was added to the task to account for tokamak orderly shut down and subsequent re-start.

The International Tokamak Reactor (INTOR) design study had a planned operations schedule. The schedule was to operate for 9 weeks per quarter year, for a total of thirty-six 5-day operating weeks per year. This schedule would give about 50%

operational availability. Accounting for unscheduled forced outages would yield an availability closer to 25%, or 90 days of unscheduled outage per year (33). The time to remotely replace four of the 12 blanket sectors and recoat the walls was given to be 18 days, and an additional 0.4 day per outage to account for maintenance equipment failures. Limiter replacement (8 plates from two of the four sectors) was estimated to take 6 days. Spampinato et al. (34) discussed the detailed breakdowns that summed to give these time estimates. A comparison of hands-on maintenance to all remote maintenance was also performed for INTOR/Fusion Engineering Device. The conclusion was that the times were nearly identical due to the in-vessel need for remote handling so that workers would avoid excessive radiological exposure. Note that Pick (26) discussed how JET remote and hands-on maintenance times were roughly equivalent, so operating experience strengthens the INTOR conclusion.

An interesting “reverse engineering” approach was to allocate reliability to components based on the allowable downtime to meet plant operational availability requirements (35). Then designers could evaluate the reliability values and determine if they were confident that the components could achieve the required reliabilities. That technique might prove to be valuable for new design approaches.

Since the ITER design was more detailed than the earlier work, the maintenance time estimates are important to consider because they are longer than those cited in the STARFIRE and INTOR studies. The enhanced design detail in ITER is thought to be the reason for the longer time estimates. ITER work in remote maintenance planning and estimating also has the advantages of advanced state-of-the-art remote handling technology, and recent remote maintenance activities in the Tokamak Fusion Test Reactor and the Joint European Torus.

14.4.3 Availability Estimates of Solid Wall Designs

Schnauder et al. (36) have calculated availabilities of blanket and first wall designs for several candidate blanket configurations that are envisaged for the European demonstration reactor (DEMO), as shown in Table 14.4-3.

Table 14.4-3. EU DEMO Blanket/First Wall Availability Results

	He-cooled out-of-tube breeder blanket (BOT)	He-cooled ceramic in-tube blanket (BIT)	Dual coolant (Li-Pb, He) blanket (DCB)	Water-cooled Li-Pb blanket (WCB)
FW availability (%)	90.6	90.6	90.6	90.6
Blanket availability without the FW (%)	93.0 / 95.3 *	93.3	96.8	95.4
FW and Blanket availability (%)	84.3 / 86.3 *	84.4	87.7	86.3

*design with bends / design without bends

From the table, the blanket/first wall availabilities vary from 84.3% to 87.7%, which is a small variance among the designs. The authors stated that absolute numbers were sensitive to modeling assumptions. The results are grouped well enough to consider them as representative values of blanket and first wall availability regardless of specific design details. M. Abdou has calculated that for a plant availability of 88%, considering six major systems (assumed: TF coils, PF coils, Divertor, FW/Blanket, plasma heating, fueling), then each system must achieve on the order of 97% availability. These DEMO reactor values are at least ten percentage points lower than needed to have a fusion reactor meet APEX project availability goals. The current generation of US fission power reactors averaged 81.7% operational availability in 1995 (37), and future plans for advanced fission reactors call for 87% operational availability (38). Therefore, availability in the 80 to 90 percent range is a value to strive for in matured fusion power plants. Early fission power reactors had low availability in the 30 to 70% range, and a growth in availability has been seen in the last thirty years (39). An effort at improving reliability over time (called "reliability growth") to provide longer lived blanket and first wall components would be needed to make solid walls attractive as a fusion power plant component.

14.4.4 Availability for Self-renewing Liquid Walls

The liquid wall designs will require some passive components to function. Some fins, vanes, or baffles to guide flow around penetrations may be needed, and flow introduction nozzles are needed. Some means of flow collection at the base of the machine will also be needed. Some vertical flow guides may also be used. Even though these components are passive and would be chosen from materials compatible with the coolant and environment, there are some concerns that would require periodic inspection. For example, flow nozzles and guide vanes might suffer from flow-induced erosion. Nozzles must not drip any liquid down onto the plasma, and coolant flow velocity and pressure must be constant to avoid flow fluctuations that could degrade nozzle performance or formation of the flowing liquid. Flow collection must not allow any voids or cavities to be introduced into the fluid, since these may cause flow perturbations in the system. Screens in the troughs at the bottom of the vacuum vessel would be used to filter the coolant, and should be visually inspected for debris buildup and for intactness. Such inspections might entail only a remote TV visual inspection of the surface condition, but over years of operation, a more detailed inspection would be needed to satisfy ASME design code requirements (probably the ASME Boiler and Pressure Vessel Code, Division 3, Section XI). Visual inspections might be annual, while detailed radiographic inspections might be scheduled for every ten years. The annual inspections will cause the plant to be down perhaps a few days per year, but that is not a major impact to the plant schedule. The TV inspections might be timed with other work on the facility. The detailed inspections could be performed when the plant is down for other reasons, such as primary coolant piping 10-year in-service inspection, heat exchanger inspection, turbine blade inspection, or other requirements.

Solid wall reactors would also need periodic camera inspections of the walls to verify that there was no plasma damage to the plasma facing surfaces, and that there was no other damage. Other damage might be from electromagnetic forces from unbalanced plasma operation, hot spots created by multifaceted axisymmetric radiation from the edge (MARFE) of the plasma, or other effects.

14.4.5 Summary and Future Efforts on Reliability/Maintenance Issues

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

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

14.5 Waste Disposal Issues

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

The environmental impact of waste material is, however, determined not only by the level of activation, but also the total volume of active material. A tokamak power plant is large, and there is a potential to generate a correspondingly large volume of activated material. The adoption of low activation materials, while important to reduce the radiotoxicity of the most active components, should be done as part of a strategy that also minimizes the volume of waste material that might be categorized as radioactive, even if low level. Waste management strategies have typically concentrated on minimizing the activity of first wall and blanket components where the level of specific activity (Bq/kg) is highest (40).

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

14.5.1 Class C Waste and Recycling

The U.S. 10CFR61 regulations define the following disposal categories for the radioactive waste: geologic disposal, near-surface disposal (Class A, B and C, shallow-land burial low-level waste), and below regulatory concern. In terms of gamma-equivalent dose rate, the waste classifications are determined as shown in Figure 14.5-1:

Clearance or Below regulatory concern – less than 10 mrem/y, or 1 microrem/h,
Near-surface waste – above 1 microrem/h and less than 60 mrem/h,
Deep geologic waste – above 60 mrem/h.

Due to high neutron flux in the fusion reactor first wall and blanket components, some of the fusion materials and impurities will be activated and produce long-lived radionuclides. These radionuclides have half-lives longer than 500 y (e.g., the dominating radionuclides are Nb94 (half-life 20,000 y), Ag108m (418 y), Ir192m (241 y), and Ho166m (1,200 y)). The near-surface disposal has an intruder barrier designed to be monitored for 500 y, primarily dealing with shorter-lived radionuclides with half-lives generally lower than 100 y (e.g., the dominating radionuclides are Ni63 (100 y), Sr90 (29 y), and Cs137 (30 y)). The practicality of near-surface burial disposal appears to be questionable when large quantities of fusion waste are to be managed in the next century.

Recycle and reuse of fusion materials is an attractive option to reduce the waste quantity. Remote-handling recycling (RHR) is necessary to handle the modestly activated materials. The current dose rate limit (which needs to be reexamined based on current remote handling technology) is assumed to be about 1 rem/h.

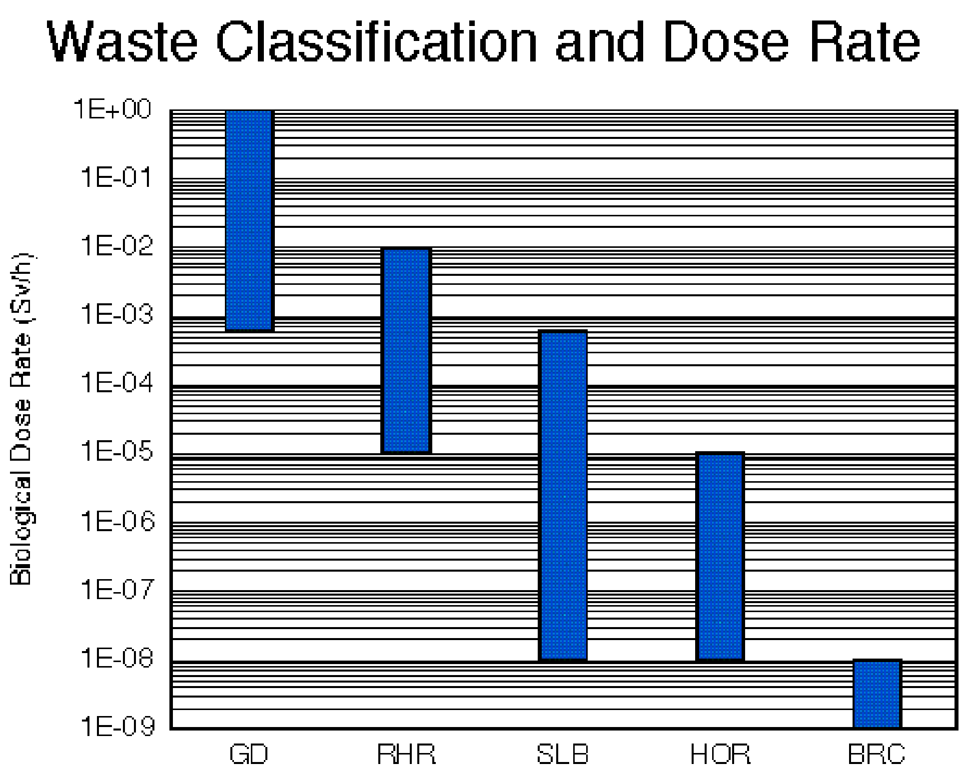

Figure 14.5-1. (GD - Geological Disposal, RHR - Remote Handling Recycling, SLB - Shallow Land Burial, HOR - Hands-on Recycling, BRC - Below Regulatory Concern)

The present fusion waste management strategy in the US is to minimize the waste hazard below the level of near-surface burial. As shown in Table 14.5-1, the waste volume generated in a fusion power plant such as ARIES-RS (1 GWe) is about 1,400 m*3 during the 40 full power year lifetime (because there are complete designs for ARIES, it is used often in this section for comparison). This waste volume is primarily from the in vessel and vacuum vessel components. About 25% of it is due to the replacement components. After 50 years of cooling, this discharged waste can be managed as near-surface burial waste based on the present regulations. There is additional 600 m*3 of ex-vessel components and magnets to be decommissioned from the power plant. However, after 50 years of cooling, the activation level of these components drops below the regulatory concern and these components can be cleared from the regulatory control.

Table 14.5-1. Comparison of Waste from ARIES-RS and fission

Plant	Waste Classification	Reactor Components in Plant		Total Waste upon Decommissioning		Waste due to component replacement			
		Mass (tons)	Volume (m*3)	Mass (tons)	Volume (m*3)	Mass (tons)	% of total	Vol. (m*3)	% of total
ARIES-RS	Class C	7240	1030	9800	1400	2560	26	366	26
LWR	Deep Geological Disposal	100	10	1320	132	1220	92	122	92

Recycling option will allow the in-vessel components and vacuum vessel to be reused after re-fabrication. To first order, recycling of in-vessel components can be performed until the activation exceeds the assumed remote-handling limit of 1 rem/h. This corresponds to 16 recycles before disposal for ARIES-RS. The volume upon decommissioning in this case is only 88 m*3 which is 20 times less waste than in the case of no recycling. Recycling will reduce the overall waste volume but will also increase the hazard of the recycled material because of buildup of certain impurities via reuse in a fusion machine (e.g., Nb-94). The 88 m*3 of waste following recycling would have to currently be disposed of via deep geological burial (e.g., Yucca Mountain). Thus, a strategy to dispose of this material once recycling is no longer feasible has to be established.

14.5.2 <u>Influence of Higher Wall Load</u>

For a given power output, the higher wall load afforded by a liquid first wall and blanket can reduce the volume of waste from the first wall and blanket because of the reduction in size of the FW/blanket, the lower structural content of liquid wall designs, and the potential for increased lifetime of a liquid blanket. Based on ARIES-RS, the FW/blanket usually represents about ~ 20-25% of the total waste volume from the plant after 40 full power years of operation (based on the 2.5 year lifetime of the FW and 7.5 year lifetime of the blanket). The degree to which liquid walls can reduce the volume of FW/blanket waste depends on the amount of structural material in the FW/Blanket system, the wall loading, the thickness of the liquid, and the lifetime of the structure in the FW/blanket. To first order,

$$V \propto A\,\delta\,\chi\,N,$$

where V is the total waste volume, A is the surface area of the first wall, δ is the thickness, χ is the structure fraction in the FW/blanket and N is the number of planned replacements of the structure for the 40 years of plant operation. The surface area is related to the power, P, and wall load, Γ, via the equation $A = P/\Gamma$. Therfore, a higher wall load would allow for a reduced FW surface area, A, which would translate into a smaller volume of the FW/blanket for a given power. However, because of radiation damage to structure in the FW/blanket, such components are not lifetime and must be replaced periodically. The number of replacements is given by $N = 40/(L/\Gamma)$ where L is the lifetime assumed to be ~ 15 MWyr (~ 150 dpa). In this case, higher wall load leads to more frequent replacement because of the greater rate of fluence accumulation. This would increase the volume. As the thickness of the liquid blanket increases and the structural content of the blanket, x, decreases, there is less activated metal and thus some reduction in waste volume is possible. The exact value of volume reduction depends on design details. Combining all of these factors yields the following scaling relationship for the volume of waste for replaceable components like the FW and blanket:

$$V \propto \chi/L$$

This relationship is true when comparing two conventional blankets and it says that for such a blanket, wall load does not matter since the benefit of reduced surface area is offset by the increased replacement frequency. Thus, all that matters is what the structure content of the blanket. Thus, as shown in the Table 14.5-2, for the case of a complete thick liquid blanket with no structure in the first wall/blanket, the waste volume reduction was considered by some to be infinite. Furthermore, if structures such as nozzles, flow baffles or guides are needed to maintain the flow pattern, then the reduction will be much less. For example if only 1% structure is needed, its limited lifetime will require replacement and then the reduction in FW/blanket waste volume is a factor of 10 relative to a conventional design such as ARIES-RS. In the case of a thick liquid Flibe blanket (~ 50 cm thick) the scaling relationship above is not exactly correct. If one can develop a design in which all of the structure is behind the liquid then there is additional gain in waste volume reduction. For the case of 4% structure behind the liquid, the thick liquid reduces the radiation damage so that this structure is lifetime. In this case, the lifetime does not depend on the wall load and only one replacement is expected. Thus, the waste volume reduction relative to ARIES-RS is about a factor of 70. However, the material behind the blanket could just as well be considered the shield and thus the comparison should not be taken too far.

The values depend on the assumed lifetime of the conventional solid wall, and those estimate should be looked at further. If solid first walls and blankets can be recycled and reused then that material is no longer waste and thus the waste volume from the different concepts would be very similar. (It is important to note that we have not examined the activation and resulting waste streams from the breeding material in this comparison. For the liquids, we would have to examine the waste stream from Li, Flibe, LiPb, or LiSn and for solids it is important to examine the waste streams from the solid breeder (e.g., lithium zirconate, silicate, titanate or aluminate) and the Be neutron multiplier.)

Table 14.5-2 Rough order of magnitude comparison of FW/blanket waste volumes for different concepts

Concept type	Peak Wall Load MW/m^2	FW/Blanket Structure Fraction	Approximate Structure Replacement Time	Reduction in waste volume of FW and blanket components for liquid wall high power density designs relative to ARIES-RS
ARIES-RS	5.5	10%	2.5 FPY	
Thick Liquid walls with no structure	10	0%	n/a	infinite
Thick Liquid Flibe walls with 4% structure behind the liquid	10	4%	40 FPY	70
Thick liquid walls with structure to guide the flow	10	1%	~ 1.5 FPY	10

In addition to a reduction in the FW/blanket waste volume, higher power density designs result in a more compact machine which reduces the volume of the shield, vacuum vessel and magnet components. Figure 14.5-2 plots the volume of these components as a function of wall load. Comparison of volume of these components at the 10 MW/m^2 value with the volume at the ARIES-RS value of 5 MW/m^2 shows a reduction in overall volume of these components by about 50%. This value assumes that all of these components are permanent lifetime components. For permanent components the volume scales inversely with the wall load. However, in ARIES-RS, part of the shield is not a lifetime component. In this case, the overall volume reduction afforded by high wall load might be reduced to about 30%.

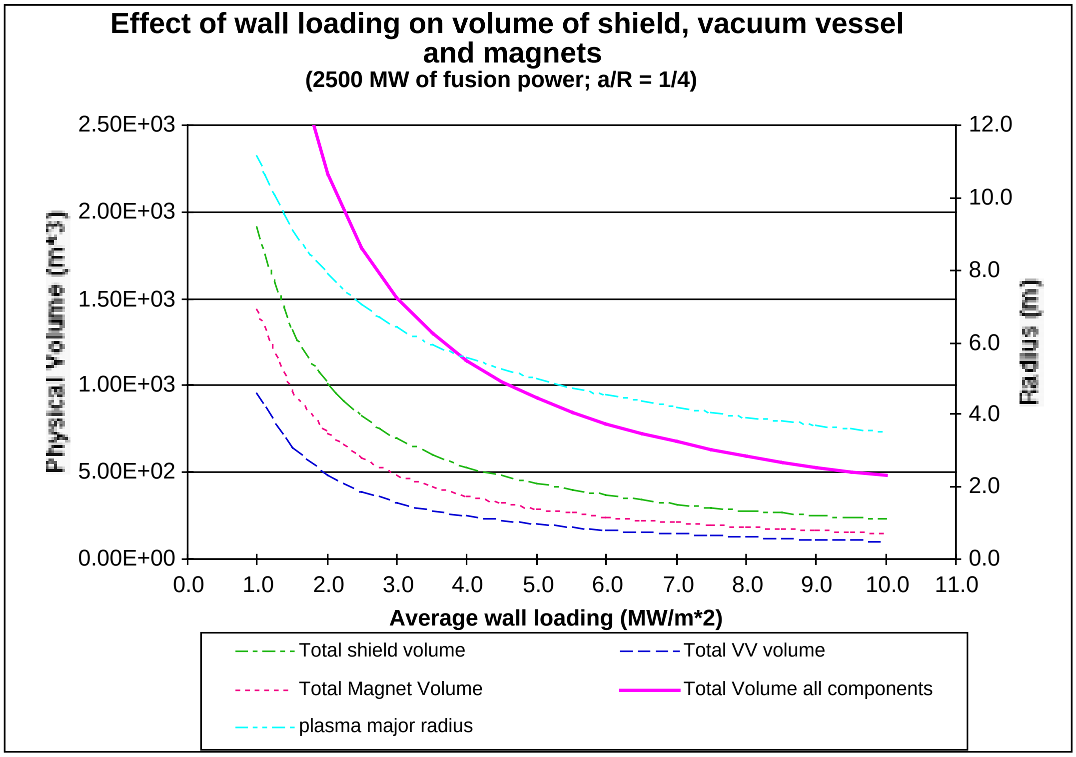

Figure 14.5-2 Volume of the shield, vacuum vessel and magnet components as a function of wall load.

14.5.3 Waste summary

A waste management strategy focused solely on low activation materials does not address the entirety of the radioactive waste picture for fusion. We recommend a strategy that is balanced with respect to minimizing both the hazard (via low activation materials) and the volume (via reduction of ex-vessel activation). As such we propose the following minimum design goals:

a) To reduce the overall radioactive waste volume by limiting vessel/ex-vessel activation so that the bulkier large volume components be cleared or recycled for re-use.
b) To minimize activated material in a fusion plant that cannot be cleared or recycled

A serious study of the economics and technical tradeoffs and the environmental impact associated with recycle is needed to determine the efficacy of this approach and the impact on the environmental picture for fusion. Such a study should examine the economics of recycling and the criteria used for recycling. It is also important to understand the tradeoffs associated with volume reduction via recycling versus increasing the hazard of the waste because of impurity buildup via reuse in a fusion machine.

In addition to their improved performance potential via high wall load and high efficiency, high power density/high wall load concepts offer important advantages relative to the overall volume of activated waste in a fusion machine. Liquid wall concepts with higher wall loads can reduce the volume of FW and blanket structural material wastes somewhat relative to conventional solid wall concepts. Furthermore, the higher wall load produces a more compact machine, which in turn reduces the volume of the bulkier activated components (e.g., shield, VV, and magnets) by 30 to 50%.

14.6 Safety andEnvironment Summary/Future Direction

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

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

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