Materials Flow, Recycle and Disposal for Deuterium— Tritium Fusion

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December 1978

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MATERIALS FLOW, RECYCLE AND DISPOSAL FOR
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ABSTRACT

The flow of materials to and from a deuterium-tritium fusion power plant is investigated. Three tokamak conceptual reactor designs are described and materials requirements are detailed. Various process options are considered for the reprocessing segment of the materials cycle. A conceptual materials resource cycle is selected from these options and details of a conceptual reprocessing segment are described. Environmental control technology considerations associated with the reprocessing segment are outlined.

EXECUTIVE SUMMARY

As with other energy technologies in current and anticipated future use, the environmental impact of fusion facilities associated with the complete materials cycle may be more significant than the impact of the power plant itself. The complete materials cycle includes facilities required for mining, milling, refining, and fabrication of power plant components and fuel, as well as those required for materials processing for re-use and waste disposal. Environmental effluents resulting from operation of these facilities should be considered as part of the impact of utilizing that technology.

This report presents a material balance for fusion technology, developed around three conceptual tokamak fusion reactor designs. The study considers only the materials at the power plant, and the amounts of materials needed, radioactivity levels and environmental control aspects are discussed.

The designs considered are deuterium-tritium fueled tokamak reactors using helium as the primary coolant, liquid lithium metal or solid lithium compounds as fertile material for in-reactor tritium breeding, and stainless steel or nickel alloy as the structural material. The quantity of materials required for plant construction is determined. Annual makeup quantities and outflow of materials are evaluated based on estimates of component lifetimes in the fusion radiation and thermal environment.

Radioactive material inventories in a fusion power plant include tritium in the fuel stream, tritium-contaminated equipment, and neutron activation products. Tritium is present as stored fuel in metal hydride form, and in the fuel input stream, plasma, fuel effluent, effluent separations system, and breeding modules. Tritium-contaminated equipment includes the complete fuel and heat transfer systems and blanket, absorption beds, fuel storage equipment, pump fluids, and steam system resins. All equipment near the reacting plasma will become activated by neutron reactions. Activation products will be generated in the blanket and first wall, neutral beam injectors, magnets, and reactor building air. Activation product-contaminated systems might include the primary coolant system, vacuum systems, and reactor building air.

The materials resource requirements for deployment of fusion power plants on a large scale depend to some extent on the degree to which materials can be recycled. The disposition of reactor components after they are removed has been investigated. All component removal operations must be performed remotely because of high radiation levels from activation product decay. The degree to which used blanket materials can be recycled is dependent on three factors: the physical state of the lithium fertile material, the ease of disassembly, and size of blanket modules. Liquid lithium can be drained and recycled. Solid lithium compounds, neutron-multiplying material such as beryllium and lead, and graphite reflectors can be removed and recycled if the blanket module design lends itself to straightforward disassembly. Steel or nickel alloy structural materials are highly radioactive and must be stored at least for several decades before recycle is viable. Small blanket modules from which lithium, graphite, and neutron-multiplying materials have been removed may be crushed or melted or minimize transportation and storage volume. Large modules or those which cannot be disassembled must be shipped and stored intact.

Consideration of environmental control technology focused on tritium control. Tritium is the most significant effluent associated with a first-generation fusion power plant. By maintaining the blanket at operating temperatures for several hours up to a day after shutdown by suitable throttling of the steam generator valves, a significant reduction in tritium inventory in the blanket breeder material is achieved.

Another consideration in limiting tritium effluents resulting from fusion reactor operation is to control the releases to the reactor building. A large vacuum containment structure encompassing the reactor could prevent most of the tritium from escaping and provide a favorable environment for reactor operation.

Environmental control measures that are necessary vary according to the type of breeder material used. Solid breeder compounds pose some hazard in the reprocessing operations, but possess low tritium inventories making them amenable to outgassing after reactor shutdown. Liquid lithium is corrosive

to many structural materials and requires that environmental control measures be taken in event of a spill or lithium fire. On the other hand, neutron multiplication and lithium enrichment is not required to obtain a favorable tritium breeding ratio with liquid lithium.

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MATERIALS FLOW, RECYCLE AND DISPOSAL FOR DEUTERIUM-TRITIUM FUSION

I. INTRODUCTION

Development of fusion power holds the promise of a long-term solution to current problems of world energy supply. Nuclear fusion is expected to be a safe, environmentally attractive, economical process for which fuel resources are freely available to all nations in virtually unlimited quantities. It produces no chemical combustion products, does not inherently involve any radioactive by-products, and neither requires nor produces material which could contribute to proliferation of nuclear weapons. Commercialization of electric power based on thermonuclear fusion is a major goal of the U.S. Department of Energy (DOE) and of a number of other nations throughout the world.

As with any energy technology, effluents are released during the generation of power. Careful investigation of environmental effluents will reveal ways in which to reduce the consequences of effluent release. For fusion based on the deuterium-tritium reaction, attention has been focused on the release of tritium from the power plant. Tritium is expected to be the most significant environmental effluent from the power plant site and the studies of tritium containment technology being funded by DOE's Office of Fusion Energy should allow development of technology to limit effluent release to acceptable levels by the time fusion power plants are built. Environmental effluents associated with the rest of the fusion materials resource cycle have not yet been addressed.

A research program has been initiated at Pacific Northwest Laboratory (PNL) to identify environmental control technology (ECT) requirements associated with all aspects of the deuterium-tritium fusion materials resource cycle. The complete materials resource cycle is being investigated, from raw materials extraction to ultimate waste disposal. Early identification of ECT needs

will ensure that environmental control considerations will not limit the development of fusion energy. Availability of information regarding potential environmental impacts should also strengthen the influence of environmental considerations in decisions affecting the direction of the fusion technology program.

The first step of the PNL program to identify ECT requirements is to develop a generic description of the complete materials resource cycle. Since the quantity of materials required for construction and operation of a fusion power plant will depend on a large extent on how much material can be recycled, the focus of the investigation has been on power plant materials requirements and recycle/disposal options. This report summarizes an investigation of materials requirements and disposition of spent reactor equipment for a particular type of deuterium-tritium fusion power plant. Future reports are planned to address the rest of the materials resource cycle, potential environmental effluents and release rates from each facility, and alternative fusion reactor types.

DEUTERIUM-TRITIUM FUSION POWER GENERATION

The fusion reaction likely to be exploited in the first commercial power plants is the deuterium-tritium reaction:

This reaction requires initial kinetic energy of the reacting species, which is provided in a reactor by heating the fuel to extremely high temperatures in excess of 5 keV (55 million degrees Kelvin). At these temperatures the fuel must be confined in a vacuum to prevent severe thermal conduction losses to solid materials. The fuel is confined to locations away from the walls of the reaction chamber by powerful magnets.

A schematic of a magnetic fusion power plant is shown in Figure 1. At the temperatures required to achieve fusion energy gain, the deuterium-tritium fuel is a plasma, confined in a vacuum by magnetic fields. High

FIGURE 1. Schematic of a Magnetic Fusion Power Plant

energy neutrons produced in the plasma are absorbed in the surrounding blanket material, heating up the blanket. Thermal energy is removed from the blanket by the primary coolant, which transfers the heat to steam to run a turbine and generator.

The plasma fuel is a mixture of deuterium and tritium. Deuterium is a natural isotope of hydrogen which can be economically extracted from water. It exists naturally in water in quantities which ensure adequate fuel supplies essentially forever. Tritium is an unstable isotope of hydrogen which exists naturally in very small amounts. It must therefore be produced in the power plant in self-sustaining quantities. This is accomplished by incorporating lithium into the blanket. Neutrons interact with lithium to produce tritium and helium. The tritium is then extracted from the blanket where it is combined with unconsumed deuterium and tritium from the plasma exhaust and injected into the plasma.

The blanket is surrounded by a radiation shield to protect the magnets from neutron damage and radiation heating. A combination of normal and superconducting magnets is used. A biological shield protects personnel from neutron and photon radiation.

There are a number of physical concepts being investigated for fusion plasma confinement. The approach being pursued most aggressively throughout the world is the tokamak concept. For the purposes of this study the fusion power plant was considered to be a tokamak operating on the deuterium-tritium-lithium fuel cycle. The primary coolant considered was high-pressure helium and the predominant reactor structure material was stainless steel or a nickel alloy.

A number of conceptual design studies have been made of demonstration tokamak power plants. Many of these have been developed around the use of alternative structural materials such as aluminum, niobium or molybdenum, or of alternative coolants such as liquid lithium or molten salt. At least three conceptual designs have been developed for tokamaks with helium coolant and steel or nickel-alloy structures. These designs were investigated in the present study. The General Atomic Demonstration reactor study⁽¹⁾ (GA-DEMO)

represents a compact tokamak utilizing solid lithium breeding material and Inconel 718 structures. The UWMAK-II reactor design⁽²⁾ represents a larger tokamak utilizing solid breeders and stainless steel structures. The ORNL-DEMO⁽³⁾ represents a tokamak utilizing liquid lithium breeding material and stainless steel structures. These designs were reviewed as the basis of a design envelope of possible fusion power plants to determine ECT needs. The purpose of this study was not to compare the three designs or to improve them.

The major facilities associated with the materials resource cycle for deuterium-tritium fusion are shown schematically in Figure 2. Steel constituents and copper or aluminum for the magnets are mined, refined and milled before fabricating the blankets. Lithium is mined and refined before being incorporated into the blanket. Some designs also call for isotopic enrichment of lithium to increase the ^6Li content. Fabricated blanket modules are inserted into the reactor. Helium, used as the primary reactor coolant as well as the superconducting magnet coolant, will probably be extracted from natural gas wells. Deuterium is produced by heavy water extraction and electrolytic decomposition. At the end of the blanket module lifetime (several years) the highly radioactive modules are removed from the reactor and new modules inserted. Some part of the blanket modules can be reprocessed, while other parts must be placed in long-term radioactive waste storage.

This study investigates the segment of the materials resource cycle shown on the right side of Figure 2. The segment, consisting of the reactor, reprocessing operations, and radioactive waste management, was chosen as the subject for this study for two reasons. First, the degree of materials recycle which is possible greatly affects the quantity of materials required in the structures and lithium segments. This should therefore impact the quantity of environmental effluents from other areas of the fuel cycle. Second, the reactor, reprocessing operations, and radioactive waste management are also the areas for which an environmental analysis can have the greatest influence

FIGURE 2. D-T Fusion Materials Resource Cycle

on technology program decisions. Careful design and materials selection can significantly affect the degree to which materials can be recycled, as will be shown in the later chapters of this report.

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II. SYSTEM DESCRIPTIONS

The components immediately surrounding the plasma chamber of a tokamak power plant are shown schematically in Figure 3. The toroidal plasma is surrounded by a layer of vacuum, followed by the first wall. The first wall serves as protection against the radiation from the plasma. It may or may not be an integral part of the surrounding structures or the vacuum boundary. Outside the first wall is the blanket, which absorbs the neutron energy, breeds tritium, and contains the primary coolant. A radiation shield lies outside the blanket.

Other equipment in the nuclear island includes the toroidal field coils, neutral beam injectors, vacuum system, and coolant loops. The toroidal field coils contain superconducting magnets which provide the toroidal field required for plasma confinement. The neutral beam injector provides heat to the plasma in the form of energetic neutral atoms, and may also be used for plasma fueling or vacuum containment. The vacuum system maintains the required vacuum and removes plasma impurities and spent fuel. The primary coolant, which is high-pressure helium in the cases considered here, cools the blanket and transfers heat to the steam generator which is located externally. The entire nuclear island is supported by steel structures.

REFERENCE SYSTEMS

The envelope of fusion power plant systems used for this investigation is determined by consideration of three conceptual tokamak demonstration reactor designs. The fuel is a mixture of 50% deuterium and 50% tritium. Lithium is the tritium breeding material; high-pressure helium is the primary coolant. The reactor structural material is stainless steel or another Fe-Cr-Ni alloy. Plasma heating and fueling are supplemented by neutral beam injection. The individual design parameters are presented in Table 1.

FIGURE 3. Tokamak Nuclear Island Components

TABLE 1. Reference System Parameters

	GA-DEMO ⁽¹⁾	UWMAK-II ⁽²⁾	ORNL-DEMO ⁽³⁾
Average Power Level (MWt)	1676	4000	1615
Plasma Major Radius(m)	7.0	13.0	6.0
Plasma Minor Radius (m)	1.8	5.0	1.5
First Wall Radius (m)	2.2	5.5	2.1
First Wall Neutron Loading (MW/m ²)	0.97	1.16	3.0
Total First Wall Neutron Flux (neutrons/cm ² /s)	1.5x10 ¹⁵	5.4x10 ¹⁴	9.9x10 ¹⁴
Primary Structural Material	Inconel 718	316 SS	SS
Fertile Material	Li ₇ Pb ₂ / Li ₄ SiO ₄	LiAlO ₂	Liquid Li
Lithium Enrichment (% ⁶ Li)	7.5/90	90	7.5
Neutron Multiplying Material	Pb contained in Li ₇ Pb ₂	Be	None
Blanket Replacement Period (yr)	4	2	6

GA-DEMO

The GA-DEMO design study produced a conceptual design of a doublet tokamak demonstration plant to identify significant technological problems which must be resolved "and to provide an assessment of other important factors such as economic viability, safety and environmental impact."⁽¹⁾ The reactor is designed to produce a peak power of 1764 MWt, resulting in a net electric power of 611 MWe during operation. The first wall is a silicon carbide liner to minimize high-Z impurities in the plasma. The first wall neutron loading is about 1.0 MW/m². The blanket is composed of two zones: the inner zone contains powdered Li₇Pb₂ to breed tritium and multiply the number of neutrons, while the outer zone contains powdered Li₄SiO₄, enriched to 90% in ⁶Li to breed more tritium. The blanket structural material is Inconel 718. The design temperatures are high -- 600-650°C at the first wall, 600°C in the inner blanket zone, and 700-800°C in the outer blanket zone. Maximum first wall surface temperatures can be as high as 1440°C. The primary coolant is helium, which is used to generate steam for a single-reheat supercritical steam system.

Plasma fueling and impurity control is accomplished with a low temperature plasma boundary which flows over the plasma surface from the top of the Doublet to the bottom where it is scraped off with impurities and pumped out of the chamber.

UWMAK-II

UWMAK-II is a conceptual design study of a tokamak reactor with a circular plasma cross section.⁽²⁾ The design is considerably larger than the others discussed here. The peak power level is 5000 MWt, producing an average electrical power of 1716 MWe. The plasma major and minor radii are 13.0 and 5.0 m, respectively. The first wall and blanket structural material is 316 stainless steel with a carbon curtain. The breeding material is powdered LiAlO_2 , enriched to 90% ^6Li . A beryllium metal neutron multiplier is used. The first wall neutron loading is 1.16 MW/m^2 . First wall and blanket operating temperatures are 450 to 650°C. The primary coolant is helium, which is used to generate steam via a liquid sodium intermediate cycle. Neutral deuterium beam injection is used for auxiliary plasma heating. Fueling is accomplished with injection of deuterium-tritium pellets. A double-null poloidal field divertor is used to maintain plasma purity and vacuum conditions.

ORNL-DEMO

The ORNL-DEMO conceptual design⁽³⁾ allows a higher first wall neutron loading, 3.0 MW/m^2 , than the other designs considered. It is a tokamak design with an elliptical plasma cross section. The power level is 1700 MWt during plasma burn, with a 95% duty cycle. The first wall and blanket structural material is austenitic stainless steel, which is assumed to be Type 316 for this study. The blanket vessel is filled with liquid lithium of natural isotopic abundance for tritium breeding. No additional neutron multiplier is needed. The ORNL interim report⁽³⁾ suggested a molten salt primary coolant, which is a mixture of potassium nitrate, sodium nitrite, and sodium nitrate; the reference design used in the current investigation differs from that suggested in the design study. Helium is considered as the primary coolant to permit a better comparison of the three systems. Blanket temperatures are 400 to 500°C. Plasma impurity control is accomplished with a bundle divertor. Auxiliary plasma heating is accomplished by neutral beam

injection of deuterium. Plasma fueling is accomplished through a combination of edge recycling and pellet injection.

BLANKET DESIGNS

GA-DEMO Blanket

The GA-DEMO blanket design uses Inconel as the module structural material and incorporates solid lithium compound breeding regions. The module design chosen allows some contact maintenance external to the shield and attempts to reduce the remote handling and maintenance of large, bulky modules. Although large module designs require less coolant ducting and vacuum/pressure seals and tend to minimize reactor downtime, smaller, compact module designs offer the possibility of isolation in event of coolant failure. A cross-section view of the torus is shown in Figure 4. The location of the outer shield and coils is shown relative to the plasma chamber and blanket. The placement of vacuum ducts, pumps and neutral beam injectors will be important in establishing a blanket maintenance strategy. The first wall/blanket consists of approximately 2,000 cylindrical modules mounted on the inside of the inner shield. Located around the top of the torus are 16 vacuum ducts used for chamber evacuation and plasma exhaust removal. These ducts also serve as access ports for the remote handling devices used to replace blanket modules.

Figure 5 shows a drawing of the GA-DEMO blanket module and associated structures. The blanket is composed of four regions. The region nearest to the plasma is a 16 cm thick zone of solid Li_7Pb_2 . The second blanket zone, 14 cm thick, consists of lithium orthosilicate, Li_4SiO_4 (90% ^6Li). These breeding zones are encased in graphite and located in a 70 cm-diameter removable Inconel module. Porous graphite distributors channel helium coolant throughout the breeding regions. The Inconel module is supported by a 4 cm-thick module base plate. The fourth region of the blanket module is the 35 cm-thick inner shield area, composed of stainless steel and boron carbide. Graphite filler placed between the modules makes up a portion of the inner shield. The blanket is protected from interaction with charged particles, neutrons, and photon radiation by a 2 cm-thick silicon carbide first wall liner, which is bolted to the module wall with a ceramic fastener.

FIGURE 4. GA-DEMO Torus Cross-Section⁽¹⁾

FIGURE 5. GA-DEMO Blanket Module⁽¹⁾

The multiplying breeder, Li_7Pb_2 , was chosen because of resource and procurement uncertainties associated with beryllium. The helium coolant flow in the blanket module is designed to allow the coolant to enter at the front of the module. As the coolant flows to the back of the module the helium is heated; the breeding material in the second blanket zone must therefore have a high temperature capability. The breeding compound for the second zone, Li_4SiO_4 , has good thermal properties as well as a high lithium density.

The two breeder zones (30 cm thick) will require removal and replacement since they are exposed to neutron fluxes of 1.4×10^{22} neutrons/cm²/year. The blanket lifetime is expected to be about four years. The maximum blanket exposure is 3.2 MW-yr/m² for a wall loading of ~ 1 MW/m² and an 80% plant factor. The GA-DEMO replacement plan calls for 500 modules or one-quarter of the entire blanket to be removed and replaced with fresh modules every year. The total weight that the remote blanket handling machine would have to carry at any one time, about 460 kg, is such that the modules could be replaced wholly within the containment building. Four of the 16 ports or access areas will be used during an outage for module replacement. It is possible that four remote handling machines could operate simultaneously to reduce downtime. In this arrangement 125 modules will be removed or serviced from each of the four ducts. For a 30-year plant life approximately 6 to 8 breeder replacements will be required.

The GA-DEMO was designed to allow remote maintenance. The basic steps in separating the breeding blanket region from the rest of the blanket so that a new module can be installed in the blanket are:

1. Remove graphite sleeve which fills the space above the module clamping ring by rotation and withdrawal. These sleeves are held in place by a locking device.
2. Cut the vacuum seal welds and remove the vacuum seal ring from the module so that the module clamping ring can be reached.
3. Free helium coolant connections and the module by rotating the clamping ring.

Material inventories in the entire GA-DEMO blanket are shown in Table 2. These inventories are the amounts of material that would be replaced every four years.

TABLE 2. GA-DEMO Material Inventories

Liner	21 tonnes SiC
First Wall	60 tonnes Inconel
Breeding Zone 1	504 tonnes Li_7Pb_2 42 tonnes C 59 tonnes Inconel
Breeding Zone 2	80 tonnes Li_4SiO_4 62 tonnes C 86 tonnes Inconel

UWMAK-II Blanket

A cross-section of the UWMAK-II torus is shown in Figure 6. The primary structural material is stainless steel with a solid lithium compound (LiAlO_2) used for breeding tritium. Helium serves as the primary coolant. The cross-section view of the torus shows the helium coolant supply lines, particle collection plates, graphite curtain, blanket and shield, and toroidal field magnets. The "extended D" toroidal field magnets reach out beyond the shield. Thus the shield can be opened to allow entry to the blanket for removal or servicing. The blanket sections can be removed without taking out the heavy coils. This design feature will be emphasized during the discussion on blanket management strategies.

The UWMAK-II torus is divided into 24 sections, one for each toroidal field magnet. The blanket is composed of three parts: the removable wall cells which contain the breeding material, the vertical structure joining the front wall cells to the helium headers, and the helium gas supply and return headers. The total length of the blanket section from the helium supply headers to the front wall is 89 cm. Figure 7 shows a blanket section. Five cells form a blanket section and there are about 2,840 sections in the reactor.

FIGURE 6. UWMAK-II Torus Cross-Section⁽²⁾

FIGURE 7. UWMak-II Blanket Section⁽²⁾

A cell is composed of four material regions: LiAlO_2 breeder material, Be neutron multiplier, more LiAlO_2 and a graphite reflector. The removable portion of the blanket is composed of semi-cylindrical cells. These cells are 23.1 cm wide, 27 cm high, and 0.6 to 1.5 m long. The first breeder zone is made up of 3 cm of LiAlO_2 (90% ^6Li enriched). This is followed by the 18 cm-thick beryllium metal neutron multiplication region. The outer breeder zone is made up of 10 to 13 cm of LiAlO_2 . Tritium collection manifolds are located near the outside of the outer breeding zone. The breeding material and neutron multiplier are placed in stainless steel pins 3.5 cm in diameter, approximately 30 cm long, with 0.75 mm wall thickness. These pins are closed at the plasma end and connected to a vacuum pumping system at the outside end. The bred tritium is collected and stored for future use.

There are approximately 1.1×10^6 breeder/multiplier tubes in the reactor, 1,430 helium radiator modules, and about 43,000 breeding tube manifolds. The stainless steel-clad graphite reflector is 35 cm high. Helium coolant channels run the length of the cell. The blanket cells are located toroidally so the actual length depends on the blanket location. The connection between the removable front wall cells and the vertical structure serves as an attachment between the two structures and a helium manifold header. The sections have skirts which extend beyond the flat portion of the cell wall. A seal is provided by welding the skirts from each section on the plasma side. The vertical structure that connects the front wall to the headers has a thickness of 1.2 cm and has 1 cm diameter holes spaced 2 cm apart.

The front wall replacement strategy should keep downtime to a minimum. The annual downtime for 12 modules replaced every year is approximately 23 days. In order to service the blanket, the following steps must be performed:

1. Disconnect and remove divertor vacuum pumps.
2. Move back the two outer vertical field coils.
3. Disconnect helium supply and return lines.
4. Swing open outer shield halves to expose blanket.

Cutting torches will be used to eliminate the seal between adjacent modules. This operation is carried out remotely. Special support carriages are used to roll out the blanket piece. The inner and central blanket region can be serviced once the outer blanket sections are rolled out. Once in the service area the graphite curtain is removed and the seal welds and gaskets connecting the sections to each other are burned off. The freed section can now be slid or pulled out.

New front wall sections are loaded with fresh breeding material, LiAlO_2 and Be, assembled on the vertical structure containing the graphite reflector, and welded together. After the graphite curtain is attached the refurbished modules are reinstalled in the reactor blanket and welded/sealed to the adjoining sections. The modules weigh 330 t each, while the largest piece weighs 175 t.

The UWMak-II blanket material inventories are shown in Table 3. This material is to be replaced every two years.

TABLE 3. UWMak-II Material Inventories

Carbon Curtain	1.6 kg C
First Wall Structure	796 tonnes SS
Header Manifolds and Vertical Structure	2213 tonnes SS
Breeder and Moderator Cladding Material	658 tonnes SS
Neutron Multiplier	400 tonnes Be
Breeder Material	436 tonnes LiAlO_2
Moderator/Reflector	1886 tonnes C

ORNL-DEMO Blanket

The blanket module design for the ORNL-DEMO does not attempt to develop specific design criteria but instead defines an envelope or range of typical parameters. Both a cassette and a contour blanket design were investigated for the ORNL-DEMO. The blanket options are described in the following paragraphs.

The major components of the ORNL tokamak are shown in Figure 8. Both blanket options are shown. This figure shows the position of the modules in relation to the plasma chamber as well as the location of the shield and toroidal field coils.

FIGURE 8. ORNL-DEMO Torus Cross-Section⁽³⁾

Both the contour and cassette blanket concepts utilize stainless steel as structural material. Noncirculating liquid lithium is the tritium breeding medium, and helium is the primary blanket coolant in this study.

Contour Design

In the contour blanket design there are 54 segments, or three segments for each toroidal field coil. The placement of the coils and number of segments for each field coil are important considerations in blanket replacement strategies. There are 12 individual modules in a blanket segment, located

poloidally around the segment frame. The segment has a tubular first (inner) wall. In addition to supplying support structure for the modules, the segment framework has coolant inlet and outlet manifolds.

A closeup view of a contour blanket module is shown in Figure 9. There are five regions of absorber/breeding natural lithium contained in the module. The reflector material (two zones) is graphite. Helium coolant tubes run throughout the module. The coolant enters the module at the coolant inlet pipe and is discharged along the walls at the bottom. The module is 75 cm thick and 60 cm wide. The distance along the first wall is 85 cm. There are 648 modules in the blanket.

Blanket lifetime is expected to be on the order of $10\text{-}20 \text{ MW}\cdot\text{yr}/\text{m}^2$, assuming a duty factor of 95%. The expected neutron wall loading is $3 \text{ MW}/\text{m}^2$. For a replacement cycle of six years, one-sixth of the blanket, or 108 modules, should be removed and replaced each year. The blanket material inventories are shown in Table 4.

The problems associated with maintenance and servicing were not addressed in the ORNL conceptual design study. The preliminary design of the blanket is based upon connection of adjacent blanket modules and segments with seam welds in order to avoid any leakage. When the blanket modules are to be serviced or replaced, these welds must be cut. If one-sixth of the blanket is replaced, about nine segments must be handled. The weight of one segment is $\sim 20 \text{ MT}$, so removal of entire blanket segments must be accomplished using an overhead crane. The replacement strategy is to cut the welds between adjacent segments and remove one of the segments through the space between the toroidal field coils. A possible solution to the difficult problem of removing these massive, bulky segments would be to enclose the entire torus within a vacuum building. This would provide a secondary vacuum system and the number of welds could be reduced. Thus the blanket would consist of a set of blanket segments and modules with mechanical connections and gaskets.

FIGURE 9. ORNL-DEMO Contour Blanket Module⁽³⁾

TABLE 4. ORNL-DEMO (Contour Blanket)
Material Inventories

Blanket Module Structure	859 tonnes SS
Breeding Material	114 tonnes Li
Reflector Zone	157 tonnes C

Cassette Design

A cross-section view of a cassette blanket design is shown in Figure 10. The cassette approach to the problem of blanket engineering emphasizes ease of fabrication and maintenance. The location of the neutral beam injectors, cryo-vacuum pumps, and removable shield blocks are shown in the cross-sectional view of the cassette blanket.

The cassette concept makes use of blanket zoning, where the front portion of the blanket is replaced most often. The remaining cassettes, farther from the plasma and thus exposed to less neutron flux, are designed to last longer. Tritium recovery is achieved by allowing the tritium to diffuse out of the cassette through niobium or vanadium recovery windows. A vacuum and pumping system attached to these windows will collect the tritium in getter beds and canisters. The module is 4 cm long, 14 cm thick, and ~1 m wide. It includes supply and return headers for the helium coolant and noncirculating lithium. Expansion plena are provided to accommodate changes in lithium density during operation. The cassettes are long, slender, and rectangular. The reactor is divided into 18 sectors with 15 cassettes in each sector. The cassette module weighs from 3 to 5 tonnes. There are 270 cassettes in the reactor (first zone cassettes) and a total of 810 for all three zones.

Figure 11 is a closeup of a cassette blanket module. The walls are a series of U-shaped tubes containing coolant and enveloping the lithium fluid within. Removal and replacement is accomplished by pulling the cassettes straight out and replacing them with fresh cassettes. The wall loading is 4.5 MW/m^2 . With an 80% plant factor and two year first zone blanket life, the maximum exposure is approximately 7.2 MW-yr/m^2 . The ease of maintenance in the cassette design makes it possible to replace 135 modules with fewer handling difficulties than the contour approach.

FIGURE 10. Cassette Blanket Removal⁽³⁾

FIGURE 11. Cassette Blanket Cross-Section

Blanket Comparison

The total blanket material inventories of the three designs investigated are given in Table 5. This amount of material must be replaced every four years in the GA and six years in the ORNL-DEMO design. The front wall region of UWMAK-II including all of the lithium, beryllium, and aluminum, and some steel, must be replaced every two years.

TABLE 5. Material Inventories - Elemental Composition (tonnes)

	<u>GA-DEMO</u>	<u>UWMAK-II</u>	<u>ORNL-DEMO</u>
Aluminum	--	169	8.6
Beryllium	--	399.6	--
Carbon	113	1882	157
Chromium	47	638	154.6
Iron	41	2200	532
Nickel	116	496	120.3
Lithium	65 ^(a)	46 ^(b)	114
Molybdenum	--	71	17.2
Manganese	--	71	17.2

(a) 20 tonnes ⁶Li
(b) 41 tonnes ⁶Li

TRITIUM

The power reactors chosen for this study, like all fusion power reactors, require the storage, transfer and processing of very large quantities of tritium (on the order of 10^8 Ci). Most of the tritium in fusion reactors is either in the fuel handling and storage system or in the breeding blanket. The task of demonstrating satisfactory control and containment of this highly mobile isotope under the wide variety of conditions encountered in a fusion reactor is difficult but necessary.

The most important blanket characteristic, with respect to tritium inventory, is the choice of a breeding material. The principal alternatives for

tritium breeding materials are liquid lithium metal, flibe, and a variety of solid lithium compounds.

The solid breeders Li_7Pb_2 and Li_4SiO_4 were selected for the GA-DEMO; these appear as roughly spherical particles, housed in hexagonal graphite prisms in two distinct regions. The Li_7Pb_2 section is near the plasma to take advantage of the higher energy neutron spectrum and lower coolant temperature. Blanket characteristics which are important in determining tritium inventories are shown in Table 6.

TABLE 6. GA-DEMO Blanket Recovery System Characteristics

Breeder Summary:	<u>Li_7Pb_2</u>	<u>Li_4SiO_4</u>
Particle diameters	400 μ and 170 μ	210 μ and 90 μ
Packing fraction	75%	75%
Average temperature	500-600°C	600°C
% of breeding rate	30	70
<u>Coolant Parameters:</u>		
T_2 partial pressure	10 ⁻¹⁰ torr	
O_2 partial pressure	10 ⁻² torr	
Max. water vapor partial pressure	10 ⁻¹ torr	
Fraction diverted to T_2O removal	1% or 12 kg/sec	
Temperature of diverted stream	350°C	

Tritium is released to the primary coolant by a process of permeation, first through the pellets themselves, then through the loosely-packed particle bed, and finally through the thin graphite walls of the breeder canister. The only significant resistance to flow is expected to be the breeding material itself. The extent of the resulting holdup can be estimated from the results of Wiswall and Wirsing.⁽⁴⁾ Though not obtaining a diffusion coefficient, these workers did find a holdup time of 13.9 hr for Li_7Pb_2 and

3.3 hr for Li_2SiO_3 (lithium metasilicate)^(a) with sample temperatures of 550°C and 500°C, respectively. At the higher temperatures expected to prevail in the GA-DEMO, holdup times should be even shorter. A tritium holdup (due to diffusion) of 120 g was obtained for the GA-DEMO with 99+% occurring in the Li_4SiO_4 . Solubilities for these materials are almost completely unknown but, based on information for the similar ceramic Li_2AlO_3 , the relative holdup due to this source should be small. It is concluded, then, that the intrinsic inventory associated with solid breeders is likely to be quite low. As noted by others, however, the possibility of massive sintering or crystal formation of the solid breeder particles under intense radiation exists. In the extreme case of complete fusing into one massive lump (per rod), the retained tritium inventory could rise by up to 3 orders of magnitude.

Tritium will leave the breeder particles principally in the form of molecular hydrogen but some release as tritiated water is also to be expected. The bulk of hydrogen species entering the coolant will be bred tritium but small amounts of protium and deuterium as well as tritium will enter from other sources, such as permeation through the first wall and steam generator and hydrogen-producing nuclear reactions.

The breeder material and the beryllium in the UWMAK-II blanket are placed inside stainless steel tubes which are approximately 3.5 cm in diameter with a wall thickness of 0.75 mm and nearly 30 cm long. These tubes are closed at the end facing the plasma and connected at the other end to a small gas plenum attached to a tritium extraction system external to the reactor. Alternatively, the exhaust ends of these tubes could be closed by the use of a porous metal plug. This plug would retain the ceramic powder inside the tube but permit the tritium to diffuse into the helium coolant where it would be extracted. Such a technique would increase the tritium inventory within the tubes, increase the partial pressure of

(a) The orthosilicate form (Li_4SiO_4) used in the GA-DEMO is assumed to have similar holdup characteristics.

tritium in the helium coolant, and subsequently increase the permeation of tritium through the primary heat exchanger. The design philosophy chosen is to minimize the tritium contamination in the helium coolant and extract the tritium directly from the tubes in a separate recovery system.

A diagram outlining the main coolant flows of UWMAK-II and showing the tritium extraction systems is given in Figure 12. The tritium gas is removed from the tubes by a flowing helium stream. For a helium pressure inside the tubes of 1.4 atm, the tritium pressure can be kept to 10^{-4} torr. The calculated tritium release rate to the environment is 0.8 Ci/day.

ACTIVATION PRODUCTS

Fusion power plants operating on a deuterium-tritium fuel cycle will produce copious amounts of 14.1 MeV neutrons. These neutrons will travel out of the plasma and be absorbed in the surrounding systems, producing activation products. Many of the activation products are radioactive and may represent a potential hazard at various steps in the material flow cycle. Radioactive activation product inventories in the nuclear island of tokamak reactors are estimated in the following paragraphs. Activation product inventories were calculated using a one-dimensional neutronics model. Residual elements in stainless steel⁽⁵⁾ were addressed in the GA and ORNL-DEMO designs, while the steel of the UWMAK-II was considered to be pure iron, nickel, and chromium.

The total GA-DEMO blanket activation product inventory during reactor operation is shown in Figure 13. It is seen that the total activity at the time of blanket replacement -- four years after startup -- is 2.5×10^9 Ci. At this activation level, extreme caution must be observed while handling the blanket materials. This activity builds up very rapidly with time. Within four days after initial startup, the blanket activity already reaches 10^9 Ci. In fact, after the very first fusion burn cycle of 800 sec, the activity is 6.2×10^8 Ci. The major nuclides which contribute to the activity are also shown in Figure 13. At early times most of the activity is due to ^{28}Al which is produced by reactions with the silicon liner. This

FIGURE 12. Coolant Flows for UWMAK-II ⁽²⁾

FIGURE 13. GA-DEMO Blanket Activity During Operation

nuclide has only a 2.24 min half-life, so that the activity decays to only five Curies within one hour after shutdown. At times greater than three months, the activity is dominated by the cobalt isotopes: ^{58}Co , ^{57}Co , and to a lesser extent, ^{60}Co , which are caused primarily by (n,p) reactions with nickel. These isotopes are all very hazardous. Manganese-56, which has a 2.6 hour half-life, is also significant, as are ^{55}Fe and ^{51}Cr .

Following shutdown for blanket removal, the blanket remains radioactive for several years. Five years after shutdown the total activity is 1.4×10^8 Ci. Most of the activity at all times is in the first wall, zone 4, and for the first year most of this is due to the cobalt isotopes. The ^{28}Al product in the liner decays very quickly, reaching insignificant levels within one hour after shutdown (see Figure 14).

The UWMAK-II inner blanket is replaced every two years. The activation product inventory during reactor operation is shown in Figure 15. The activity is dominated by ^{55}Fe , ^{58}Co , and ^{56}Mn , while ^{51}Cr , ^{57}Co , and ^{54}Mn are also significant. The outer blanket flux is much softer than that of the inner blanket. As a result, outer blanket activation is controlled by neutron capture processes, particularly the reactions of $^{50}\text{Cr}(n,\gamma)^{51}\text{Cr}$, and $^{54}\text{Fe}(n,\gamma)^{55}\text{Fe}$, although the $^{56}\text{Fe}(n,2n)^{55}\text{Fe}$ plays a significant role in ^{55}Fe production.

Adding the activities of both blanket regions, the total activity two years after initial startup is 4.1×10^9 Ci. This is considerably higher than in either of the other two blankets considered, because it is a bigger reactor, designed to generate 5 GW of thermal power. The ORNL blanket generates 1.7 GWt, and the GA-DEMO generates 1.764 GWt. Normalizing these values, the UWMAK-II activity is 0.82 Ci/W, the GA-DEMO activity is 1.32 Ci/W, and the ORNL activity is 1.04 Ci/W at two years after initial startup. In comparing these values it is significant that impurities were not considered in the UWMAK-II blanket. Not including the quickly decaying ^{28}Al , 8% of the activity in the GA-DEMO blanket is a result of the cobalt and manganese impurities.

FIGURE 14. GA-DEMO Blanket Activity After Shutdown

FIGURE 15. UWMAK-II Inner Blanket Activity During Operation

UWMAK-II blanket activity after shutdown is shown in Figure 16. The inner blanket contributes over 85% of the total activity. Since the inner blanket is replaced five times as frequently as the outer blanket, it is seen that the inner blanket produces 97% of the total activation products throughout the plant lifetime.

The results of neutron activation calculations on the ORNL-DEMO blanket are shown in Figures 17 and 18. At the time of blanket replacement, five years after initial startup, the total activity is 2.1×10^9 Ci. Without the short half-lived ^{28}Al , the early activity buildup is somewhat slower than in the GA-DEMO blanket, with early activation dominated by ^{56}Mn which is produced mostly by the ^{56}Fe (n,p) reaction in the inner zones and ^{55}Mn (n, γ) reaction in the outer zones. Iron-55, produced by ^{56}Fe (n,2n) reactions, contributes a comparable activity after several years. The cobalt isotopes contribute about 0.5×10^9 Ci at replacement.

The activity after shutdown is shown in Figure 18. Five years after removal the total blanket activity is 1.5×10^8 Ci, with the first wall and inner blanket zone each contributing about 0.7×10^8 Ci.

The presence of activation products in the blanket presents a potential hazard to personnel involved in reactor operation and maintenance as a result of decay heat and radiation doses. Afterheat has been calculated for all the designs under consideration, while dose rates have been calculated for the GA and ORNL designs. The results are shown in Table 7. In general it appears that active cooling of the first wall is desirable for several hours after shutdown. Dose rates for all structural materials except the silicon carbide liner of the GA-DEMO are very high at shutdown following power operation and remain high enough to preclude virtually all contact maintenance for years. This means that all blanket material handling must be done remotely, at least for the first few years following removal from the reactor.

FIGURE 16. UWMAK-II Blanket Activity After Shutdown

FIGURE 17. ORNL Blanket Activity During Operation

FIGURE 18. ORNL Blanket Activity After Shutdown

TABLE 7. Blanket Afterheat and Dose Rate

Afterheat (W/cm ³)	Time After Shutdown							
	0	10 min	2 hr	1 day	1 mo	1 yr	10 yr	50 yr
<u>GA-DEMO</u> ^(a)								
SiC Liner	0.7	0.008	2x10 ⁻⁶	0	0	0	0	0
Inner Blanket Wall	0.97	0.87	0.75	---	---	---	---	---
<u>ORNL-DEMO</u>								
First Wall	0.21	0.21	0.18	0.031	---	---	---	---
Shield	0.045	0.044	0.038	0.0067	---	---	---	---
<u>Surface Dose Rate (rem/hr)</u>								
<u>GA-DEMO</u> ^(a)								
SiC Liner	2.3x10 ⁶	3x10 ⁴	7	0	0	0	0	0
Inner Blanket Wall	2.3x10 ⁶	2x10 ⁶	1.8x10 ⁶	---	---	4x10 ⁵	3x10 ⁴	1x10 ²
<u>ORNL-DEMO</u>								
First Wall	5.4x10 ⁶	5.4x10 ⁶	5x10 ⁶	2.1x10 ⁶	1.9x10 ⁶	4.8x10 ⁵	5.7x10 ⁴	---
Shield	3.5x10 ⁵	---	---	---	---	---	---	---

(a) Values interpolated from Ref. 1, following four year operation.

FUEL SYSTEMS

The GA-DEMO fuel system consists of two major subsystems: the Plasma Recycle System (PRS) and the Blanket Recovery System (BRS). The PRS collects, processes and recycles unburned tritium and deuterium from the plasma exhaust. The BRS collects, purifies and stores tritium bred in the blanket.

Tritium bred in the blanket permeates into the primary coolant as molecular hydrogen, as described earlier. This tritium is removed from the coolant by straightforward means in the BRS. Oxidation to water of all hydrogen species entering the coolant is accomplished by maintaining an O₂ partial pressure of 10⁻² torr, sufficient to hold the tritium partial pressure well below the approximately 1 x 10⁻¹⁰ torr necessary to maintain a leakage rate of $\lesssim$ 2 Ci/day. The coolant is dried on chilled molecular sieves.

Recovery is accomplished by heating the sieves and thus removing the water vapor, electrolyzing the product, and removing protium and deuterium by cryogenic distillation. The D-T product is then fed into the Plasma Recycle System.

The GA-DEMO Plasma Recycle System is shown schematically in Figure 19. Gas exiting the plasma chamber first encounters the Flowing Plasma Boundary (FPB) and auxiliary cryosorption pumps. Acetylene and any other volatile impurities are expected to be trapped out on the liquid nitrogen-cooled chevrons of these pumps. Normal regeneration (at 80°K or less) of the cryopumps will not release these impurities, but occasional special regenerations at higher temperatures will remove obstructing deposits. It should be possible to collect the high-Z impurities in concentrated form by preceding a special regeneration with a thorough normal regeneration. The tritium content of these impurities will be recovered in a batch process by burning (oxidizing) the special regeneration effluent, collecting the water and utilizing the normally idle electrolysis unit of the primary coolant purification system to break down the water. The hydrogen isotopes thus recovered are returned to the plasma recycle stream prior to liquefaction.

The more volatile (D, T, H, He) principal components of the recycle stream pass through the liquid nitrogen-cooled chevrons and are condensed or sorbed on the 4.2°K molecular sieve (e.g., Linde 4A or 5A) surfaces of the cryopumps.

These pumps are regenerated on a two hour schedule for both the FPB and auxiliary pumps, by valving them off from the main chamber, heating them up by simply shutting off the liquid helium flow and pumping away the effluent. As short an operating schedule as appears reasonable was selected to minimize tritium inventory.

Oil droplets originating in the pump train, water droplets and any particulates are then removed in a separator/fog filter unit. The gas stream then enters a liquid nitrogen-cooled refrigerator-dryer where less volatile impurities (i.e., water vapor) are condensed and the temperature is decreased.

FIGURE 19. GA-DEMO Plasma Recycle System⁽¹⁾

Passage through a silica gel purifier at 65°K then removes more volatile impurities such as O_2 and N_2 . The gas then enters the liquefier where liquefaction via Joule-Thompson expansion occurs. The product is collected in a liquid D-T storage tank where its isotopic content is monitored. The more volatile helium will presumably have left the liquid by this stage. When the protium concentration equals 1.0%, the cryo-distillation isotope separation unit is activated.

The cryodistillation unit employed for isotope separation consists of two columns and an equilibrator. The feed to the first column is assumed to consist of a high temperature equilibrium distribution of the six molecular forms of hydrogen. With an overall protium (H) atom fraction of 1%, the molecular concentrations become: 24.5% T_2 , 24.5% D_2 , 49.0% DT, 0.99% HT, 0.99% HD, and 0.01% H_2 . To minimize column size and tritium inventory, about half the D_2 is withdrawn with the overhead of the first column. This results in some losses of both deuterium and tritium.

The purified D-T stream from the distillation units is returned to the storage tank from which a continuous liquid flow is extracted for the FPB pellet injectors. A small amount of the liquid is passed through a warmup coil and returned, as gas, to the gaseous D-T storage tank to provide makeup for the once-per-cycle backfill of the plasma chamber.

Crucial to any analysis of potential tritium releases is a knowledge of the amount, distribution, and condition of the tritium inventory. Table 8 summarizes this information for the GA-DEMO. The principal tritium concentrations are seen to reside in the cryopumps, the dessicant beds, and in storage with relatively little being contributed by the distillation units and transfer equipment.

The fuel systems for UWMAK-II are functionally similar to those for the GA-DEMO. The major tritium processing systems extract tritium from a helium coolant and from the divertor which collects plasma exhaust.

TABLE 8. Tritium Inventories in the GA-DEMO⁽¹⁾

System	Component	Inventory (g)	Form
Plasma recycle	Cryopumps	1540	solid
Plasma recycle	Liquefaction	600	liquid
Plasma recycle	Purification	20	liquid
Plasma recycle	Pellet manufacture	1000?	liquid
Blanket recovery	Li ₄ SiO ₄ /Li ₇ Pb ₂	120	dissolved gas
Blanket recovery	Operating dessicant bed	600(max)	water
Blanket recovery	Residual (2 beds)	300	water
Blanket recovery	Traps/collectors/ electrolysis	1400	water ice/water
Storage	(2 cryopump reloadings) or ~6 days burnup	3100	low pressure gas
Total		8700	

The UWMAK-II has separate tritium recovery systems for tritium bred in the blanket and tritium which contaminates the primary helium coolant. The tritium breeding and recovery system was described above.

The divertor tritium recovery system is the major tritium-fuel system component of UWMAK-II. The tritium and deuterium leakage from the plasma is very large, nearly 950 g T/h and 630 g D/h. These very energetic fuel particles will impinge upon a flowing lithium surface that will absorb their kinetic energy. The liquid lithium serves also as a pump for approximately 96% of the (D+T) particles. Based upon the lithium flow rate across the divertor plates, 4.94×10^6 kg/h, the combined (D+T) concentration will increase nearly 1×10^{-6} at. fraction during each pass of the lithium through the divertor. This lithium at 325°C passes through a sodium-cooled heat exchanger which cools the lithium to 200°C.

A cold trap for the removal of the Li(D+T) from the Li is not practical because it would need to operate at 190-195°C which is too close to the exit temperature, 200°C, from the intermediate heat exchanger. An yttrium metal bed is used instead. Based upon a 200°C operation temperature for the yttrium bed, the hydrogen isotope pressure is 2.7×10^{-5} torr and the concentration of these isotopes in the lithium is approximately 1.2×10^{-6} at. fraction. During one transit of the divertor, the lithium receives an additional 1×10^{-6} at. fraction (D+T) so that a combined concentration of 2.2×10^{-6} at. fraction (D+T) reaches the heat exchanger. Precipitation of Li(D+T) in the cold end of the heat exchanger should not occur as the hydrogen isotope concentration is 1/25 of the saturation value at 200°C.

The amount of yttrium required to absorb one day's accumulation of D+T, is 676 kg. At an assumed efficiency of 20%, the total mass of yttrium required is 3500 kg.

The tritium inventory in the various components of the divertor system is shown in Table 9.

TABLE 9. Equilibrium Tritium Inventories for UWMAK-II
(in grams of T)

Location	
Breeder	
Material	120 to 725
Coolant External to Breeding Zone	0.2
Recovery Beds (12 hr cycle)	420
Total in breeder system	540 to 1145
Divertor	
In Lithium	8
Recovery bed (4 hr cycle)	3500
Vacuum system (4 hr cycle)	125
Fuel System	12850
(1 day supply)	
Total in reactor area	17.1 to 17.7 kg ^(a)

(a) This number is very dependent on the reserve amount of T₂ in the fueling system. For example, if as much as one week's reserve must be maintained, the total tritium inventory would be ~95 kg.

HEAT TRANSFER SYSTEMS

For all designs considered, thermal energy which is extracted from the blanket is transferred to a steam generator through one or more heat exchangers. Tritium which permeates into the primary coolant may permeate across the heat exchangers into the steam system. This tritium widely contaminates spent resin beds in the steam system and floor drains in the turbine building when water leaks occur. Less than two Curies of tritium per day are expected to be released to the environment.

AUXILIARY SYSTEMS

In addition to radioactive effluents that result from the operation of the fuel and heat transfer systems, there are radioactive wastes associated with the vacuum system and also the plasma heating systems. In this section the plasma heating and vacuum pumping systems will be characterized. The components that are important to radioactive waste management will be emphasized.

Vacuum System

A schematic drawing of the GA-DEMO primary vacuum system is shown in Figure 20. The vacuum pumping system must perform the following functions:

1. Provide a low pressure in the torus (plasma chamber) to minimize impurity levels.
2. Evacuate spent plasma and particles prior to initiation of the next burn phase.
3. Maintain/extract/pump tritium and its compounds without release to outer environment.
4. Avoid backstreaming or contamination of torus by impurities.

In addition to these requirements, the pumps must be able to sustain large pumping speeds (millions of liters per second) and allow a continuous throughput of D, T and spent plasma particles.

Vacuum system openings to the plasma chamber are shown in Figure 20. Most of the spent plasma is removed at the lower reactor hemisphere opening, which is an annular slot 170 cm wide and 150 cm deep. A total of 16 ducts

FIGURE 20. GA-DEMO Vacuum System

extend from the base of the slot out through the shield and between the B coils. The ducts are flared at the slot but change to a 200 cm diameter cylindrical configuration. A large cryosorption vacuum pump is utilized for each pumping duct. There are 16 cryosorption pumps, and any eight pumps can be operating at once. The operational schedule is to allow the pumps to operate for two hours and then regenerate for two hours.

The auxiliary vacuum pumps are located in the upper reactor hemisphere. There are 16 cylindrical ducts, about 100 cm in diameter, located around the torus. This pumping system removes the remainder of the outlet stream that the primary cryopump (lower hemisphere) did not extract. Each vacuum duct has two cryopumps associated with it. Thus there are 32 total cryopumps, 16 of which are operational at any one time.

The UWMAK-II primary vacuum system has three functions:

1. Evacuate the toroidal plasma chamber to a high vacuum for conditioning and cleaning the vacuum wall.
2. Maintain a pressure of 1×10^{-4} torr.
3. Evacuate the magnet enclosure around the torus.

The pumping systems employed in the UWMAK-II design concept are:

1. Cryopumps for evacuating the toroidal vessel.
2. A liquid lithium trapping surface in the divertor.
3. Mechanical pumps for evacuation of the magnet enclosure building.

The cryopumps used in UWMAK-II consist of a molecular sieve bonded to a substrate and two sets of chevron-shaped metal surfaces. One of the surfaces is cooled to 80°K while the second is cooled to 20°K. The molecular sieve panel is cooled to 5°K.

A schematic of the UWMAK-II vacuum system is shown in Figure 21. The cryopumps are located between the toroidal field magnet and the shield. There are two cryopumps connected to the upper and lower divertor slots or ports, making a total of four cryopumps for each module section. Thus there are 96 cryopumps operating around the torus. Each cryopump has a roots blower and two mechanical pumps for regeneration/evacuation. The cryopump

FIGURE 21. UWMAK-II Vacuum System

system is valved off from the roots blowers and mechanical pumps and is open only during servicing or regeneration.

In order to regenerate the pumps, the trapped gases and exhaust are pumped out and collected in canisters. During the burn cycle, a pump will collect 2×10^4 torr-liters of gas every hour. This quantity of gas is exhausted from the cryopumps in 200 seconds by two mechanical pumps, which can remove 1600 liters/sec. The vacuum exhaust line is about 10 meters running from the cryopumps to the mechanical pumps.

Energetic particle bombardment of the first wall will cause surface atoms to be expelled from the wall. There are several mechanisms for wall erosion, but charged-particle sputtering appears to be the dominant factor. The flux of deuterons and tritons from the plasma striking the wall during full power operation is about $4 \times 10^{13} \text{ cm}^{-2} \text{ s}^{-1}$, with a mean energy of 20 keV. These ions result in a wall erosion rate of about $3.6 \text{ } \mu\text{m}/\text{yr}$. Fusion-produced alpha particle bombardment adds $1.5 \text{ } \mu\text{m}/\text{yr}$ to this erosion rate. When atoms are sputtered and removed from the first wall, many of them will strike the wall again. This self-sputtering of heavy ions adds another $2 \text{ } \mu\text{m}/\text{yr}$ to the erosion rate. The total erosion rate is estimated to be $7 \text{ } \mu\text{m}/\text{yr}$ for a steel first wall.

The first wall in the GA-DEMO design is silicon carbide with an estimated erosion rate of $1.8 \text{ mm}/\text{yr}$.⁽¹⁾ It was shown above that the only radioactive activation product of this material is ^{28}Al with a 134 sec half-life. This small radioactive component is considered negligible. The ORNL design has a first wall area of 474 m^2 , and UWMAK-II has an area of 2850 m^2 . An erosion rate of $7 \text{ } \mu\text{m}/\text{yr}$ results in the loss of 25 kg/yr of first wall material from the ORNL blanket and 150 kg/yr from UWMAK-II. The activity concentration of the ORNL first wall after five years is $39 \text{ Ci}/\text{cm}^3$, and that of UWMAK-II after two years is $42 \text{ Ci}/\text{cm}^3$.

Atoms which are sputtered from the first wall can either be redeposited on the first wall, enter the plasma, or be diverted to the collector plates in the divertor system. Material entering the plasma will be either swept out to the divertor, or deposited on the first wall at the end of the burn

cycle. Sputtered material which is redeposited is indistinguishable from unsputtered first wall material in terms of radioactive inventory.

The maximum amount of radioactive sputtering products present in the plasma is about 0.03 Ci in the ORNL design and 0.2 Ci in UWMAK-II, assuming a particle confinement time in the plasma of 10 seconds. This represents a biological hazard potential in a restricted area of $2.5 \times 10^4 \text{ m}^3$ in ORNL and $8 \times 10^4 \text{ m}^3$ in UWMAK-II; this is an acceptable amount even if it were all released to the reactor containment building.

The radioactive inventory of the divertor collector plates is very much higher. In the ORNL design the activation level of the collector plates is $2.1 \times 10^5 \text{ Ci}$ after five years. The ^{55}Fe level is $1.6 \times 10^5 \text{ Ci}$, the ^{57}Co level is $6.9 \times 10^4 \text{ Ci}$, the ^{54}Mn level is $5.0 \times 10^4 \text{ Ci}$, and other products contribute about $5.0 \times 10^4 \text{ Ci}$. The UWMAK-II divertor activation level is expected to be comparable.

Plasma Heating

Since ohmic heating alone will not meet the heating requirements needed for D-T plasma ignition, auxiliary heating will be required. Although there are several auxiliary heating alternatives, neutral beam injection and radio-frequency wave heating are the chief systems being studied.

A schematic drawing of a neutral beam injection system is shown in Figure 22. The injection system consists of an ion source and power supply system, high voltage electrodes for particle acceleration, magnets to bend ion beams through the desired path, and a neutralizer cell to produce high energy neutral atoms. The neutral beam injection system also requires a vacuum system to limit cold gas leakage and scattering or diffusion of neutrals. The vacuum system also helps to limit tritium contamination of the injector.

Some beam injection designs may employ a bending magnet so that the injector structure would not be in line of sight of the duct and ports exiting around the plasma chamber. Table 10 gives some typical ranges of parameters for a neutral beam injection system for a tokamak device.

FIGURE 22. Neutral Beam Injection System

TABLE 10. Neutral Beam Injection System Parameters

Power Into Torus	380 MW
Beam Energy	200 keV
Total Beam Current	1900 A
Number of Beam Lines	16
Beam Power Into Torus	24 MW
Beam Current	120 A
Duct Dimensions	0.8 m x 0.96 m
Duct Area	0.77 m ²
Cryopanel Area	21.5 m ²
Neutralizer Efficiency @ 200 KeV	20%
Overall System Efficiency	~50%

The beam injection vacuum system utilizes cryosorption panels or pumps to evacuate the vacuum enclosure surrounding the injector. The cryopumps are maintained at 4°K by liquid helium cooling. These cryopanel are placed in the enclosure around the ion source and magnet, duct region around the neutralizer, and the duct region surrounding the beam port area at the plasma chamber edge. The pressure in these areas ranges from 10^{-4} to 10^{-6} torr.

The total cryopanel area is $\sim 21 \text{ m}^2$. The cryopanel must be regenerated after several hours of operation. Mechanical pumps are used for this purpose as well as for initial evacuation of the injector vacuum enclosure. In addition to the mechanical pumps used as backing and roughing, tritium collection canisters are used to collect H_2 , D_2 , T_2 , N_2 , He, and Ar.

The neutral beam is injected into the plasma chamber through a beam port approximately 30 cm by 70 cm. The beam ducting extends between the coils. Some of the shielding surrounding the torus must be removed so that the injector can be operated. This results in the coils being exposed to 14 MeV neutrons streaming around and through the beam port. In this manner the toroidal field coils, injector duct, and cryopumps/panels may be subject to heat and activation. Unneutralized deuterons also present a problem since they may be deflected into the beam dump, react with deuterons already deposited on the dump surface, and produce 2.6 MeV neutrons. The injector structure is activated by 14 MeV neutrons that stream through the injector ports. The structure of the injector is stainless steel while the cryopanel material is aluminum.

Energetic neutral deuterium atoms are injected into the plasma by the neutral beam injector. The charge neutrality is necessary for penetration across the magnetic field lines but creates a shielding problem in the injector. Since neutral particles cannot be deflected around bends, the injector duct must provide a straight path from the point at which the beam is neutralized to the plasma injection location. This provides a direct path for the neutral beam to heat the plasma, but it also provides a path for the fusion neutrons to reach the injector unattenuated. Thus the neutron flux at the injector is a combination of that which has been moderated and attenuated through the blanket, and 14 MeV neutrons which reach the injector unattenuated.

If we assume for simplicity that the injector duct is square, with a width of 50 cm, then the direct streaming flux is 2×10^{12} neutrons/cm²/sec. This value was calculated by considering only those neutrons which enter the duct along a direct "line-of-sight" from the plasma to a plate at the end of the duct. The flux spectrum of direct streaming neutrons is that produced in the plasma, i.e., a Doppler-broadened 14.1 MeV source. An estimate of the injector activation rate can be made by modeling the injector as a square, 50-cm wide duct extending through the blanket with a 5-cm thick stainless steel plate at the outer blanket radius to represent the injector's internal structure. The flux is assumed to be the sum of the direct streaming flux and the attenuated flux at the outside radius of the blanket. This approximation ignores penetration-assisted streaming in which a large part of a neutron's path is within the duct. Neutron attenuation and slowing down within the injector structure are ignored. The neutral beam injector activation density during operation and after shutdown is shown in Figure 23. The total activity after ten years of operation is 4.89×10^4 Ci.

Iron-55 dominates the activity in the injector after one year of operation, and at all times after shutdown. Within one day after shutdown the ⁵⁶Mn and ⁵²V decay, while the ⁵⁸Co, ⁵¹Cr and ⁵⁴Mn decay in less than one year, leaving only ⁵⁵Fe and ⁶⁰Co after one year following shutdown. Iron-55 has a low biological toxicity, leaving ⁶⁰Co as the most hazardous radionuclide after one year.

Radiofrequency heating uses waveguides or antenna structures to generate radiofrequency waves which are absorbed within the plasma. This type of auxiliary heating is desirable because it is efficient in transferring energy into a plasma. Several types of wave heating are being studied:

1. Magnetosonic - MHz range
2. Ion cyclotron
3. Electron cyclotron - GHz range

The RF heating system described here is based upon the design developed for the UWMAK-III study.⁽⁸⁾ Two hundred megawatts are supplied by 12 separate feed systems (17 MW feeds). The feeds are assembled into pairs located toroidally around the reactor. The RF field is held to ~ 3 kV/cm in the RF feeds.

FIGURE 23. Neutral Beam Injector Activation

The pressure outside the plasma confinement region needs to be $\sim 10^{-5}$ torr. The waveguide feeds are protected from the neutral and charged particle flux during the time the RF heating source is shut down by a movable blanket section. The RF heating system is illustrated in Figure 24.

The heating system must deliver 200 MW at 60 MHz for ~ 15 seconds. A copper coaxial line from the RF source to the torus has a diameter of ~ 1 m. The waveguide transition from coaxial could be made near the torus wall. The waveguide must be maintained under vacuum conditions. A molybdenum ridged waveguide is recommended for the last two meters of each RF feed. Thus the feed system is composed of twelve $1\text{ m} \times 1\text{ m}$ ridged waveguides. Each waveguide will have a single cryopump ($\sim 0.64\text{ m}^2$). The cryopump can pump 8.2×10^4 liters/second of D-T and maintain a pressure of 2×10^{-5} torr in the waveguide. During the burn cycle, the cryopumps are valved off and periodically regenerated.

SUMMARY OF SYSTEM DESCRIPTIONS

The major components of a tokamak power plant which require replacement have been described in this section. These components include the blanket, first wall, and shield; heat transfer and fuel systems; vacuum systems; and plasma heating systems. These descriptions will serve as the basis of an investigation of the recycle/recovery options which might be exercised during component replacement. The results of this investigation are summarized in Section III.

FIGURE 24. Radiofrequency Heating System

III. PROCESS OPTIONS

The blanket structures of these tokamak designs were studied to identify potential waste recovery or recycle options for fusion plants. None of these three designs is expected to be typical of a future commercial fusion power plant, but a study of the options for disposal of blanket structure and materials may provide some engineering insight to this radioactive waste problem. Specific options for each design are addressed but the objective is to extrapolate to a generic fusion unit of the future. The insight gained should assist later designs toward a more complete consideration of plant facilities. In particular, consideration of waste disposal or preparation facilities during plant design may ensure minimal interference with plant power generation, yet assure an acceptably low public radiation hazard and cost factor. To do this, the radioactive waste should be in a state or form that minimizes 1) storage, 2) shipping, 3) handling, and 4) environmental interactions of the blanket materials. Recycling of activated materials and separation of radioactive materials from nonradioactive materials are two major approaches to these goals.

POST-SHUTDOWN TRITIUM SCAVENGING

Tritium is the single most troublesome radioactive species associated with tokamak power plants, but several steps can be taken to reduce its presence in the solid breeder designs. When the tokamak is to be shut down, some period of time is required to cool the blanket down from operating temperature. The tritium breeding zones have ceased to produce tritium but still contain substantial tritium in solid solution and in void volumes.

Since the tritium diffusion rate increases with temperature, maintaining a high temperature (near operating) in the blanket would assist in reducing the tritium inventory in the blanket. This is especially true where the diffusion path is short, as it is in the UWMAK-II and GA-DEMO. Delaying the start of temperature cooldown of the blanket by several hours to a day

would reduce the tritium inventory to residual solubility values. This implies a very low release rate from the blanket materials at ambient temperatures once the blanket is removed from the torus.

The afterheat of the blanket would be more than sufficient to maintain the required temperature in the absence of active cooling. It would be necessary initially to actively cool the blanket to prevent damage from the afterheat buildup, but this could be controlled to maintain the prescribed temperature values. It would also be necessary to operate the vacuum system at least periodically to remove the tritium that the blanket outgassed. The coolant tritium recovery system would also have to be operational during cooldown.

BLANKET DISPOSAL

Preparation, shipment and recycle options of materials in the blanket structure are examined next. The discussion is design specific but the options described apply to a variety of reactor concepts. A more specific consideration of the options is presented in Chapter IV.

GA-DEMO Blanket Disposal

Radioactive waste engineering for this design has two interesting features: module uniformity of shape, and ease of module materials separation. The uniform cylindrical shape of the modules is convenient for intact disposal of the entire blanket radioactive waste outflow in an easily manageable, reasonably safe and compact mode that requires a minimum of processing. If no recycling of the blanket materials is desired, then the modules could be prepared for storage at the plant site by "canning" them, four modules to a can (similar to a 55 gallon drum), and then placing four cans together into a re-usable shipping cask. This would result in 16 modules or 7.4 tonnes of activated blanket material in each cask. Each cask would be 0.7 m in internal diameter and about 5.5 m long. Each annual outage would thus require about 32 such casks.

If recycling of all or part of the module is desired, then the module would be opened from the back by removing the module base plate. The breeding material is contained in graphite prisms which can now be removed in layers. The first layer is the most likely to be recycled as it contains Li_4SiO_4 (enriched to 90% ^6Li) and graphite. Recycling would consist of crushing the graphite prisms, removing and discarding much of the graphite (prism shell and powder inclusions), replacing the burned-up Li, and then milling the breeder prior to placing it in a new graphite prism.

If the moderator/breeder (Li_7Pb_2) (not enriched) in the second layer is to be recycled then the recycling steps would be the same for this layer as for the first layer. If this layer is not to be recycled, then it would still be removed to empty the Inconel shell and to separate an essentially nonradioactive component (prisms) from the activated shell. The graphite from recycling the enriched layer and the intact second layer prisms would be treated as low-level activated waste (some tritium present) and shipped to terminal storage.

If the Inconel is not to be recycled, then it would be crushed (or melted), placed in a storage container, shipped to a retrievable storage facility in a reusable shipping container, and stored until the activation has decayed away sufficiently to permit the Inconel to be mined.

Immediate recycling of the Inconel would require a completely remotely operated foundry and machine shop to produce components for use in the nuclear industry.

UWMAK-II Blanket Disposal

The most detailed design considered was that of the UWMAK-II. A summary of the blanket design and module replacement steps as specified by the designers was presented earlier.

Once a module has been removed from the torus, it is transferred to a reactor service building hot cell. There the front wall from all four portions (inner, outer, and two central pieces) of the module is removed in

sections (five cells each) using a plasma cutting torch. The module shell is now removed to another hot cell and refurbished by attaching a new front wall and graphite curtain to it and reinstalling the module in the torus. Two materials processing options are available at this point: to store or recycle (reprocess and reuse) the front wall.

At either the power plant (before shipping) or the storage facility, the blanket materials would be prepared for storage. The expended front wall sections are not further separated into pins or cells but are placed intact after preparation into shipping casks for transfer to a storage facility. A possible method of preparation would be to cast an encapsulating material around one or more sections. By a suitable selection of casting matrix, the sections could be sealed against gas escape and water entry.

Casting in concrete is a possibility, but the section would then be rather difficult to remove. Since the components can be recovered and reused without radiation hazard after approximately a century, such terminal storage does not seem advisable. A glass or polymer matrix may be more suitable, but this would require additional study.

Reusable shipping casks are another option. These casks are rectangular in shape and would be 1.4 m x 1.4 m x 3.6 m dimensionally, thus enclosing a volume of about 7 m³. An average of nine sections would be placed in each cask. This would average out to about 7.2 tonnes per cask and 12 casks would be used for each front wall module.

The original design of UWMAK-II specified that beryllium was to be recovered for recycling; however, no details were provided. To recycle beryllium, it is necessary to remove the lithium collection manifold piping from the pins and then remove the pins from the cells. The pins come in three lengths and shapes and each must be cut into three sections. The two end pieces contain enriched (with ⁶Li) LiAlO₂ and the center piece contains beryllium. The beryllium is removed from the opened pin section, milled, and recanned.

The LiAlO_2 pin sections can now be prepared for and shipped to storage, or they can be recycled. Preparation for storage would consist of stacking these pin sections in a storage container, and sealing and placing this container in a reusable shipping vessel.

Recycling the LiAlO_2 first requires that the pin sections containing LiAlO_2 be separated from the blanket, but the majority of the steps necessary for recovering the LiAlO_2 are partially accomplished already in recovering the beryllium. To recycle the LiAlO_2 it is now only necessary to remove the LiAlO_2 from the pin pieces, mill it and replace the burned-up lithium. It is then ready for recanning. This leaves the empty pin shells and front wall cells, all of stainless steel. Storing these items will first require that the volume be reduced. This can be accomplished by crushing or melting. After this is done the material would be encapsulated and then shipped to storage in a reusable shipping container.

This radioactive stainless steel can be recycled immediately or after some storage time. The storage time would be approximately a century, when the majority of the activation products have decayed away. This delayed recycling would involve no special techniques other than long-term retrievable storage. The earlier recycle option implies a shortage of stainless steel, prohibitive storage costs, and a large inventory of such materials in order for the capital to be available to build the remotely operated mill and foundry that would be necessary to process this activated steel. The steel would be reused in the nuclear industry.

The location of these recycle processes could be at the plant to minimize shipping costs, or at a large central unit to minimize total capital costs.

Every 10.5 years it is necessary to replace the entire module, not just the front wall. This should not involve any change to processing the front wall as it would still be removed from the module; however, this would now be the first step in disassembly of the module rather than refurbishment. After the front wall has been removed and disposed of via established procedures,

the vertical structure and headers are disassembled by removing the canned graphite blocks and then cutting apart the module, primarily with an arc saw or plasma cutting torch. These smaller pieces (mostly piping) would now be reduced in volume, prepared for shipping, and shipped to either storage or a recycling plant. The graphite blocks would be stacked into a storage container and shipped to storage in a reusable shipping cask.

ORNL-DEMO Blanket Disposal

This design is significantly different from the two preceding designs in that it utilizes metallic lithium (a liquid at operating temperatures) for the tritium breeder blanket. The blanket design has two variations: the modular form and the cassette form. A preceding chapter provided brief descriptions of the blanket structures and the prescribed method for blanket renewal.

The torus is formed of 54 sections, which are removed in groups of three for reasons of efficiency. Each section contains 12 modules forming the torus inner wall and attached to the section support structure. If no recycling is to be done, then the modules are removed from the sections (in a reactor service building hot cell), sealed and then set aside to cool and ensure that the lithium has solidified. The module would first be prepared for shipping and storage, then placed in a reusable shipping cask and transferred to a storage facility. If the storage is specified to be retrievable storage then this option would involve entombing the lithium in its irradiated container, the module or cassette, for about a century. After this time the stainless steel and/or lithium could be recovered and recycled. However, during this time it should be stored so as to minimize the possibility of an accident that would rupture the module and allow water access.

A recommended method for retrievable storage would be to encapsulate the module in a tough waterproof polymer, concrete, or glass shell and store this shell in a geologically stable repository. This encapsulation should be done at the power plant, to reduce the hazards of a major accident during shipping. This shell would then constitute the bulk of the shipping preparation and shielding for the module. It is possible to store the lithium and module shell separately but this requires transferring the lithium to another container while it is still molten, and this must be done by remote handling

in either a vacuum or inert atmosphere. The transferring process also involves a greater probability of tritium release, as well as consuming additional materials for casks fabrication.

The modules, once drained of lithium, would be crushed or melted to reduce the volume, encapsulated for shipping and storage, placed in reusable shipping casks and transferred to a storage facility. If recycling is desired, then it is most probable that the lithium would be recycled by draining it out of the sections just removed and transferring it into the modules of the replacement sections. To do this the metal must still be liquid (m.p. 186°C) and the operation must be done in a vacuum or inert atmosphere. The afterheat might be sufficient to maintain the lithium in its liquid state; if not, then heat would have to be supplied.

Continuous reuse of the lithium at the power plant by draining it from expended modules into replacement modules would significantly reduce the volume of out-going radioactive waste as well as reduce the potential biological hazard of each shipment by eliminating the shipping of the tritium which is bound in the lithium. This would leave the empty, compacted and encapsulated stainless steel modules as the only outgoing radioactive waste.

Immediate recycling of the stainless steel would require a special facility, whereas delayed recycling would require retrievable storage of the material for about a century.

The ORNL-DEMO cassette blanket design affords several advantages in the recycling process just described. Chief among them is the ease of preparation of the activated steel waste that leaves the plant. The replacing of only the first breeder zone (FBZ concept) results in similar amounts of activated steel that must be prepared for storage annually. However, these cassettes are of a regular shape that readily permits compact storage of the expanded whole units; the empty shells can easily be crushed and stacked for even more compact storage. Whether the shells are full of lithium and stacked; or empty of lithium and compacted by crushing or melting, they must be sealed against tritium release and water egress before shipping from the plant. A polymeric glass covering may suffice.

The cassette version thus has the distinct advantage of being a much more convenient shape for shipping and handling but all other considerations are generally the same for both the cassette and modular designs.

OPERATIONAL WASTES

Hydrogen is one of the most mobile of all substances, and therefore, the containment of large quantities of tritium is a challenging task. Tritium-contaminated wastes make up the major fraction of routine radioactive wastes from fusion power plants, except, of course, for wastes generated by the routine replacement of neutron-activated blanket components.

Adsorption beds of many types are used extensively for tritium control in fusion power plants. Unfortunately, fusion reactor designs are rarely specific as to the number, size or duty cycle of tritium adsorption beds. The annual accumulation of tritium-contaminated adsorption bed waste from the reference designs considered here are estimated to be :

GA-DEMO:

Molecular Sieves -	12 kg/yr
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UWMAK-II:

Yttrium beds -	3.4 t/yr
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Molecular Sieves -	30 kg/yr
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Storage of tritium on metallic beds is currently favored for safe control of the large quantities of tritium required to fuel a fusion reactor. The duty cycle or operational lifetime of such adsorption beds is currently unknown, and therefore, the quantity of metallic adsorber material in such beds that would require annual replacement is not estimated here.

Tritium is pumped extensively around fusion power plants and, in the process, contaminates vacuum pump fluids. Again, a reliable estimate of the amount and contamination levels of pump fluids is rarely specified in conceptual fusion reactor designs. Moreover, these particular reactors make extensive use of cryopumps, and thus have smaller amounts of contaminated fluids than some of the earlier fusion reactor design concepts. A very rough

estimate of contaminated pump oils from the GA-DEMO or the UWMAC-II is about 200 liters per year, containing less than one Curie of tritium. If extensive repairs are made to one of these plants during a given year and a large amount of degassing is required, the quantity of pump oil waste would probably be greater.

The steam systems of these power plants will require ion exchange equipment for water quality and corrosion control. The tritium that permeates through steam generators will contaminate this equipment, complicating the periodic disposal of spent ion exchange resins. The tritium contamination levels will be quite low, however, permitting these resins to be treated as low-level radioactive wastes. A rough estimate of the annual replacement rate of spent ion exchange resins is about 500 kg, containing less than one Curie of tritium in the form of tritiated water.

Reactor equipment associated with the primary coolant channels will become contaminated with activation products. These radioactive products arise from three sources: impurities such as air leakage in the coolant, neutron sputtering products from the channel walls, and corrosion products. The radioactivity associated with these products is not expected to be important in a helium-cooled blanket. Activation products will deposit either on the channel walls from which they originate or on tritium adsorption beds, which were described above.

Activation products in the first wall will enter the plasma by neutron and ion sputtering. In the UWMAC-II design, which has a graphite shield in front of the vacuum vessel, and the GA-DEMO design, which has a silicon carbide first wall, the radioactivity associated with sputtering products will be very low because silicon and carbon have low activation cross sections. Even for a steel first wall the maximum activity in the plasma will be less than one Curie, most of which will be deposited on the first wall at the end of burn.

After five years of operation the activity which has accumulated on the divertor from first wall sputtering may be as high as 10^6 Ci. The divertor collector may be packaged and removed for permanent storage, or electropolished to remove the radioactive surface impurities and reused.

Neutrons which escape the nuclear island may react with air in the containment building surrounding the island, producing gaseous activation products. This is particularly true for the areas around large blanket penetrations such as neutral beam injector ports. Activated air contains three nuclides of interest: ^{16}N , ^{14}C and ^{41}Ar . With no shielding outside the toroidal field coils, the equilibrium concentrations of these nuclides, assuming a six-hour reactor building air evacuation time, are 45 mCi/m^3 , $1.6 \text{ }\mu\text{Ci/m}^3$ and 0.1 Ci/m^3 , respectively. These numbers are well above maximum permissible concentrations for ^{41}Ar and ^{16}N .

Heavy shielding around streaming ports may greatly reduce these values by reducing the external neutron flux. Air activation may be completely eliminated by evacuating the reactor building, i.e., maintaining vacuum conditions in the building. Once such high activation levels are produced, three options are available for minimizing personnel exposure: restriction of personnel access until the activity has decayed to acceptable levels, venting the activated air to the atmosphere, and collecting the activated species. Nitrogen-16 has a seven second half-life, so it will clearly decay to acceptable levels by the time the building can be entered. Carbon-14 has a 5700 year half-life, so it must be either collected or vented. The maximum permissible concentration of ^{14}C in unrestricted areas⁽⁶⁾ is $0.1 \text{ }\mu\text{Ci/m}^3$. Venting the reactor building air to this level would require about four days. Chemical trapping of the ^{14}C is more appropriate, followed by conversion to a solid such as calcium or barium carbonate. The carbonate would then be transferred to a repository for storage.⁽⁷⁾ Argon-41 has a 1.83 hour half-life. Decay to acceptable levels for entry to the reactor building would take 1.5 days. Venting would take about the same amount of time.

SUMMARY OF PROCESS OPTIONS

Three specific designs have been investigated in this section and the feasibility of various materials recycle/recovery options discussed. The results of this investigation of specific reactor designs are applied in Section IV to a characterization of the process which might be expected in a generic description of the fusion materials resource cycle.

IV. MATERIALS RESOURCE CYCLE CHARACTERIZATION

The three reference designs were studied to identify some radioactive waste design and engineering parameters that facilitated waste handling and material conservation. The following objectives were considered:

- (1) to recycle blanket materials as much as possible to recover resources, reduce radioactive waste storage, and reduce shipping;
- (2) to make radioactive waste effluents dense, of minimal leachability, subject to minimal handling and shipping, and separated from non-radioactive waste as much as possible; and
- (3) to make all steps have a cumulative effect of providing a low biological hazard potential.

In the current state of the art, lithium is the only material available for breeding tritium, which must be used as fuel in the first generation of fusion reactors. It may be a metal (thus a liquid at operating temperatures) or a solid lithium compound (LiAlO_2 , Li_4SiO_4 , etc.). If lithium metal is used, then a high tritium inventory exists in the irradiated lithium. If a solid lithium compound is used as a breeder, then a neutron multiplier and/or enrichment of the lithium may be needed to obtain a favorable tritium breeding ratio without causing the blanket to be too thick.

In all cases, the form of the tritium breeding material is a major engineering consideration in tokamak blanket designs.

LIQUID LITHIUM METAL AS TRITIUM BREEDER

The major aspect of radioactive waste handling for liquid lithium metal breeders is that a high concentration of tritium (several parts per thousand) would be contained in the lithium, mostly as LiH . This is a very stable form, not readily disassociating nor amenable to cold trapping. Therefore the irradiated lithium must be prevented from any accident which could release the tritium until the tritium level has decayed enough to not be a biological

hazard. Recycling the lithium would minimize this hazard, but to easily do this, a means for draining the lithium from the expended blanket units into the replacement blanket units very soon after shutdown of the reactor for replacement of blanket must be provided. This results in: 1) reducing the hazard of tritium release, 2) recycling the lithium, and 3) reducing the outgoing radioactive waste volume.

A proposed method for handling radioactive material for the ORNL-DEMO tokamak design with the contour module option could occur as follows: the sections that are to be replaced during the scheduled outage are removed, using the procedure presented earlier, from the torus as soon as possible after shutdown. The objective is to have the lithium in a molten state. Rapid removal would permit the afterheat to keep the metal molten. The section is trundled into a hot cell room. There the modules are broached while under a vacuum and the lithium is transferred into the replacement modules. This completes the lithium (and the tritium it contains) recovery and recycle.

The now empty modules are stripped from the section structure frame and melted down. The new modules are installed on the support structure and placed into the torus. Thus, the only blanket materials that are not immediately reusable are the module shells. These are melted down rather than crushed because of the tritium inventory that exists in the steel and lithium residue of the module. Crushing is unable to solve the problem presented by the presence of the tritium its escape to the environment must be controlled. If the modules were crushed, then they would also have to be sealed against tritium release as part of the preparation for shipping and storage. Melting the modules in a vacuum furnace is expensive in terms of energy, but is justifiable on the basis of environmental safety and results in far simpler (cheaper) shipping and storage preparations. To elaborate, let us assume that the modules are remotely sectioned (arc saw) to a size that can be fed to a remotely operated vacuum casting furnace. The cooling water of the arc saw will rapidly become contaminated with tritium as it interacts with the residual lithium, but in a closed system this can be controlled. The module is melted, degassed and cast while under a vacuum. This removes the excess tritium that is in solution. The end product advantages of this process are:

- The total volume and surface area of the module shell material are minimized.
- The tritium inventory in the module shell material is significantly reduced. It is in fact reduced to such a low value that coupled with the reduced surface area, the lower tritium release rate means that there are less stringent shipping and storage containment requirements.
- The cast shape would be convenient for handling and storage (by design).
- This process would be done at the power plant site, making the tritium readily available for use as fuel and permitting the casting to be cooled via a process that recovers some of the energy for the power plant.
- By careful selection of additives to the molten steel, a slag could be formed that removes some of the activated species from the casting metal, but this part of the process would heavily depend upon the shell alloy, the species to be removed, and the economic desirability of this particular option, which would be in the nature of a first step in recycling the shell steel.

A convenient shape for the casting is that of a rectangular billet, 20 cm x 40 cm x 200 cm. Assuming a 5% void fraction, then such a billet would weigh 1.22 tonnes. The average empty module shell would weigh 1.33 tonnes, thus a complete blanket section would become 13 billets. A six-year lifetime would result in an average of per year 117 billets and this would be the sum total of the blanket activated waste for the ORNL-DEMO contour module blanket design.

The ORNL-DEMO cassette module blanket design also uses a liquid lithium metal tritium breeder, and the lithium is recycled in the cassette design as it was in the contour design. The same precautions and recycling reasons apply. However, in the cassette design only the first breeder zone (FBZ) is replaced. This results in some savings in handling, especially since removing the FBZ does not require first removing a massive support structure from the torus. However, there is no significant reduction in materials removed from

the plant as annual activated waste. The cassette blanket (FBZ only) may be one-third the size of the contour blanket, but it has an estimated lifetime of two years, one-third that of the contour module. The advantage of simplified handling is thus negated to some extent by the requirement for more frequent changeouts. Blanket process options for the ORNL-DEMO are shown in Figure 25.

A typical material resources cycle for a reactor using liquid lithium as the tritium breeder is characterized by in-plant recovery and recycling of the lithium, primarily to avoid the expense of safeguarding the used but tritium-contaminated lithium. However, there is merit in reusing the lithium solely to save replacing it. The tritium contamination of the blanket materials makes melting the preferred method of preparing the remaining blanket materials for shipment to a retrievable storage facility. These billets of stainless steel can be recycled with little difficulty in about a century when the bulk of the activation has decayed away.

SOLID LITHIUM COMPOUNDS AS TRITIUM BREEDER

Lithium compounds that are solid at operating temperatures present an entirely different set of engineering problems to tokamak blanket designers. The solid compounds are less limiting in terms of corrosive interactions with possible structural materials. Additionally, the lower tritium solubility in the solids, coupled with the requirement that the breeding material be a powder, results in a much lower tritium inventory in the blanket.

The powder is required because of the production of tritium and helium in the breeder. A solid material would present a long diffusion path for the tritium, resulting in a high inventory; but, the gas would collect in voids and cause the material to swell and rupture its container in a short time. A powder provides a short diffusion path to a surface and as the particles swell, they also sinter and fill the powder void spaces without rupturing the powder container. Of course, as sintering progresses the powdered breeder becomes more of a solid. One result is that the particle dimensions increase as the

FIGURE 25. Blanket Materials Flow Chart for ORNL-Demo (contour and cassette)

surfaces merge. This increases the diffusion path length and gradually causes an increasingly higher inventory of tritium in the solution in the breeder compound. This encourages increased swelling which is one major factor determining the breeder lifetimes and recycle options.

Behavior of a solid breeder is typified in the UWMAK-II, a tokamak design using a solid breeder compound (LiAlO_2) and a neutron multiplier (Be). The blanket replacement timetable and techniques were discussed earlier. This discussion assumes the front wall sections are removed from the module and are in a hot cell awaiting further action. The tritium inventory in the blanket has been reduced to residual solubility by afterheat outgassing for tritium scavenging. Thus, the significant activation remaining is that associated with the stainless steel.

The tritium collection manifold is removed from the sections with a remotely operated plasma cutting torch. This is chosen over the more efficient arc saw for reasons of mobility, and also because the plasma torch is part of the original design specified equipment.

With the manifold removed, the breeder/multiplier pins can now be removed and placed in a semi-automatic cutting device that cuts the pin into three sections. Since the pins come in three forms, three such devices would be operating simultaneously. The center pin section is cleaned of beryllium by ramming the sintered metal form out of the stainless steel cylinder. It is presumed that the metal is no longer in the powdered form and that sintering of the beryllium to the steel is minimal. If the first assumption is in error, then the powder is poured from the cylinder instead of rammed. If the second assumption is in error, then chemical leaching or mechanical boring techniques may be required to separate the beryllium from the steel. However, presuming that ramming is sufficient, then the beryllium metal is milled into a powder and reinserted into a UWMAK-II breeder/multiplier pin.

The breeder in the sectioned pins is similarly treated except that the lithium is added to the used breeder compound to restore the original species balance (i.e., LiAlO_2). Then it is milled with precautions taken to confine the small amount of tritium released by diffusion.

The beryllium and LiAlO_2 were originally inserted in the pins as a powder with no physical separation. In the pins, the two powders mix somewhat and then sintering also allows some migration of metals. Thus, when the breeder and multiplier are recovered and recycled, each is present in the other as an impurity. This is acceptable and refining the materials for this reason is not necessary.

The pin shells are melted together to reduce the volume and surface area. This protects against leaching of the breeder, multiplier and steel into the environment when it is placed in retrievable storage. There may be some tritium released from the steel when it is vacuum melted, degassed, and cast into a billet. The steel of the cells would have only a nominal tritium contamination and melting it would seem uncompetitive with crushing this stainless steel, but the savings in shipping, handling and storage costs of this activated waste would indicate otherwise. Thus, the blanket of the UWMAK-II is reduced to a single waste stream; billets of activated stainless steel go to retrievable storage (where they can be later mined).

The vertical structure and headers would be treated in a similar manner after their 10-year lifetime is done. The graphite blocks are left in their stainless steel shells but these are removed from the vertical structure and headers assembly and stacked in a storage container. When this container is full, it is sealed, placed in a shipping container, and shipped to a storage facility (the steel shell is activated--not the graphite). The graphite is removed from the headers structure only because that structure must be cut apart into smaller and more manageable pieces. The existing plasma torch used to remove the front wall sections could be used to section the vertical structure and headers. This activated steel would also be melted into billets because the equipment was available and the handling techniques established.

If the original UWMAK-II design requirement that the beryllium be recycled at the plant is ignored, then a second option becomes viable. This option involves placing the intact pins into reusable shipping containers and shipping them to a central blanket recycling unit. Here the materials operations on the pins would be the same as outlined earlier, but consolidated at one plant. This would represent a savings in capital by centralizing these complex operations.

Thus the input to the recycling unit would primarily be the tritium breeder pins and the output would be new UWMAK-II tritium breeder pins and stainless steel billets (the old pin cladding). The pins would not be radioactive and could be shipped by conventional methods to the power plant for installation into the front wall. The billets would be activated and would need to be shipped to a retrievable storage facility in reusable shipping containers.

The empty cells, vertical structure and headers would need to be prepared for shipping to retrievable storage and melting the stainless steel into billets would present the simplest solution. The graphite blocks would be removed from the vertical structure, stacked in a storage container, and shipped to a storage facility. Shown in Figure 26 are the various blanket processing options that are possible in the UWMAK-II design.

Let us now examine a different design--one also using a solid breeder compound but characterized by small modules (in comparison to UWMAK-II). The GA-DEMO has been briefly described earlier, so this discussion will start with the module removed and in a hot cell.

The optimum processing stream among those previously presented would appear to be to ship the modules out to a recycling center. The modules, in groups of four, are canned for shipping in containers similar in shape and size to 55-gallon drums. Four such drums are placed in a reusable shipping cask. The shape (cylindrical) of the modules greatly facilitates this course of action. It allows a fairly high density of radioactive waste material to be obtained with a minimum of processing. Since the tritium breeding materials are still contained in unbreached Inconel shells, the rate of tritium diffusion from the canned modules is reduced. Of course, using the afterheat of the blanket on shutdown to degas the blanket by diffusion has already reduced the tritium inventory in the module to a low value at near equilibrium with residual solubility in the components. On cooling, the remaining tritium is trapped in the module due to a low diffusion rate.

At the processing center, the modules would be removed from their containers and cut open using an arc saw. The module materials would then be separated into three major streams: 1) the Inconel components; 2) the Li_4SiO_4 (90% ^6Li enrichment initially) in graphite prisms; and 3) the Li_7Pb_2 in graphite

FIGURE 26. Blanket Materials Flow Chart for UWNK-II

prisms. The Inconel is compacted by crushing (or melting), placed in a storage container (the primary activation shield), and shipped to a retrievable storage facility in a reusable shipping container. After about a century, the Inconel is used as a source of raw materials.

The Li_4SiO_4 prisms are crushed and the majority of the graphite prism shards and powder inclusions are removed. Perhaps this could be done by floating the graphite in a water rinse. The nominally Li_4SiO_4 compound, which is no longer stoichiometric, is restored to isotopic species balance by adding ^6Li to replace the "burned-up" metal. The material is then milled into a powder, diluted with graphite powder and canned in a graphite prism. It is now ready to be installed in a new module and reused in the reactor.

The Li_7Pb_2 prisms are treated like the Li_4SiO_4 prisms, except that the final milled product is not diluted with graphite before being recanned in a graphite prism. Blanket processing options for the GA-DEMO are illustrated in Figure 27.

OPERATIONAL WASTES

Radioactive wastes handled during routine operation and maintenance are likely to be a significant source of exposure to plant personnel. Processing of normal operational wastes may result in some environmental release of radioactive material. A reference process stream is described here for waste materials which will be handled on a routine basis. Where data are unavailable, radioactive material inventories and operational lifetimes are based on previous PNL estimates.⁽⁹⁻¹⁰⁾

The divertor collector is assumed to be a flowing lithium type. Spent fuel and plasma impurities impinge on the flowing lithium and are transported away from the divertor. First wall sputtering products are trapped on zirconium foil getters. Assuming that these getters are replaced annually, the activity level at replacement may be about 10^6 Ci due to first wall activation products. If a graphite or silicon first wall liner is used, this activity should be reduced by several orders of magnitude. The total rate of replacement of zirconium foils is about 100 t/yr. This will be sealed in drums and disposed of as radioactive waste.

FIGURE 27. Blanket Materials Flow Chart for GA-Demo

Tritium in the divertor and in the primary coolant streams will be trapped on yttrium adsorption beds. Hydrogen isotopes are adsorbed on the yttrium beds which are periodically removed and heated for fuel recovery. Assuming a useful lifetime of one year, the consumption rate of yttrium is 3.4 t/yr. The activity is estimated to be about 10^3 Ci of activation products and 2×10^6 Ci tritium at time of replacement. The cost of yttrium should make its recycle a desirable process. In that case the tritium will be removed by thermal degassing and the yttrium will be reprocessed and returned to the reactor.

Tritium enters the steam system by permeation through the heat exchangers. Some of this tritium will be released to the environment by permeation across the hot surfaces of the steam generator walls. The rate of this environmental release has been estimated by various researchers between 0.5 Ci/day and 30 Ci/day, depending on reactor size, and assumptions made regarding the permeation versus pressure relationship and the reduction achieved by tritium barriers such as oxide or metal films.

Ion exchange resins in the steam system will be replaced at the rate of about 500 kg/yr. These resins will contain no activation products and less than one Curie of tritium, so they are not considered here as significant radioactive wastes.

Release of activated air will be dealt with by several means. Access to the reactor building will not be permitted for at least 24 hours after shut-down. By this time ^{13}N and ^{16}N will completely decay. Argon-41 has a two hour half-life and will be vented to the atmosphere at rates below maximum permissible limits ($0.04 \mu\text{Ci}/\text{m}^3$). Carbon-14 may be collected by means which have only been suggested.⁽⁷⁾ After collection it is assumed here that it will be disposed of as calcium carbonate.

The wastes associated with and arising from blanket treatment seem to be primarily two forms: tritium and fine particulates. Both forms are associated with three processes used in blanket treatment: milling of solid compounds used for tritium breeding or neutron multiplication; cutting of blanket components with a plasma torch; and sectioning of blanket components with an arc saw. This last process is water-cooled (surface--not internal) and much but not all of the waste is suspended or dissolved in the cooling water.

Control of these wastes primarily results from treating blankets in a room or building that is designed for tritium containment and fine particulate filtering. Such building would be nearly standard equipment for blanket processing at power plants. The arc saw has the same requirements, plus the additional need that the recirculating coolant be controlled.

V. ENVIRONMENTAL CONTROL TECHNOLOGY CONSIDERATIONS

In conducting this study, several features arising from either a specific reactor design or from possible recycling processes were noted to have environmental control technology considerations meriting additional discussion.

TRITIUM CONTROL

Tritium will probably be the most potentially hazardous species associated with a first-generation fusion power plant. This is due in large part to its high diffusion coefficient in most materials, especially the solids. This makes it difficult to contain, especially at elevated temperatures. However, several steps, described below, can be taken to improve tritium control.

Offgassing After Shutdown

The breeder material for the first generation of fusion power plants will undoubtedly contain lithium in some form. If the lithium is contained in the blanket as a pure metal, then it will exist as a liquid at reactor blanket operating temperatures. The tritium produced will largely stay in solution. The high solubility of tritium in lithium does not readily assist in removing the tritium. However, this situation does not exist for the compounds of lithium that remain a solid at reactor blanket temperatures. Swelling as well as diffusion considerations require that the compound be powdered. Lithium powder allows rapid diffusion of the tritium. This is particularly important when the reactor is to be shut down. Once the fusion processes are terminated, tritium production ceases. But if the afterheat of the blanket is permitted to maintain the blanket at operating temperatures for several hours to a day after shutdown by suitable throttling of the steam generator valves, then a significant reduction in tritium inventory in the blanket breeder materials results. Reduction in inventory will greatly assist in tritium control during the blanket renewal operations, as the tritium has been reduced to residual solubility levels. When this is matched with the lowered temperature of the blanket units when the actual replacement operations are underway, then a very low release rate by diffusion is the net result.

Vacuum Containment

Another consideration in limiting effluents resulting from fusion reactor operation is the control of radioactive releases to the reactor building. Small amounts of tritium will leak out from the blanket structures, coolant lines and recovery systems. Some means of controlling this tritium is needed. It is possible that a large vacuum containment structure encompassing the fusion reactor could prevent much of this tritium from escaping while at the same time providing a favorable environment for reactor operation. The containment building will have five major functions:

1. Maintain vacuum environment around fusion reactor,
2. Provide a protective barrier against leakage of radioactive material to the environment in accident and routine operational modes,
3. Protect against leakage from plasma recycle and primary coolant purification systems,
4. Act as a biological shield to limit radiation dose onsite, and
5. Provide a barrier against external missiles or projectiles.

The following discussion will describe the concept of vacuum containment and point out the advantages and disadvantages of using such a system.

The entire fusion reactor is to be enclosed in a sealed building and subjected to a vacuum. The vacuum building lessens rigid vacuum-tight requirements for seals, welds, etc. Attachments between adjacent modules and blanket segments can be made by using mechanical connections if a vacuum building is utilized. No vacuum-tight welds will be required which means that the amount of remote welding is reduced, and thus the time required to disassemble the modules is decreased. The downtime can therefore be reduced for module replacement outages. This would be particularly true for the UWMAK-II and ORNL-DEMO reactor concepts since extensive weld cutting is required to dismantle blanket modules. In the GA-DEMO reactor concept, blanket module removal is accomplished inside the primary vacuum region. Remote disassembly of other components such as diagnostic equipment or neutral beam injectors is also made easier because they do not have to traverse a vacuum-tight barrier. Another

result of using a vacuum containment building is reduction in the overall dimensions of the reactor. If the normal protective barriers are utilized, sufficient space for access to vacuum welds must be provided. The use of the remote welding/sealing machine requires that the blanket segment and module welds be accessible.

The most important advantage from an effluent control perspective is that routine and accidental releases of tritium can be controlled so as to minimize its effect on the environment. The tritium would be passed through a recovery system composed of sieve collection beds.

The use of a vacuum building also has some disadvantages and technical problems which must be solved. Enclosing the entire fusion reactor in a vacuum-tight sealed building will be a major construction and engineering problem. Another problem is that the time needed for pump-down might be excessive. Also, the number of vacuum pumps and blowers and the power required to operate them could prove to be troublesome.

A schematic drawing of one possible configuration of vacuum containment and a fusion reactor is shown in Figure 28. The location of the reactor and the vacuum inner containment are shown in the figure.

LIQUID VERSUS SOLID BREEDER

Selection of the form that the tritium breeder takes is primarily a decision whether to use a pure lithium metal, which is liquid at operating temperatures, or to use a lithium compound, which is a powdered solid when in use.

The liquid has the disadvantage of being corrosive to many potential structural materials as well as having a high solubility for tritium. On the other hand, neither enrichment nor neutron multiplication is necessary to obtain a favorable tritium breeding ratio, and changes in the breeder structure are not limiting factors in blanket lifetime. From the recycling aspect, the liquid metal breeder is an elegant solution in that it is necessary only to transfer the liquid from the old to the new blanket units. Of course, this presumes that an inert atmosphere (a vacuum is very inert) is provided for

FIGURE 28. Vacuum Containment Building⁽³⁾

the whole process, and that the operations are designed to handle a thermally and radioactively hot fluid. Lithium's only significant drawback from the environmental control aspect is minimizing the hazards of a lithium fire or spill that releases its dissolved tritium. A lithium interaction with water would be particularly troublesome.

The solid breeder compounds avoid the high tritium inventory problem by their powdered form. Solids tend to swell from gas production when irradiated. This swelling limits lifetime for blanket units using powdered solid breeder compounds as the powders sinter and become solid. The other major drawback to using compounds is that lithium enrichment with ^6Li or use of a neutron multiplier (and frequently both) may be required to obtain a favorable tritium breeding ratio without an excessively thick blanket. From the recycling viewpoint, solids are difficult in that the recovery steps are frequently awkward due to remote handling requirements, but if recycling is a concept designed into the blanket units, then the difficulty can be limited to reasonable levels.

The environmental control technology considerations for the solid breeder compounds are good, as the low tritium inventory is amenable to outgassing a part of the reactor shutdown. There are suitable compounds that have a low resultant activation, thus presenting no additional radiation hazards. The actual reprocessing operations do pose some hazard in the milling steps. Some tritium may be released from enclosed voids and some hazardous fine particulates are formed, but both of these can be controlled with existing technology.

STRUCTURES

In considering the recycling options for the blanket components of the three reference designs selected for study, several specific points relating to structures were noted as having potentially important environmental control technology considerations.

Ease of Disassembly

The designs selected for study can be regrouped into another classification besides the form (liquid or solid) of the breeder. There is size similarity of the blanket units and their relative ease of disassembly. The UWMAK-II

and the ORNL-DEMO (contour module) are grouped together because in both the entire blanket structure is removed from the torus in order to renew the expended blanket materials. The structures removed from the blanket are massive and a sizeable portion is returned to the torus unchanged. The GA-DEMO and the ORNL-DEMO (cassette version) have small blanket units that are removed from the torus and replaced totally. The original design descriptions of the two groups show that the latter group is more easily disassembled and this advantage could be expanded into a design cost savings by reducing downtime for blanket replacement.

This concept of easily disassembling a structure also shows up in the recycling of the solid breeder compounds. In the UWMAK-II, the pins must be sawed apart into sections that contain most (but not all) of the breeder or multiplier. Then the material must be forcibly removed from this container. This is very unlike the GA-DEMO where the breeder and neutron multiplier are in separate containers and these containers are very amenable to separation from their contents.

This ease of disassembly is a concept that has cost savings as an implied benefit.

Melting versus Crushing of Stainless Steel Components

From the radioactive waste viewpoint, melting of "consumed" stainless steel reactor components is far superior to crushing them. Melting (which implies casting) results in a lower volume of radioactive waste (higher density) much less susceptible to leaching (minimal surface area) and the reshaping makes shipping and long-term storage easier by providing a uniform shape that is a single piece of material. Additionally, there is the possibility that for fusion components, the melting would release the excess dissolved tritium and storage of the casting would no longer need to be airtight.

The singular disadvantage to melting is that it requires much energy--about a megawatt-day/tonne of stainless steel. This alone makes crushing competitive in the preparation of irradiated stainless steel structures. The capital costs are comparable and the operation complexity of both systems is

similar. It is possible that if the melting was done at a power plant (a daily operating basis is presumed), then the heat from the castings could be recovered, thus reducing operating costs.

VI. RECOMMENDATIONS FOR FURTHER RESEARCH

The investigation reported here describes the materials requirements for a specific class of fusion reactor designs and process options for the back end of the materials resource cycle. This effort is part of a continuing program to identify environmental control technology requirements associated with all aspects of the deuterium-tritium fusion materials resource cycle. Current plans for this program include expanding the scope to include other reactor types such as inertial confinement and steady-state magnetic confinement schemes as well as a broader range of blanket options such as alternative coolants and structural materials. The complete materials resource cycle illustrated in Figure 2 will be addressed, from resource extraction through fabrication and plant operation to ultimate waste disposal or recycle. This materials resource cycle characterization will then form the basis for an assessment of effluent releases and the adequacy of available environmental control technology to deal with these releases.

Several specific areas have been identified which may greatly reduce the environmental release rates from the fusion materials resource cycle but which are outside the scope of this investigation. The greatest single factor in determining the amount of material which can be recycled is the degree to which ease of disassembly has been factored into blanket module design. As fusion reactor designs evolve a major design consideration should be the disassembly of blanket modules to enhance their recycle potential and therefore reduce materials resource requirements and radioactive waste inventories.

Personnel access to the reactor building or other buildings which house irradiated reactor components will be severely restricted if not entirely excluded. This means that most of the materials handling and maintenance procedures must be performed remotely. Remote handling may also reduce effluent release by limiting the number of times that buildings containing volatile radioactive materials must be accessed. Development of remote maintenance technology for fusion reactors will be a key factor in controlling personnel exposure and effluent releases.

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