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ENVIRONMENTAL ASSESSMENT
FOR
FUSION REACTOR
UTILIZING
BROOKHAVEN MINIMUM ACTIVITY BLANKET

AEC Research and Development Report

Atomics International Division
Rockwell International

8900 De Soto Ave.
Canoga Park, California 91304

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By
R. V. ANDERSON

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ABSTRACT

The University of Wisconsin designed Tokamak Fusion Reactor, UWMAK-I, has been modified to accept the Brookhaven designed minimum activity blanket. This report includes the design integration analysis, accident analysis, and environmental assessment for the resulting design. The minimum activity blanket is helium cooled, with tritium breeding in a solid lithium alloy. This system provides several advantages over the original lithium cooled blanket, including decreased tritium inventories and decreased induced radionuclide activity.

I. INTRODUCTION AND SUMMARY

The objective of this report is to assess the environmental impact resulting from the operation of a Tokamak Fusion Reactor which utilizes the Brookhaven-designed minimum activity blanket (MAB).⁽¹⁾ Because the reactor design is a modification of a Tokamak design developed at the University of Wisconsin (UWMAK-I),⁽²⁾ a similar evaluation of that reactor was used as a point of reference for the blanket integration design study. Thus, in achieving its required objective, this report includes the design integration analysis, accident analysis, and environmental assessment analysis for the resulting design.

One environmental impact resulting from the operation of the University of Wisconsin Tokamak Reactor, UWMAK-I, is the generation of a large inventory of long-lived radionuclides in the blanket and structural materials. UWMAK-I used stainless steel as a structural material, with liquid lithium acting as an coolant - tritium breeding medium. The minimum activity blanket design is an alternative to the UWMAK-I design. The blanket and structural materials in UWMAK-I are replaced by a modular blanket-shield unit. This modular blanket was developed with an integral moderator, breeder, and shield. Ultra-purity sintered aluminum product (SAP) was used as the basic structural material, with gaseous helium as the coolant. Tritium breeding occurs in a solid lithium-aluminum alloy (LiAl) which is part of the module structure. By incorporating this modular design into UWMAK-I, the long-lived radionuclide inventory is reduced by many orders of magnitude. The blanket and coolant tritium inventories are also reduced by approximately one to two orders of magnitude. The initial activity of the blanket, following 3 years operation, is similar to that of the UWMAK-I, but consists primarily of short-lived Na^{24} , with a half-life of 15 hr. The only significant long-lived activity in the blanket results from 800 Ci of Al^{26} (half-life of 730,000 years). The total blanket radionuclide activity after 3 years in operation is $\sim 1 \times 10^9$ Ci. This drops to <5000 Ci after 2 weeks, and <900 Ci after 10 years. The low volumetric activity associated with the spent blanket material should permit disposal by direct burial after ~ 1 year cooldown period.

As a first approximation, the environmental hazards resulting from MAB - controlled thermonuclear reactor (CTR) plant accidents are reduced in proportion with the total radioactive inventory. An arbitrary accidental ground-level release of 1% of the tritium in the plant coolant results in a site boundary dose of <0.1 rem. This is a three order of magnitude reduction below a similar event for UWMAK-I. Further hazards analysis for a newly postulated event - vaporization of two blanket modules - demonstrated that the only significant site boundary dose (>1 rem) resulted from sodium-24 activity (half-life of 15 hr). No credit was taken for holdup in the CTR containment structure, although order of magnitude reductions could be realized from this, due to the short Na²⁴ half-life.

The alternative design of the first wall provides limited direct access to the blanket modules. This feature, coupled with the reduced radioactive inventory, allows for maintenance of certain portions of the first wall-blanket without prior removal of a 30° magnet sector. Future designs incorporating minimum activity blanket modules could include resizing of the magnet for maximization of maintainability.

II. BLANKET DESIGN AND INTEGRATION

A. MODULE DESCRIPTION

The blanket design employed here is a modification of the Brookhaven Minimum Activity Blanket, Type 2, Design 4, as described in Reference 1. Modification of the BNL blanket was required for integration into the University of Wisconsin reference reactor design, UWMAK-I. The design is of modular construction, encased in a cylindrical shell of SAP. Elevation and cross-section views of the module appear as Figures 1 and 2, respectively. The modules are integrated into a blanket structure as a close-packed hexagonal array. The circular module head thus becomes the first wall of the fusion blanket. Figure 3 illustrates the appearance of such a wall, as viewed from a plasma surface.

An aluminum-base material was chosen for the blanket structure to minimize the induced radioisotope inventory. Aluminum provides minimal induced activities, because almost all radioisotopes are generated from impurity activation. Exceptions to this are only short-lived sodium-24 ($t_{1/2} = 15$ hr) and long-lived aluminum-26 ($t_{1/2} = 730,000$ years). These are generated by (n, α) and $(n, 2n)$ reactions in aluminum-27, respectively. State-of-the-art aluminum processing techniques allow reductions in impurity concentrations to the part-per-million level, thus minimizing the trace-element-induced activity.

The modules are cooled by helium at ~ 10 atm pressure. In order to minimize the temperature of the SAP structural material, the module has been radially divided into two temperature regions: (1) the "cold" region, including the SAP shell, and (2) the "hot" region, including the neutron-multiplying and neutron-moderating materials. Helium coolant entering the module is first circulated through the annular space connected to the SAP shell. It then passes into the "hot" region through radial ports, and exits through the multiplying and moderating materials. The regions are separated by annular insulating shells of aluminum oxide (alumina) and lithium-aluminum alloy. The alumina insulator provides a low thermal leakage path between the regions. Approximately one-third of the thermal power developed in the module is deposited in the "cold" region. This design thus allows for helium exit temperatures greatly above the maximum temperature set by thermal stress limitation on the SAP shell.

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Figure 1. Blanket Module Design — Elevation View

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Figure 2. Blanket Module Design – Cross-Sectional View

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TOTAL FIRST WALL AREA	2830 m ²
CIRCULAR AREA PER MODULE	0.159 m ²
EFFECTIVE AREA PER MODULE	0.175 m ²
NUMBER OF MODULES	16137
AVERAGE POWER PER MODULE	0.311 MWt

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Figure 3. First Wall as Viewed from Plasma Surface

For discussion purposes, the blanket is divided axially into four zones. The first zone is a SAP first wall. The second zone is the fast-neutron zone or neutron-multiplying zone. The third zone is the neutron-moderating zone. The fourth and final zone is the shielding zone. These zones are illustrated in the elevation view of Figure 1. A detailed description of each of the zones follows.

The first wall, and structural shell, are SAP 895, a sintered aluminum product composed of 89.5 wt % aluminum with a 10.5 wt % aluminum oxide dispersoid. SAP exhibits the high thermal conductivity and low induced radioactivity of aluminum with increased strength at elevated operating temperatures. The maximum allowed temperature in this design was limited to 400°C, where SAP exhibits a failure stress of ~10,000 psi. The Al_2O_3 dispersoid provides increased strength in the absence of alloying impurities. The SAP could then be made with ultra-purity aluminum. Discussions with Alcoa indicate that this ultra-purity aluminum (99.99 wt % Al), with the required low impurity concentrations, could be made in large quantities for \$1.00 to \$1.50/lb (in 1974 dollars). The composition in such aluminum is shown in Table 1, and is similar to Alcoa 1199 selected grade with reduction in the iron, zinc, and manganese concentration.

The second zone, or neutron-multiplying zone, is composed of high-purity beryllium. As illustrated in the elevation and cross-section views of the blanket module, Figures 1 and 2 respectively, the beryllium is in the form of small spheres which fill the cylindrical volume within the aluminum oxide insulating shell. Beryllium is used to increase the number of neutrons available in the blanket for tritium breeding, and thus increase the modular blanket tritium breeding ratio. Tritium breeding occurs in the solid LiAl alloy which is sandwiched between the Al_2O_3 insulating shells. This alloy is isotopically enriched to 90 wt % Li^6 . The $(n, 2n)$ reactions in beryllium provide lower-energy neutrons, which are utilized in the $\text{Li}^6 (n, \alpha) \text{H}^3$ reaction in the breeding material.

Because of its high melting point, the beryllium can withstand significantly elevated temperatures of the helium coolant. The beryllium is not a structural or containment material, and is fabricated in the form of small spheres to minimize manufacturing costs and maximize heat transfer capability. The

TABLE 1
IMPURITY CONCENTRATIONS FOR
ULTRA-PURITY ALUMINUM(3)

Impurity	Concentration (wppm)
Si	20
Fe	1
Mn	1
Mg, Pb, Cu	1
Cr	0
Zn	1
Ti	2
V	2
Sn, Bi	1
B	2
Ga	8
Zr	4
Cd, Sb, Be, Ni	0
Co, Mo	0
Na, Ca	<1

utilization of beryllium spheres results in a pseudo-homogeneous helium flow path through Zone 2. It is assumed that local overheating resulting from flow path blockages will cause only minor perturbations in the overall heat transfer scheme.

The third zone, or neutron-moderating zone, is composed of graphite. As illustrated in Figures 1 and 2, this graphite is in the form of cylindrical rods, stacked within the Al_2O_3 shell. The graphite is utilized to slow down neutrons passing through the fast zone or generated in the fast zone. These neutrons are parasitically captured in additional LiAl alloy, located in the annular ring surrounding the graphite structure. The high melting point of graphite allows elevated helium coolant exit temperatures. The graphite is not a structural material; however, the coolant passages between rods must maintain low impedance to helium flow, in order to prevent local melting or overheating.

Zone 4, the shielding zone, consists of a shielding plug which mates into the remainder of the shielding structure (see Figure 2). This plug is of a stepped design, to minimize neutron streaming. It is mounted outboard to the module. The design of the shielding plug and the shielding structure is a modification of the original UWMAK-I shield design, and is schematically illustrated in Figure 4. Shielding is provided by three repeated layers of boron carbide followed by lead. A final vacuum plus insulation zone is provided to minimize thermal throughput to the magnet. Helium cooling of the shield plug is provided through the SAP inlet piping to the module, which spirals through the shield plug and into the module head. This SAP tube is in intimate thermal contact with the shielding material, and removes the 25 MWt generated in the shield plugs, thereby providing thermal preheat for the helium coolant entering the module. Removal of the remaining 25 MWt generated in the shield structure is provided by helium circulation through four layers of SAP tubes within the shield (see Figure 4).

The helium outlet tubing from the module also spirals through the shield plug. The outlet tubing is formed from titanium, and is inertially welded to the head of the module. Aluminum oxide insulation is provided around the outlet tubing to minimize heat losses to the shield plug, and to minimize the heat leakage path back to the helium inlet. The helium inlet and outlet are available for connection to the main coolant manifold as they exit the shield plug at the mating flange. Further description of integration of the module into the first wall structure will be provided in Section II-B.

B. MECHANICAL MODULE INTEGRATION

The UWMAK-I toroid⁽²⁾ was formed from twelve 30° sectors, each sector being individually removable for maintenance and/or replacement. The modified Brookhaven minimum activity blanket was incorporated into the UWMAK-I design, subject to the constraint that magnet size be maintained. Thus, the modular blanket and shield, the shield structure, the divertor, and the helium manifold were all designed to fit within the magnet envelope. A cross-sectional view of the upper half of the redesigned toroid sector appears as Figure 5.

The modular blanket design requires that the shield-structure exist directly behind the first wall and blanket, because the shield plug is mounted integrally

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Figure 4. Shield Cross-Sectional View

Figure 5. Torus Cross Section - Upper Half

SECTION A-A

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Figure 6. Helium Manifold Installation

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on the module. In certain instances, this required reduction in the length of individual modules. These sublength modules (0.80 m long, rather than 1.2 m long) are located on the inner and outer peripheries of the toroid. They are similar in design to those appearing in Figure 2. However, the length of the second zone (beryllium neutron-multiplying zone) was reduced to 0.4 m, and the length of the third zone (graphite neutron-moderating zone) was reduced to 0.4 m. These sub-length module zones comprise $\sim 35\%$ of the total first wall area. The presence of these sub-length modules only slightly degrades the thermal performance of the system, and also reduces the breeding ratio for tritium. Note that the shield thickness of 1 m has been maintained in all shielded areas. This provides minimum thermal deposition in the magnets, and guarantees minimal induced activity in the magnet and helium manifold components.

Assembly of a toroid sector would proceed in the following manner. The shielding-structure would be assembled on a 30° sector basis. The cross section of this structure would be that of the shield. Hexagonally arrayed, stepped holes would be provided in the structure, for insertion of individual blanket modules with their integral shield plugs. The modules would be inserted into the shielding structure from the plasma side. Each module would be retained by bolting the module attaching flange at the back side of the shield. Each module would have stainless steel elbows inertially welded to its helium inlet pipe (SAP) and its helium outlet pipe (prior to insertion into the shield structure). Each sector would consist of 1345 modules. The stainless steel helium inlet and outlet manifolds would then be installed (see Figure 6 for a schematic illustration of the placement of these manifolds). Assembly would continue with installation of the stainless steel coolant lines (see Figure 5 for placement of the shield coolant lines). Assembly of the shield structure and blanket would be completed with installation of the helium inlet and outlet headers (see Figure 6 for placement of these headers). The shield - blanket - first wall structure would be installed in the magnet envelope. The vacuum passages to the divertor slot would be connected, and would pass through the magnet. The helium coolant lines to the divertor plates would be connected. A description of the divertor system will follow in Section II-E.

The primary objective for integrating the minimum activity blanket into the previously designed UWMAK-I magnet envelope was twofold: (1) to demonstrate the design feasibility, and (2) to verify by calculation that the low activity is achieved. Quantitative data regarding these features of the minimum activity blanket will be discussed in following sections. It suffices here to state that the blanket-induced activity will decay to a safe value within 1 week after reactor shutdown (following the decay of sodium-24, $t_{1/2} = 15$ hr). Once this safe level is achieved, maintenance on the blanket and first wall could be performed from inside the torus. The minimum activity blanket system is thus maintainable, on a modular basis. During the integration studies for this design, maintainability was assumed to be a key factor. Referring to Figure 5, access is provided to the attaching flange and helium piping connections for $\sim 50\%$ of the modules in a given sector. Thus, removal of single modules, or groups of modules, is possible after a short interim decay period. In actuality, a system with more maintenance capability could be designed by resizing the magnet and torus to allow full access to all module attaching flanges. Although the detailed design study performed by the University of Wisconsin on Tokamak plants indicated that costs scaled directly as magnet size, no consideration of maintainability was included in their study. Experience gained with nuclear fission plants would indicate that maintainability is a key factor in the operation of any such facility. Design modifications which would increase the magnet envelope and allow easy accessibility to individual modules should prove very cost effective. However, it is not the purpose of this report to prove this point, but only to indicate it. Possible alternative module designs which would contain integral failure sensing devices could be incorporated into the basic minimum activity blanket system. These design possibilities will be further discussed in Section IV.

It cannot be too greatly stressed that leak detection, leak prevention, and module maintenance must be included in a viable first wall and blanket system. Coolant leakage into the plasma cavity (be it liquid metal, gaseous helium, or other) cannot be tolerated in a fusion system. Upon entering the plasma cavity, coolant atoms would charge-exchange with plasma ions, causing rapid transfer of the plasma energy to the leakage point. For this report, preliminary

calculations indicate that mechanisms exist to transfer several thousand thermal megawatts to a group of one to five modules. Such energy transference can vaporize sections of the first wall, shield, and magnet structures.

C. POWER CONVERSION SYSTEM

The design of the electrical power conversion system for a minimum activity blanket Tokamak Reactor will differ from the original design provided in UWMAK-I, because of the use of helium coolant. Each toroid sector will be equipped with a modular steam generator and helium circulator. The design of this unit will be similar to that used in current high-temperature gas cooled reactors (HTGR's).⁽⁴⁾ An illustration of a possible design is presented in Figure 7. Helium exiting the blanket modules is collected in the exit manifolds and transferred to the inlet of the steam generator; it passes through the shell side of the steam generator and exits to the shell side of the reheater. Helium exits the reheater and passes into the suction side of the helium circulator. A bypass line transfers 1% of the helium flow to the tritium removal system (see Section II-E for a detailed description of this system). Helium exiting the circulator is returned to the helium inlet manifold.

The modular steam generator is designed to provide: (1) minimum pressure drop through the tube bundle at low absolute helium pressures (10 atm), and (2) minimal tritium diffusion from the helium coolant to the steam circuit. The first feature is accomplished by providing finned steam generator tubes, and by increasing the tube spacing to accommodate these fins. Preliminary calculations indicate that reasonable pressure drops through such a bundle can be provided without significantly sacrificing heat transfer capability and size. To minimize tritium diffusion, the steam generator and reheater tubes are fabricated as a stainless steel - aluminum - stainless steel layered structure. Analysis indicates that such a structure will significantly reduce tritium diffusion in steam generator units. Fabricability of such systems is being demonstrated by experimental work on the Reference Theta Pinch Reactor (RTPR) at Los Alamos Scientific Laboratory (LASL).⁽⁵⁾

The helium circulator design is based upon that used in current HTGR technology. A special seal design will be incorporated to prevent leakage of

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Figure 7. Steam Generator Design

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steam at high pressure into the helium circulation system. This seal system is shown schematically in Figure 8. Low-pressure cavities are provided for the escape and capture of helium and steam. Steam is returned to the turbine seal-bleeding systems, while the helium is transferred to the tritium processing systems. Other options should be considered, including magnetically coupled shafts and diaphragm seals.

A major concern with the use of helium cooled systems is the large power required to circulate the helium. However, when comparing liquid metal cooled systems to helium cooled systems, several important points must be made. Liquid metals develop high pressure drops when pumped in the presence of magnetic fields, due to magnetohydrodynamic effects. Complex flow paths are necessary, because liquid metal flow in a magnetic field is laminar. Liquid metal heat exchangers must therefore be located far from the blanket and magnet structure, to minimize magnetohydrodynamic effects in the steam generator passages. Helium cooled systems do not suffer from magnetohydrodynamic pressure drop increases, and the steam generators may be located integrally with the torus sectors. This minimizes the primary coolant piping lengths, and minimizes the helium coolant inventory. It has been convenient to quote pumping power requirements for systems in terms of the required hydraulic power (in megawatts). However, this method does not account for different means of pumping liquid metals vs gaseous coolants. Gaseous coolants may be pumped by direct utilization of steam-driven circulating turbines. Thus, the conversion efficiency from thermal reactor power to helium circulating hydraulic power involves only the turbine efficiency and the helium circulator efficiency. Liquid metal systems necessarily require electric power for pumping, either for electromagnetic induction pumps or motor-operated centrifugal pumps. The conversion of reactor thermal power to liquid metal circulating hydraulic power thus involves the electrical cycle efficiency for the plant, the electric motor efficiency, and the centrifugal pumping efficiency. The overall thermal to hydraulic efficiency for a liquid metal system is thus at least a factor of four less than that for a helium circulation system (based on a plant efficiency of $\sim 30\%$ and an electrical motor efficiency of $\sim 90\%$). It is obvious that helium cooled systems can tolerate increases in the hydraulic pumping power without

Figure 8. Helium Circulator Seal

severely sacrificing total plant efficiency. This will be further demonstrated by the performance data of the minimum activity blanket system, as presented in Section II-D.

A conventional steam cycle is used to convert the thermal power delivered to the steam generator into electrical power. A schematic of the steam cycle is presented as Figure 9. Two identical three-stage turbine - electrical-generator systems are used to provide the 1850 MWe of output power. Each steam system employs a full-flow ion-exchange system for controlling the chemistry of the condensate stream. No continual "blowdown" of the condensate is thus required for the maintenance of the water chemistry. Loss of turbine load events requiring dumping of the steam systems are handled by venting into holding tanks. These features of the steam cycle are extremely important, because they significantly reduce the amount of tritium and/or tritiated water which is released through plant effluents. A summary of the steam cycle performance at rated operated conditions is presented in Section II-D.

D. PERFORMANCE

Basic performance parameters for the controlled thermonuclear reactor, as designed with a minimum activity blanket, are presented in Table 2. The basic reactor design provides 5000 MWt to the blanket. An additional 25 MWt is developed in the shield plugs. The net electrical output is 1850 MWe. This provides a plant net efficiency of 36.8%. Three percent of the total thermal power is recirculated to provide pumping power for the primary helium coolant. This plant design is limited to a maximum of 36.8% net electrical efficiency by the maximum outlet temperature of the blanket. This, in turn, is limited by the maximum temperature of the module's structural material, SAP. Approximately one-third of the helium coolant inlet-to-outlet temperature differential is developed in the module structural material. Thus, the module maximum outlet temperature is limited to $T_{He, m}$, as presented in Equation 1.

$$\begin{aligned} T_{He, m} &\leq T_{SAP, m} + 2(T_{SAP, m} - T_{He, i}) \\ &\leq 3T_{SAP, m} - 2T_{He, i} \end{aligned} \quad \dots (1)$$

Figure 9. Steam Cycle Schematic Diagram

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TABLE 2
MAB-CTR PLANT OPERATING CONDITIONS

Thermal Power From Blanket (MWt)	5000
Net Available Thermal Power From Shield (MWt)	25
Total Net Thermal Power (MWt)	5025
Net Electrical Power Generated (MWe)	1850
Net Efficiency (%)	36.8
Thermal Power Fraction for Helium Coolant Pumping (%)	3
Tritium Breeding Ratio (tritons/fusion neutron)	1.2
First Wall Neutron Flux (MWt/m ²)	1.25
Total Fusion Energy Produced/Total First Wall Area (MWt/m ²)	1.77
Total First Wall Area (m ²)	2830

where

$T_{\text{SAP}, m}$ = maximum SAP temperature (strength controlled)

$T_{\text{He}, i}$ = helium inlet temperature

It is thus possible to increase the helium module outlet temperature by decreasing the helium inlet temperature or increasing the structural material temperature capability. However, to utilize minimum activity materials, such as SAP, a maximum material temperature must be restricted to provide adequate stress margins during reactor operation. This maximum material temperature is not expected to greatly increase due to material selection, for any viable aluminum alloy. Further reduction in the helium inlet temperature is prohibitive, and requires reduction in the steam cycle maximum pressure, due to reduction in the pinch point temperature differential.

The controlled thermonuclear reactor with minimum activity blanket provides a breeding ratio of approximately 1.2 tritons/fusion neutron. This breeding ratio is a slight reduction below Design 4A.⁽¹⁾ This reduction results from use of sublength blanket modules in the inner and outer periphery of the torus. The breeding ratio may be increased, if required, by increasing the inventory of lithium-aluminum alloy and/or increasing the beryllium inventory. However,

both increases present severe additional plant costs, due to the high expense of isotopically enriched lithium and high-purity beryllium. Thought should be given to the feasibility of designing two types of fusion reactors. The first type, an optimized breeder reactor, could provide an extremely high breeding ratio, possibly 1.8 to 1.9 tritons/fusion neutron. The cost of such a plant would be absorbed by minimizing the cost of the second type of plant, a power plant. This would provide minimal tritium breeding (0.9 breeding ratio), and would receive tritium from the breeder plant. No beryllium, and a small inventory of LiAl, would be required for such a plant. The breeder - pure burner concept might also be advantageously employed. Burner plants, located near large metropolitan areas, could provide power from remote-breeder-supplied fuel.

Performance parameters for a blanket module are given in Table 3. Note that these numbers represent an average value; deviations from these numbers exist, due to the presence of sublength modules and variations in the plasma density within the torus. Each module develops ~ 310 kWt, with 1 kWt additional being developed in the shield plug, which is integral with the module. Table 3 indicates the division between power deposited in the "cold" zone (first wall) and the "hot" zone (interior) of the module. Heat leakage from the "hot" zone to the "cold" zone is included. The helium flow rate per module is 1116 lb/hr. This amounts to a total flow rate for the plant of 18×10^6 lb/hr. Helium enters the module at 245°C and exits at 650°C. The helium operating pressure is 10 atm. The total helium inventory in the primary coolant circuit is 30,000 lb (as scaled from current HTGR designs).

E. TRITIUM PRODUCTION AND TRITIUM PROCESSING

Tritium production occurs directly in the lithium-aluminum alloy, located on the periphery of the cylindrical shell of each blanket module. Bred tritium is released directly to the helium coolant, and is carried by the helium coolant stream through the primary helium circuit. A 1% bypass line is taken off the helium circuit, and carries tritium and helium to the processing system. The following paragraphs discuss the production, retention, removal, and processing of tritium in the controlled thermonuclear reactor.

TABLE 3
MAP-CTR MODULE OPERATING CONDITIONS

Module Diameter (cm)	45
Total Module Power (kWt)	311
Power Deposited in Shield Plug (kWt)	1
Power Deposited in "Cold" Zone (kWt)	78
Power Deposited in "Hot" Zone (kWt)	232
Power Removed From "Cold" Zone (kWt)	99
Power Removed From "Hot" Zone (kWt)	211
Heat Leakage, "Hot" to "Cold" Zone (kWt)	21
Helium Flow Rate (lb/hr per module)	1116
Helium Inlet Temperature (°C)	245
Helium Outlet Temperature (°C)	650
Helium Operating Pressure (atm)	10
Pressure Drop Across Module (psi)	3
Wall Temperature Drop, Maximum (°C)	45

Tritium is bred in the fusion blanket at a rate of ~ 30 gm/hr (based on the 1.2 breeding ratio at 5000 MWt). Based on experiments performed at Brookhaven National Laboratory,⁽⁶⁾ the mean residence time of tritium in the lithium aluminum of the blanket is ~ 1 hr. The total lithium blanket inventory is thus 30 gm, or $\sim 300,000$ Ci. Tritium released to the helium coolant reaches equilibrium at a concentration of 0.35 appm. The total tritium inventory in the primary coolant circuit is thus 3.6 gm (assuming 1% bypass flow to the processing system, with complete removal in the processing system). The scavenging of tritium from the lithium-aluminum alloy in the blanket is equilibrium-limited for the lower limit of Sievert's constant of tritium in LiAl ($k_s^2 = 2 \times 10^7$), estimated from BNL experiments. The partial pressure of tritium in the lithium-aluminum alloy is 2×10^{-3} torr, and the partial pressure of tritium in the helium is 2.7×10^{-3} torr.

The tritium processing and storage systems will use a zirconium matrix for capture of the tritium. A schematic of the tritium production and processing system is shown in Figure 10. Tritium input rate to the traps and to the storage

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Figure 10. Tritium Inventory and Tritium Processing Systems

facility are indicated on Figure 10. Ten kilograms of tritium was allocated to the fuel storage and processing systems. During a loss of tritium removal capability, this fuel storage could provide several days' operation while repairs are made. Eight kilograms of tritium are allocated to the storage areas, while 2 kg are allocated to the fuel processing system, which provides purification and liquefaction of the tritium for eventual pellet manufacture.

The tritium removal system was designed with a zirconium matrix, in order to utilize the low partial pressure of tritium in zirconium at low temperatures. At the temperature of the tritium stream (245°C), the partial pressure of tritium in ZrH_2 is $\sim 10^{-6}$ torr. The partial pressure of tritium in the processing stream of helium is 2.7×10^{-3} torr, thus assuring essentially 100% removal in the processing system. Approximately 6500 ft^2 of trapping surface area are required for the tritium removal. The helium flow rate through the processing system is 6500 ft^3/sec , with an assumed 1 ft/sec velocity through the bed to limit the trap velocity well below the fluidization velocity of the material. The total zirconium inventory in the bed can be as low as 2 lb; however, at least 10 to 20 lb would probably be necessary to assure favorable reaction kinetics. The tritium bed would be regenerated every hour. Several beds would be used, with one in the processing system and the others in the regeneration system. It is thus assured that no large quantities of tritium are localized within the processing facility. The maximum tritium stored in a given processing bed would be 30 gm, or 300,000 Ci.

Tritium fuel storage for the thermonuclear reactor plant is also provided by zirconium beds. A minimum of eight beds will be utilized, thereby reducing the total tritium inventory per bed to 1 kg. These storage beds could be located at isolated, separated points along the plant periphery. The size of the bed is minimal ($< 2 \text{ ft}^3$, with shielding). The beds can thus be mounted in small underground vaults, minimizing the possibility of multiple release accidents. Under normal conditions, fuel would be taken from the zirconium trap to the purification and liquefaction system. Bred tritium, not required for plant fueling, would be processed and put into fuel storage.

It is important to note that the minimum activity blanket system with tritium breeding in solid lithium-aluminum alloy can provide extremely low concentration of tritium in the helium coolant circuit. This is important from the

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Figure 11. Divertor Design and Block Diagram

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standpoint of total tritium inventory. The relatively high partial pressure of tritium in the helium circuit (as compared to the partial pressure of tritium in a liquid lithium circuit) allows essentially complete removal of the tritium by the zirconium processing system. However, this relatively high tritium pressure increases the possibilities for tritium diffusion through the heat exchanger and into the steam system. This diffusion has been minimized by use of a layered structure for the steam generator tubes (stainless steel - aluminum - stainless steel). In addition, back diffusion of electrolyzed hydrogen from the steam system is minimized by this design, an important feature to protect the tritium scavenging system from a large hydrogen removal burden.

F. DIVERTOR SYSTEM DESIGN

The torus divertor system, which scavenges particles escaping from the magnetic containment through the particle loss cone, has been modified for the minimum activity blanket design. The liquid divertor plates have been removed. Palladium-coated, helium-cooled SAP tubes have been substituted as the divertor capture mechanism. Palladium acts as a getter and charge exchange surface for deuterium and tritium ions escaping from the plasma surface. The deuterium and tritium neutrals are removed through low-pressure pumping provided by helium cryopumps. The basic divertor cross section can be seen in Figure 5. A block diagram for the divertor and pumping system can be seen in Figure 11.

The use of palladium-covered SAP tubes as a deuterium and tritium getter was proposed by Lawrence Livermore Laboratory for use in the Fusion Engineering Research Facility (FERF).⁽⁷⁾ The unique capability of the palladium matrix to "admit" hydrogen species allows capture of high-energy released ions with minimal or no matrix damage. Referring to Figures 5 and 11, ions released from the plasma surface traverse through the separatrix, and are driven to collision with the palladium by magnetic field lines. Neutralization of the deuterium and tritium ions occurs after initial capture in the palladium metal. The deuterium and tritium neutrals are subsequently released from the palladium matrix, and are removed by the vacuum system. The palladium-on-SAP system has been postulated to provide minimal surface damage to

either of the constituent metals, thus providing a viable method of capturing high-energy ionized hydrogen species. Plasma impurities (mostly aluminum) which plate onto the divertor tube surfaces and limit the efficiency of the palladium coating will, however, necessitate periodic maintenance of the divertor tubes.

The original UWMAK-I design used liquid lithium on the divertor plates to capture tritium ions and to provide a significant percentage of the overall sector vacuum pumping capability. The palladium-covered SAP tubes of the present design only provide a surface for the capture and subsequent release of the deuterium and tritium. This greatly increases the vacuum pumping requirements for each sector of the fusion plant. However, use of helium cryopumping techniques will not require a significant increase in the recirculating power fraction for vacuum pumping. Although pumping power will increase, it must be noted that most of the kinetic energy of the neutrals is removed before they reach the cryopumps. These particles lose energy during collision and charge-exchange with the palladium, and the neutrals lose energy through collision with the vacuum system walls. Assume that the particles arrive at thermal equilibrium with the vacuum system walls at 300°C. Each particle thus has ~ 0.1 eV thermal energy upon entering the cryopumps. Because the electrical-to-thermal refrigeration efficiency at 4°K is $\sim 0.3\%$, 30 eV of electrical energy is required to refrigerate the cryopumps. At 37% plant efficiency, this is 90 eV of thermal power per particle generated. This can be compared to an average equivalent energy of 2 MeV released in the plasma per particle arriving at the divertor (10% burnup at 20 MeV/fusion). This is indeed a small percentage of the total thermal power generated.

III. ENVIRONMENTAL CONSIDERATIONS

The evaluation of environmental effects for preliminary safety analysis requires postulation of probable accidents, consideration of radiological release paths, and calculation of the total radiological inventory at the time of the accident or release. Two sources of radioactive materials exist in the plant considered here. The first is the tritium in the processing and storage system, whose inventory has been previously discussed in Section II-E. The second is the neutron-induced radioactivity in the blanket and structural materials. This section will analyze the total radioisotope inventory in the minimum activity blanket system, and will use this information to evaluate the potential radiological environmental effects resulting from operation of the plant.

A. GENERATED BLANKET ACTIVITY

Calculation of the blanket activity resulting from first, second, and third generation radionuclides was previously performed by BNL in Reference 1. A homogeneous seven-zone one-dimensional ANISN calculation was performed to determine individual radionuclide concentrations. The results of this work have been used to calculate the activity in the current blanket. Results from the calculated neutron balance were used to generate activities in individual zones. Corrections for increased power and modified volume fractions of materials were included. As suggested by the University of Wisconsin, Reference 8, a factor of two reduction was assumed in the activity generation calculated by BNL. This is a calculational difference in ANISN resulting from incorporation of a P_3 scattering kernel, as used by Wisconsin, vs a P_0 scattering kernel, as used by BNL.

The activity created in the blanket of this system after its 3-year operational life is presented in Table 4, for isotopes with a total activity of 1 Ci or more. Also shown in Table 4 are the parent and source of the parent for each isotope, the half-life of the isotope, and the maximum disintegration gamma energy for the isotope. Note that, of the sixteen isotopes listed (excluding tritium), eleven have reached saturation (equilibrium) and five have not. Of these five, only Al^{26} and Fe^{55} present a significant activity. Thus, operation of the blanket for longer than 3 years is not limited by activity buildup.

TABLE 4
MAB-CTR ACTIVATION FOLLOWING 3-year OPERATION

Radionuclide	Parent(s)	Source	Half-Life	$E_{\gamma \text{ max}}$ (MeV)	Total Activity (Ci)
$^1_1\text{H}^3$	$^3_3\text{Li}^6, ^3_3\text{Li}^7$	Blanket	12.3 years	no γ	3.4×10^6
$^4_4\text{Be}^{10}$	$^6_6\text{C}^{13}$	Graphite	1.6×10^6 years	no γ	3.5×10^1
$^6_6\text{C}^{14}$	$^6_6\text{C}^{13}$	Graphite	5.7×10^3 years	no γ	3
$^{11}_{11}\text{Na}^{22}$	$^{13}_{13}\text{Al}^{27}$	SAP	2.6 years	1.3	1
$^{11}_{11}\text{Na}^{24}$	$^{13}_{13}\text{Al}^{27}$	SAP	15 hr	2.8	1.2×10^9
$^{13}_{13}\text{Al}^{26}$	$^{13}_{13}\text{Al}^{27}$	SAP	7.3×10^5 years	1.18	8.0×10^2
$^{20}_{20}\text{Ca}^{45}$	$^{22}_{22}\text{Ti}^{47}, ^{22}_{22}\text{Ti}^{48}$	Structure	163 days	no γ	1.0×10^3
$^{21}_{21}\text{Sc}^{46}$	Ti	Structure	84 days	1.1	2.7×10^3
$^{21}_{21}\text{Sc}^{47}$	Ti	Structure	3.4 days	0.16	2.4×10^3
$^{21}_{21}\text{Sc}^{48}$	Ti	Structure	1.8 days	1.3	2.5×10^4
$^{25}_{25}\text{Mn}^{54}$	$^{25}_{25}\text{Mn}^{55}, ^{26}_{26}\text{Fe}^{54}$	SAP Impurity	303 days	0.84	5.0×10^3
$^{26}_{26}\text{Fe}^{55}$	$^{26}_{26}\text{Fe}^{56}$	SAP Impurity	2.7 years	0.007	1.1×10^3
$^{30}_{30}\text{Zn}^{65}$	$^{30}_{30}\text{Zn}^{66}$	SAP Impurity	244 days	1.1	8.0×10^2
$^{40}_{40}\text{Zr}^{89}$	$^{40}_{40}\text{Zr}^{90}$	SAP Impurity	3.3 days	1.71	4.2×10^3
$^{40}_{40}\text{Zr}^{95}$	$^{40}_{40}\text{Zr}^{96}$	SAP Impurity	65 days	0.76	6.0×10^2
$^{50}_{50}\text{Sn}^{117\text{m}}$	$^{50}_{50}\text{Sn}^{118}$	SAP Impurity	14 days	0.16	1.0×10^3
$^{50}_{50}\text{Sn}^{119\text{m}}$	$^{50}_{50}\text{Sn}^{120}$	SAP Impurity	245 days	0.06	1.5×10^3

The total initial blanket inventory is $\sim 1.2 \times 10^9$ Ci, equivalent to a volumetric activity of ~ 0.25 Ci/cc (based on a total blanket volume of 5×10^9 cc). This results mostly from sodium-24 [$\text{Al}^{27}(\text{n}, \text{d})\text{Na}^{24}$] which has a 15-hr half-life, and thus decays to negligible levels in 2 weeks. The time history of the specific activity [Ci/W(th)] of the blanket is presented in Figure 12. For comparison, the specific activity of the original UWMAK-I blanket is also presented.⁽²⁾

The biological hazard potential (BHP) is a tool often used for comparison of the effective environmental hazard associated with a given radioactive inventory. For a given radionuclide, the BHP is that volume of water or air required to dilute the isotopic activity of the given radionuclide to its maximum permissible concentration in water (MPC_w), or air (MPC_a). The MPC_w used is for continuous exposure to the critical organ of nonoccupational personnel. Table 5 presents the MPC_w 's and BHP_w 's for each of the isotopes in Table 4. MPC_w values were obtained from 10 CFR 20,⁽⁹⁾ or were calculated, based on ICRP-2⁽¹⁰⁾ and References 11 and 12 (see Appendix A). The volumetric activity of each isotope in nonreclaimed waste is also shown. Certain assumptions were made, in order to approximate the volume of activated material which would not be reclaimed. Upon removal of the module from the blanket, it was assumed that lithium-aluminum alloy, beryllium, and graphite would be reclaimed. Cleanup would include removal of these individual materials from the modules, with separate storage. The shield plug would be reclaimed, and could be used for several 3-year operational periods. An insignificant amount of activation occurs in the SAP and titanium of the shield. Lead activation within the shield plug is considered minimal, because of the placement of the lead behind the boron carbide in the layered shield structure. Reclamation of the SAP, aluminum oxide, and titanium of the blanket would not occur. The volumetric activities in Table 5 are thus calculated for the initial activity of the nonreclaimed radionuclides after 3 years of reactor operation.

Another comparative tool for the study of effective environmental hazards of a set of radionuclides is calculated based on 10 CFR 20.⁽⁹⁾ This quantity can be called the effective concentration hazard (ECH), as defined by Equation 2:

Figure 12. MAB-CTR Specific Activity History

TABLE 5
MAB-CTR BIOLOGICAL HAZARD POTENTIAL AND VOLUMETRIC
ACTIVITY FOLLOWING 3-year OPERATION

Radionuclide	Initial Specific Activity [Ci/W(th)]	MPC _w * (μCi/ml)	BHP [m ³ water/W(th)]	Initial Volumetric Activity† (μCi/ml)
1H^3	6.8×10^{-4}	3×10^{-2}	2.3×10^{-2}	==
4Be^{10}	7.0×10^{-9}	$7.2 \times 10^{-5} \S$	9.7×10^{-5}	==
6C^{14}	6×10^{-10}	8×10^{-3}	7.5×10^{-8}	==
11Na^{22}	2×10^{-10}	3×10^{-5}	6.7×10^{-6}	5×10^{-4}
11Na^{24}	2.4×10^{-1}	3×10^{-5}	8.0×10^3	5×10^5
13Al^{26}	1.6×10^{-7}	$5.7 \times 10^{-6} \S$	2.8×10^{-2}	3.5×10^{-1}
20Ca^{45}	2×10^{-7}	9×10^{-6}	2.2×10^{-2}	1.3×10^{-1}
21Sc^{46}	5.4×10^{-7}	4×10^{-5}	1.4×10^{-2}	3.4×10^{-1}
21Sc^{47}	4.8×10^{-7}	9×10^{-5}	5.3×10^{-3}	3.0×10^{-1}
21Sc^{48}	5×10^{-6}	3×10^{-5}	1.7×10^{-1}	3.2
25Mn^{54}	1×10^{-6}	1.0×10^{-4}	1.0×10^{-2}	2.5
26Fe^{55}	2.2×10^{-7}	8×10^{-4}	2.8×10^{-4}	5.5×10^{-1}
30Zn^{65}	1.6×10^{-7}	1×10^{-4}	1.6×10^{-3}	4.0×10^{-1}
40Zr^{89}	8.4×10^{-7}	$6.6 \times 10^{-5} \S$	1.3×10^{-2}	2.1
40Zr^{95}	1.2×10^{-7}	6×10^{-5}	2.0×10^{-3}	3.0×10^{-1}
$50\text{Sn}^{117\text{m}}$	2×10^{-7}	$4.1 \times 10^{-4} \S$	4.9×10^{-4}	5.0×10^{-1}
$50\text{Sn}^{119\text{m}}$	3×10^{-7}	$3.7 \times 10^{-4} \S$	8.1×10^{-4}	7.8×10^{-1}

*Continuous, nonoccupational exposure — minimum value for soluble-insoluble radionuclides

†Based on nonreclaimed materials only — radionuclides indicated by "==" are completely reclaimed

§Calculated — see Appendix A

Figure 13. Effective Concentration Hazard History

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$$ECH = \sum_{i=1}^n \frac{C_i}{MPC_i} \quad \dots (2)$$

where

C_i = volumetric activity of isotope i

MPC_i = maximum permissible concentration for isotope i
(continuous in water).

The ECH is basically the volumetric activity of a nuclide divided by the maximum permissible concentration for the nuclide, summed over all applicable nuclides. Figure 13 presents the time history of the ECH for nonreclaimed blanket materials. For the purpose of classifying radioactive wastes in waste disposal programs, high level wastes are defined as materials with an $ECH > 10^6$ (see Reference 13), intermediate level wastes, an ECH between 10^4 and 10^6 , and low level wastes, an $ECH < 10^4$. Thus, the waste materials from the 3-year operation of the minimum activity blanket fusion plant are considered high level wastes for only the first week following plant shutdown, and then become intermediate level wastes. The total volume of these wastes is $\sim 2000 \text{ m}^3$ ($\sim 70,000 \text{ ft}^3$). A dilution factor of sixteen would be required for reduction to the low level classification, after a holding period of 2 years.

B. EFFLUENTS DURING NORMAL OPERATION

During normal operation of the minimum activity blanket controlled thermonuclear reactor (MAB-CTR), radioactive material from the plant systems will be released through various boundaries to the environment. The different isotopes involved have been discussed in previous sections. The principal radionuclides are tritium, which is bred as a reactor fuel, and blanket-structural activation products. The former was discussed in Section II-E. The latter was discussed in Section III-A. In Section II-E, it was pointed out that release paths for tritium have been minimized during normal operation of the plant. All other radionuclides are in solid form, and cannot be released except through plant accidents (to be discussed in Section III-C). We thus concern ourselves only with possible release paths for tritium, in this section.

Tritium is the principal fuel in the operation of the plant, and is a normal product of plant operation. Previous sections have discussed its production, processing, storage, and control. Several possible release paths exist for tritium in the plant. Among these are diffusion paths from the primary coolant system, either into the plant containment system or through the steam generator and into the steam system; leakage and diffusion paths in the tritium processing system and the liquification and purification system; and from the final storage system.

Except for accidents, plant designs will be such that the tritium release to the vacuum system or the steam system is minimal. Radiological controls which have been placed on the current designs of pressurized water reactors and liquid metal fast breeder reactors require minimum release of tritium in any form from these plants. These plants are thus equipped with tritium scrubbing systems, and are designed with minimum blowdown steam cycles. The thermonuclear plant will be designed with "zero blowdown" and no steam "dumping." Any tritium escaping into the containment system is captured by the containment vacuum pumps (remembering the first-level containment is held at $\sim 10^{-3}$ torr). The output of this vacuum system is scrubbed, and the tritium returned to the processing system. The steam system has been designed for zero blowdown (i. e., no continual bleeding of the condenser outlet for water chemistry control). During turbine loss-of-load events, steam will be dumped into holding tanks. These events will thus not release tritiated water into the environment.

Thus, on a first approximation basis, no major release paths for tritium or other generated radiological wastes exist in the MAB-CTR. Minor diffusion paths and release paths will exist for tritium and other generated waste materials. However, the calculation of release rates from these paths cannot proceed until a firm design for plant systems and subsystems has been assumed. We therefore assume that the plant radiological effluents during normal operation are minimal, and that positive state-of-the-art measures can readily provide their control.

C. ACCIDENT ANALYSIS

The preliminary CTR plant accident review will be approached along the same lines followed for fission plants. The approach is illustrated in the preliminary safety assessment of UWMAK-I,⁽¹⁴⁾ which describes the analytical concepts in detail, and defines the three "levels" of safety which will provide "defense-in-depth" against non-normal radioactivity releases.

Section III-D will be concerned with a preliminary safety analysis. As such, this section will develop a list of plant accidents based on hypothetical, low-probability events. Logical paths between the initiating events and final consequences will be developed; and, in one case, the safety assurance diagram (SAD) will be demonstrated as a convenient tool for this purpose.

For purposes of comparison with previous CTR safety analyses, this report will define the plant as follows: (1) the exclusion area radius is 600 m, based on current LMFBR designs, (2) four containment regions exist (viz, the first wall, magnet structure wall, low vacuum boundary, and reactor building), and (3) the reactor building is vented to the environment through a 50-m (effective height) stack. The site boundary has not been determined, based on the 25-rem requirement of 10 CFR 100. This dose value will, however, be used in this report for comparison purposes.

D. PRELIMINARY SAFETY ANALYSIS

A first step in safety analysis is to prepare a list of events. A preliminary list for the MAB-CTR is given in Table 6. Of course, a more detailed list would require more design definition and detailed familiarity with this design. This is, however, the fundamental starting point for any safety analysis, and, as such, should be given much thought and detailed iteration.

Note that the list of accidents presented in Table 6 consists of plausible events, based on accidents external to the plant or malfunctions occurring in the plant. In this section of this report, we will consider five accident events that result from several possible initiating malfunctions. These events concern the release of tritium from the plant containment, either at ground level or through the 50-m containment stack, and are summarized in Table 7. One

TABLE 6
POSSIBLE MAB-CTR ACCIDENTS
(Preliminary)

<u>Accidents External to Plant</u>	
Earthquake Tornado Flood Airplane "Collision" Event Sabotage Loss of Off-Site Power	These accidents are to be considered initiating events for plant malfunctions.
<u>Plant Malfunctions</u>	<u>Initiating Mechanisms</u>
Loss of Primary Cooling With reactor at power Shutdown system activates Failure of shutdown system With reactor "down"	Primary pipe rupture Primary system blockage Helium circulator failure Loss of site power
Loss of Secondary Cooling With reactor at power Shutdown system activates Failure of shutdown system With reactor "down"	Secondary pipe rupture Secondary system blockage Pump failure Loss of site power
Steam Generator Tube Failures	
Magnet Failures	Helium warmup Super-normal transition
Helium Warmup Accidents	Failure of helium containment boundary
Overpower Accidents	Increase in magnetic field Increase in fuel injection rate Failure of control system
Tritium Processing System Accidents	Processing system pipe failure Zirconium storage bed failure
First Wall Failure	Loss of cooling Overpower event Plasma energy "dumping" Magnet failure with arcing Misdirection of neutral beam

TABLE 6 (Continued)

<u>Plant Malfunctions</u>	<u>Initiating Mechanisms</u>
Missile Accidents	Missile from helium warmup event Magnet rupture
Loss of Shield Cooling	Shield coolant pipe rupture
Loss of Divertor Tube Cooling	Divertor coolant pipe rupture
Failure of Vacuum System	Loss of cryopump refrigeration

TABLE 7
MAB-CTR ANALYZED ACCIDENTS

<u>Accident</u>	<u>Description</u>
1	Release of tritium in one 30° sector of blanket (0.3 gm) Release occurs through 50-m stack
2	Release of tritium in one 30° sector of blanket (0.3 gm) Release occurs at ground level
3	Release of 1% of tritium in the fuel processing system (20 gm) Release occurs through 50-m stack
4	Release of 1% of the total plant tritium inventory (100 gm) Release occurs through 50-m stack
5	Vaporization of two blanket modules and their associated shielding and structure Venting of condensed particulates through 50-m stack Release of tritium in one 30° sector through 50-m stack

event additionally concerns the release of a vaporized section of the first wall, consisting of the materials of two modules and their shield plugs, plus the tritium contained in one sector (30°) of the torus. The following paragraphs discuss these accidents.

The first accident is the release of the tritium contained in the total helium inventory of one sector. The total tritium inventory in the primary helium circuit is 3.6 gm; thus, this accident represents the release of 0.3 gm (3000 Ci) of tritium, assumed to occur through the 50-m stack which vents the containment building. Release of the radionuclide at elevated levels minimizes the delivered dose at ground level. The mechanistics for this accident can be derived from Table 6. Plausible initiating mechanisms are failure of the primary piping system, loss of helium circulator seal, or loss of the steam generator shell. The low vacuum system (10^{-3} torr) is designed to contain such a release, with no effect on the environment. If it is assumed that the initiating mechanism was related to an external plant accident, such as an earthquake, which is further assumed to cause the breach of the low-vacuum containment structure, there could be release of the tritium through the stack. To provide an estimate of the hazard from this accident, AI has performed radiological calculations to determine the consequences of this release to the environment. The criteria from the AEC Safety Guide 3⁽¹⁵⁾ were used for the dose calculations.

Three processes contribute to the total dose: skin absorption, inhalation, and cloud immersion. This dose will be calculated at various distances from the release point, with doses compared at the assumed 600-m site boundary. The calculations assume a short-term release (<8 hr release time), which is further assumed to have occurred under worst-case meteorological conditions. In the case of an elevated release, the worst-case meteorological conditions are not the same for all distances from the point of release. Under certain meteorological conditions, the ground level concentrations can maximize at distances >600 m from the release point, because the cloud or plume does not reach ground level until it has moved downwind for considerable distances. For this reason, elevated releases are best evaluated on the basis of the envelope of the entire range of meteorological conditions. Figure 1A of AEC Safety Guide 3⁽¹⁵⁾

provides data on the consequences of an elevated release, determined by considering the locus of the maxima of the concentrations at ground level resultant from Pasquill Type A through F meteorological conditions. Pasquill Type A conditions provide a maximum integrated released concentration at ~ 250 m from the site boundary (based on a 50-m elevated release). Pasquill Type F conditions provide maximum released concentration at ~ 8000 m from the site boundary (based on a 50-m release height). Figure 1A of AEC Safety Guide 3 considers the maximum envelope described by all Pasquill type conditions, and thus provides the maximum concentration at any point from the site boundary. Based on this figure, the χ/Q value for the release location is $5.8 \times 10^{-5} \text{ sec/m}^3$. The 3000-Ci release provides a time-integrated concentration of 0.174 Ci-sec/m^3 at the 600-m boundary. Assuming a breathing rate of $3.47 \times 10^{-4} \text{ m}^3/\text{sec}$ for the first 8 hr of the inhalation (ICRP-2, Reference 10), we arrive at an integrated delivered release of $60 \mu\text{Ci}$. Assuming an intake dose of $2.06 \times 10^{-4} \text{ rem}/\mu\text{Ci}$ (see Appendix B), the inhalation dose at the 600-m boundary is 0.01 rem. The skin absorption dose from the exposure to a tritium atmosphere is approximately equal to the inhalation dose (ICRP-10, Reference 16), so that the skin absorption dose is also 0.01 rem. The cloud immersion dose can be evaluated from the χ/Q ratio and the average β energy of the tritium emission by using an infinite uniform cloud immersion dose calculation, as appears in Reference 17. For tritium releases, the immersion dose is $\sim 2\%$ of the inhalation dose, or $\sim 1\%$ of the total dose, and will be neglected here. Thus, the total dose from the elevated (50-m) release of the tritium contained in the helium of one sector of the MAB-CTR is 0.02 rem at the 600-m site boundary. The variation of this dose with distance appears in Figure 14, which shows that the maximum delivered dose for such a release occurs ~ 250 m from the release site.

The second accident considered on Table 7 concerns release of the same quantity of tritium, but is assumed to accompany rupture of the total containment, so that it is released at ground level, rather than through the 50-m stack. As before, the release of this 0.3 gm of tritium was assumed to occur under worst-case meteorological conditions, with a release time of < 8 hr. The criteria from AEC Safety Guide 3 were used for the dose calculations.

Figure 14. Total Integrated Dose for 0.3 gm Tritium Release Accident vs Distance

Figure 3A in AEC Safety Guide 3 provides the χ/Q values for various distances from the release point, for a ground level release with various release times. The resultant χ/Q for our case is $1.5 \times 10^{-3} \text{ sec/m}^3$ at the 600-m site boundary. Thus, the release of 3000 Ci of tritium at ground level provides a time-integrated concentration at the 600-m boundary of 4.5 Ci-sec/m^3 . Calculations of the inhalation, skin absorption, and cloud immersion dose used the same assumptions previously made for the calculation at the 50-m release height. The resultant total dose at the 600-m boundary is 0.33 rem. Again, the total dose vs distance from the release was plotted in Figure 14. Note that, in the case of a ground-level release, for all meteorological conditions, the dose decreases monotonically with increasing distance from the release point. The release of the total tritium inventory of one sector of the fusion plant, either at ground level or through the release stack, provides <1 rem at the site boundary. This is significantly less than the 25-rem maximum suggested by 10 CFR 100.

The third accident of Table 7 concerns the release of 1% of the total tritium in the processing system. The 1% release was arbitrarily chosen to provide order of magnitude estimates of the effect of release events. It is, however, a plausible event, because the tritium will be passing through several stages of equipment in the processing system. This accident releases a total of 20 gm of tritium. The release is again assumed to occur through the 50-m stack. Mechanisms for this release include a fire in the tritium processing area, which ruptures certain tritium containment vessels and vents the released tritium into the outer containment system and through the stack. Based on AEC Safety Guide 3 and the previous calculations made in this section, the total whole body dose at the 600-m boundary is 1.7 rem. This whole body dose still remains less than the 25 rem suggested in 10 CFR 100.

The fourth accident of Table 7 concerns the release of 1% of the total tritium inventory of the plant, including that in fuel storage. This is not necessarily meant to include physical release of tritium in fuel storage, but is an arbitrarily value, to provide insight for different accident situations. One percent of the total plant inventory is 100 gm of tritium. Release is assumed to occur at the 50-m elevation. This accident could result from a severe malfunction of the fuel

Figure 15. Safety Assurance Diagram – Vaporization of Modular First Wall – Fusion Blanket

processing system. Based on previous calculations, the whole body dose received at the exclusion radius would be ~ 8.5 rem. Again, this is less than the suggested 25 rem.

Based on the previous discussions, it is obvious that a release of tritium from the MAB-CTR provides a whole body dose of ~ 2.2 rem/gm released at ground level, or ~ 0.8 rem/gm released at 50 m. Approximately 11 gm of tritium, released at ground level, would provide a 25-rem dose at the site boundary.

The fifth accident considered from Table 7 concerns the vaporization of two modules, and the release of all radionuclides contained therein, to the environment. This release is assumed to occur through the 50-m stack, because assumed initiating events (plasma energy "dumping") should not cause a breach in building containment. It is accompanied by the release of all the tritium in one sector of the torus. The safety assurance diagram (SAD) is utilized here to provide insight into the possible mechanisms for vaporization and release of the modules. The safety assurance diagram for this accident appears as Figure 15. An explanation of the symbols used in this diagram appears in Figure 16. This is necessarily only a preliminary logistics path for progression to a module vaporization event. However, it illustrates plausible mechanisms through which this accident could occur. The accident is initiated by either of two mechanisms: (1) electron runaway in the plasma, which deposits a large amount of energy locally on the first wall, or (2) a small helium leak in the first wall, resulting in charge exchange at the plasma surface and "dumping" of the plasma energy back toward the first wall. Each of these events is protected by the line of defense (LOD) shown on the safety assurance diagram. The LOD indicates the design features of the plant or the inherent capabilities of the plant to minimize, or to protect against, the occurrence of such events. Both of these initiating mechanisms result in increased helium leakage and increased "dumping" of the plasma energy onto the first wall. If the leak is large enough, under certain conditions, "dumping" of the entire 3 GJ of plasma energy could occur against the first wall. This total energy is capable of vaporizing many more than two modules. However, we assume that the energy release cannot be so completely localized, and that vaporization of more than two modules

NORMAL BRANCHING

ASSUMED EQUAL PROBABILITY
ALL PATHS

ASSIGNED BRANCHING

MORE PROBABLE PATH

LINE-OF-DEFENSE (LOD)

DEFENSE AGAINST PROCESSION
THROUGH PATH

LOD IDENTITY NUMBER

SHUTDOWN DEVICE (SD)

I1 FOR AUTOMATIC SHUTDOWN
I2 FOR MANUAL SHUTDOWN

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Figure 16. Safety Assurance Diagram — Explanation of Symbols

does not occur. The number of modules vaporized would be minimized by the shutdown system, activated by low helium coolant pressure. Quenching of the plasma by a poisoning system is assumed as a shutdown mechanism. If module vaporization is further assumed to continue into the shield and part of the magnet system, the consequent thermal deposition can cause rupture of the magnet. The vaporized module is assumed to condense and collect on dust particles outside of the low-vacuum boundary, with the tritium and condensed vaporized modules released to the atmosphere through the 50-m stack. Utilization of the SAD as a tool in the evaluation of such events is obvious. Justification of each of the lines of defense provides documented descriptions of failure protections.

The module vaporization accident releases both radiological and biological toxins. Examples of radiological toxins are aluminum-26 and sodium-24. Examples of biological toxins are beryllium and lead. We will concern ourselves only with radiological effects of the vaporization accident. Table 8 presents a list of the radioactive materials released, by isotope, and the total amount of radioactivity, in curies, delivered at the 600-m boundary by inhalation, skin absorption, and cloud immersion. Inhalation doses were based on F factors, as calculated by the methods of ICRP-2⁽¹⁰⁾ and Reference 18. The calculations are discussed in Appendix B. These doses apply to the critical body organ indicated in Table 8. Immersion doses were based on the methods of Reference 17.

Note that the only significant dose (>1 rem) results from the short-lived sodium-24 activity (half-life of 15 hr). Current practices in LMFBR accident analyses incorporate the effective holdup time of the containment structure (with an assumed low leakage rate) into the released activity calculation. The analysis of the vaporization accident assumed that the containment boundary was penetrated, in order to highlight priorities among the areas of possible safety precautions in fusion plants. This assumed penetration is, of course, an extremely unlikely event.

The analysis of the vaporization accident is a start in the determination of the potential hazards of plant operation. The synergistic effects of the large amounts of stored energy in the plasma, superconducting magnets, and liquid helium were capable of vaporization of a section of the first wall. A radioactive

TABLE 8
MODULE VAPORIZATION ACCIDENT — ENVIRONMENTAL EFFECTS
AT 600-m SITE BOUNDARY

Radionuclide	Total Release (Ci)	Inhalation Dose (rem)	Critical Organ	Immersion Dose* (rem)
1H^3	3×10^4	$2.4 \times 10^{-1}\dagger$	Body Water	2.4×10^{-3}
4Be^{10}	3×10^{-3}	1.9×10^{-5}	Lungs	1.0×10^{-8}
6C^{14}	4×10^{-4}	1.4×10^{-8}	Fat	2.8×10^{-10}
11Na^{22}	1×10^{-4}	1.6×10^{-6}	Lungs	3.4×10^{-9}
11Na^{24}	1.5×10^5	1.3×10^2	GI(LLI)§	9.7
13Al^{26}	1×10^{-1}	3.8×10^{-3}	Lungs	4.6×10^{-6}
20Ca^{45}	1×10^{-1}	7.4×10^{-4}	Bone	1.1×10^{-7}
21Sc^{46}	3×10^{-1}	1.7×10^{-3}	Lungs	9.0×10^{-6}
21Sc^{47}	3×10^{-1}	8.0×10^{-5}	GI(LLI)	1.1×10^{-6}
21Sc^{48}	3.1	2.7×10^{-3}	GI(LLI)	1.6×10^{-4}
25Mn^{54}	6×10^{-4}	1.3×10^{-4}	GI(LLI)	7.2×10^{-5}
26Fe^{55}	1×10^{-1}	1.5×10^{-5}	Spleen	9.3×10^{-9}
30Zn^{65}	1×10^{-1}	2.2×10^{-4}	Lungs	8.2×10^{-7}
40Zr^{89}	5×10^{-1}	1.8×10^{-4}	GI(LLI)	8.9×10^{-6}
40Zr^{95}	1×10^{-1}	4.1×10^{-4}	Lungs	1.2×10^{-6}
$50\text{Sn}^{117\text{m}}$	1×10^{-1}	1.5×10^{-4}	Lungs	4.4×10^{-7}
$50\text{Sn}^{119\text{m}}$	2×10^{-1}	1.7×10^{-5}	Lungs	2.2×10^{-7}

*Total body — energies great enough for epidermis penetration

†Includes skin absorption dose

§GI(LLI) Gastrointestinal tract (lower large intestine)

inventory, previously held in a solid material, was released to the environment. This accident is a very good example of the possible potential hazards which must be evaluated during any safety analysis.

E. DISPOSAL OF RADIOLOGICAL WASTES

The radiological inventory of the first wall and modular blanket has been discussed in Section III-A. It was indicated that this waste would be considered intermediate level waste, based on the definitions of Reference 13. The waste management problem typically depends on the kind of waste that is generated. The radiological wastes generated by the nonreclaimed materials of the first wall and blanket may be compared to the spent fuel of a fission reactor. The spent blanket materials contain a relatively large inventory of radioisotopes, some of which are long lived. The spent fuel contains extremely large quantities of radionuclides and fissionable materials. It is on this point that the distinction between these wastes is made. Wastes containing fissionable materials are severely controlled for direct burial. However, solid waste materials which contain no fissionable materials routinely are directly buried. The spent materials of the fusion reactor blanket thus present a different waste disposal problem from that of the spent fuel of a fission reactor.

Previous calculations indicated that the total nonreclaimed blanket waste inventory occupies $\sim 2000 \text{ m}^3$. The decay of this radiological waste inventory is shown in Figure 12. Following a 1-year on-site holding period, the waste could be shipped to a disposal facility, in a single lot, as Transport Group VI (10 CFR 71). However, division of the total inventory into smaller groups, with the groups having separate shipments, would seem more plausible. This results in the creation of $\sim 2000 \text{ m}^3$ of waste every 3 years for a 5000-MWt fusion plant. Direct burial may not be feasible for these large volumes. Based on preliminary investigations by Oak Ridge National Laboratory (Reference 13), hydraulic fracturing may be a viable method of disposal of fusion blanket waste material. The method consists of mixing the aqueous wastes with preblended dry solids containing principally cements, and then pumping the resultant slurry down a well and out into a conformable, nearly horizontal, fracture in a thick shale formation at the desired depth. After the pumping phase is completed, the cement

slurry is allowed to harden under pressure, thereby forming a thin, horizontal, grout sheet. This procedure can be repeated successively up the well, creating a stack of horizontal grout sheets. Hydraulic fracturing presents a method of deep underground storage for nonfissionable materials.

When considering disposal of the waste generated in the fusion blanket, it is important to consider the chemistry and decay modes of the isotopes created. Referring to Table 4, the only isotope of relatively large activity with a half-life >10 years is aluminum-26. The principal concern with direct burial is leeching of the radionuclides into the ground water system. The philosophy for direct burial of iodine-129 can be applied here. Leeching is minimized by large concentrations of the naturally occurring element, which exists in the surface crust of the earth. The high natural aluminum concentration will minimize leeching of radioactive aluminum-26 into underground water, just as the natural iodine concentration limits iodine-129 leeching. Hence, since aluminum-26 is the principal high-level, long-lived isotope present in the fusion blanket materials, direct burial is enhanced by the presence of large quantities of natural aluminum.

A quantitative discussion of the effects of this dilution proves interesting. The aluminum-26 activity per unit mass of aluminum for nonreclaimed waste from the MAB-CTR is $\sim 10^{-6}$ Ci/gm. The equilibrium concentration of natural aluminum in ground water is $\sim 10^{-7}$ gm/ml.⁽¹⁹⁾ Additionally, the earth's crust contains ~ 8 wt % aluminum.⁽²⁰⁾ The maximum permissible concentration of aluminum-26 in water is 6×10^{-6} μ Ci/ml (continuous exposure). It thus follows that equilibration of the MAB-CTR aluminum waste with ground water would result in a concentration of 10^{-7} μ Ci/ml, which is an order of magnitude below the required $MPC_{w,\infty}$.

IV. MODULE DESIGN ALTERNATIVES

A. DIRECT TRITIUM RELEASE MODULE

Although the minimum activity blanket system considered so far in this report reduces the tritium inventory in the blanket and in the helium coolant circuit, modifications are possible which would provide tritium release directly into the torus vacuum, without contaminating the primary coolant circuit. The modifications are based on the fact that tritium is bred in a solid alloy, LiAl, and additionally because the vapor pressure of tritium in this alloy is relatively high. Two possible designs for this type of system are illustrated in Figure 17.⁽²¹⁾ In each design, tritium breeding occurs outside the module shell. In Design A, the bred tritium is released directly to the vacuum region of the torus. Lithium-aluminum alloy pellets are contained within a cylindrical shell, with a porous outer layer of SAP. In Design B, the tritium is released to a secondary low-pressure channel, from which it is carried to a separate processing system. A corrugated aluminum jacket and outer SAP layer form the spaces for locating the lithium-aluminum pellets and the low-pressure channel. Note that Design B requires additional low-pressure piping connections to each module.

These designs offer several advantages. Internal insulation requirements for the module are minimized by elimination of the lithium-aluminum alloy from inside the module structure. The isotopic fraction of lithium-6 could be reduced to ~30%. The reference minimum activity blanket design of this report utilizes an isotopic fraction of 90% Li^6 . This reduction results from the availability of a faster spectrum for the higher-threshold-energy $\text{Li}^7 (n, n'\alpha) \text{H}^3$ reactions. The minimization of isotopic enrichment is again a great economic factor in favor of this type of tritium breeding situation. There are some disadvantages in the modified design. Fabrication of the module shell with the aluminum corrugation and the SAP containment shell for the tritium breeding medium presents difficult problems. This design also reduces the total equivalent thermal output per fusion reaction. This results from loss of the available heat generation from $\text{Li}^6 (n, \alpha) \text{H}^3$ reactions. Reduction in isotopic concentration of lithium will lower the relative fraction of these reactions which occur for breeding. The breeding

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Figure 17. Blanket Modules with Solid Lithium Compounds Outside the Module Shell (From Reference 21)

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reaction in lithium-7 is endothermic, as opposed to the exothermic lithium-6 reaction. Thus, reduction in the isotopic enrichment of lithium-6 lowers the fraction of exothermic breeding reactions, which then lowers the total heat production.

The direct tritium release design does, however, present a viable, alternate means of eliminating the tritium inventory in the primary coolant system of a fusion reactor. Utilization of the minimum activity blanket caused a great reduction in this inventory, as compared to the original UWMAK-I design. It was shown, in Section III-D, that the total radiological dose at the site boundary, resultant from release of the entire helium circuit of one sector of the reactor, is <5 rem. However, removal of the possibility of a radiological dose from a loss of the total coolant system would seem very practical, in light of increasingly stricter regulatory requirements.

B. DOUBLY-CONTAINED BLANKET MODULES

The discussions of accidents in Section III-D indicate that leakage of primary coolant through the first wall could result in massive dumping of the internal energy of the plasma back toward the first wall. It is thus very practical to consider modular blanket systems which would minimize the possibility of undetected leakage, and would allow such an occurrence to be considered an extremely unlikely operating event. A design modification of the minimum activity blanket module results in just such a system. An illustration of this design is shown in Figure 18.⁽²¹⁾ A second structural SAP shell has been added to the module design. The annular region between the inner and outer module shells contains water-filled SAP cooling tubes. The internal design of the module has been modified to allow entering helium coolant to traverse through an insulated SAP pipe down the module center. The helium coolant flow is then guided through a SAP distribution plate, and travels through the lithium-aluminum pellets, beryllium pellets, and a graphite moderator. The coolant exits through the top of the module. The internals of the module are insulated from the inner structural shell by an aluminum oxide thermal barrier. Heat generation in the first wall, outer cylindrical shell, and inner structural shell is removed via the water coolant of the inner-outer shell annulus, and the remainder of the thermal generation occurs in the module interior and is deposited in the helium coolant.

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Figure 18. Double-Containment Module — Helium and Water Coolants (From Reference 21)

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Much redundancy can be provided through this module design. For the design of Figure 18, three separate sets of water-filled coolant tubes are provided in the inner-outer shell annulus. Any single set of these tubes could provide full thermal removal capability for an individual module. There is thus triple protection against loss of a tube or flow blockage in a set of tubes. The helium coolant sees the double barrier of the inner and outer SAP structural shells. The annulus containing the coolant tubes is evacuated during the operation of the reactor. Each annulus is "sniffed" by a thermocouple vacuum gauge connected to the module. A leakage of helium through the inner structural shell, or a leakage of water through one set of the coolant tubes, would be detected by an increase in the pressure of a given module. This system design thus provides protection against single leakage failures, and additionally provides the provision for detection of such first-level failures. During normal operation of the plant, several such first-level failures could be allowed before shutdown of the plant and replacement of the failed modules were required. From a safety standpoint, therefore, the doubly-contained module design presents an extremely practical system. However, the design does present certain difficulties in the thermal transport systems and power conversion systems. Approximately 20 to 30% of the thermal power is deposited in the water coolant at low temperature. The remaining 70 to 80% is deposited in the helium coolant. Although elevated helium outlet temperatures are possible with this design, utilization of the low-temperature portion of the thermal power deposited in the water coolant presents problems in efficiency. This power could be used for reheaters or for feedwater heaters. A detailed trade study of the various possible combinations of coupling the helium coolant system, water coolant system, and power generation system is required. However, based on current design practices for liquid metal fast breeder reactors, and additionally based on possible future controls of release of radioactivity effluents to the environment, it is Atomics International's position that maximization of safety will negate the detrimental effects of possible reductions in thermal conversion efficiency. The doubly-contained module design provides plausible incorporation of inherent safety capability to the minimum activity module design.

V. CONCLUSIONS

The minimum activity blanket concept provides several desirable features for the design of fusion reactor blankets. These features combine to reduce the radiological hazards associated with the operation of thermonuclear plants, and additionally permit desirable maintenance features to be included in realizable blanket designs.

The minimum activity system uses ultra-purity aluminum in SAP as a structural material for the modular blanket. This results in approximately a six order of magnitude reduction of the total generated radioactive inventory (half-life >1 day), as compared with the stainless steel blanket of the original UWMAK-I design. Minimum activity blanket modules are cooled with helium at moderate pressures. Tritium breeding occurs in solid lithium-aluminum alloy. State-of-the-art processing techniques maintain a 34-gm tritium inventory in the blanket plus coolant. This is a two to three order of magnitude inventory reduction, as compared to the UWMAK-I lithium cooled blanket.

Reductions in the radiological inventory reduce the hazards from possible plant accidents. Preliminary investigations of several tritium release events indicate that doses at an assumed 600-m site boundary are much less than 25 rem. Further investigations of an extremely unlikely blanket vaporization accident conclude that the site boundary dose from this event is again much less than 25 rem. Inclusion of containment holdup times would greatly reduce the delivered dosage for all accidents.

The reduced radionuclide inventories allow direct burial of nonreclaimed reactor wastes. Preliminary analysis indicates that $\sim 2000 \text{ m}^3$ of intermediate level waste is generated during the 3-year operation of a 5000 MWt plant. The only long-lived nuclide with significant activity is aluminum-26. Equilibration of this isotope with naturally occurring aluminum in the earth's crust limits the volumetric activity from water leeching to well below the MPC_w for Al^{26} .

The modular design of the minimum activity system permits direct removal and replacement of $\sim 50\%$ of the first wall area. Further improvements in maintenance could be realized through modifications of the magnets. Design options for the minimum activity modules include a zero tritium inventory primary coolant circuit, and a double-contained module with leak detection capability.

APPENDIX A
CALCULATION OF CONTINUOUS, NON-OCCUPATIONAL, MAXIMUM
PERMISSIBLE CONCENTRATIONS IN WATER (MPC_w) FOR
 ${}^4\text{Be}^{10}$, ${}^{13}\text{Al}^{26}$, ${}^{40}\text{Zr}^{89}$, ${}^{50}\text{Sn}^{117\text{m}}$, AND ${}^{50}\text{Sn}^{119\text{m}}$

Based on the methods of Reference 10, the following continuous-exposure, non-occupational MPC_w have been calculated.

<u>Radionuclide</u>	<u>MPC_{w,∞} (μCi/ml)</u>	<u>Critical Organ</u>	<u>Soluble (S) or Insoluble (I)</u>
${}^4\text{Be}^{10}$	7.2×10^{-5}	GI (LLI)	S or I
${}^{13}\text{Al}^{26}$	5.7×10^{-6}	Total Body	S
${}^{40}\text{Zr}^{89}$	6.6×10^{-5}	GI (LLI)	S or I
${}^{50}\text{Sn}^{117\text{m}}$	4.1×10^{-4}	GI (LLI)	I
${}^{50}\text{Sn}^{119\text{m}}$	3.7×10^{-4}	GI (LLI)	I

The following equations were used in the calculations for radionuclides with a stable first daughter.⁽¹⁰⁾

For organs other than the gastrointestinal (GI) tract

$$\text{MPC}_w = \frac{(9.2 \times 10^{-4})(2.8 \times 10^{-3})\text{mR}}{2.92\xi\tau_e f_w [1 - \exp(-0.693t/\tau_e)]} \quad \dots (\text{A-1})$$

where

m = mass of organ (gm)

R = allowed dosage for organ (rem/week) (see following discussion)

ξ = effective energy absorbed per disintegration (MeV) (see following discussion)

τ_e = effective half-life of radionuclide (days) (see following discussion)

f_w = fraction of that taken into body by ingestion that is retained in the critical organ

t = period of exposure (days) — for continuous MPC's, $t = 50$ years, and the exponential term is effectively zero.

For the stomach only

Multiply Equation A-1 by a factor of 2 (radiation delivered from one side only), and use

$\tau_e = \tau_r$ = radiological half-life (days)

t = residence time in stomach (1/24 day)

For the remainder of the GI tract [small intestine (SI), upper large intestine (ULI), lower large intestine (LLI)]

$$MPC_w = \frac{(7.4 \times 10^{-6})mR}{(2.92)\xi\tau \exp(-\lambda_o t)} \quad \dots (A-2)$$

where

m = mass of organ (gm)

R = allowed dosage for organ (rem/week) (see following discussion)

ξ = effective energy absorbed per disintegration (MeV) (see following discussion)

τ = residence time in section of GI tract (days)

λ_o = effective decay constant (days⁻¹) (see following discussion)

t = time of entry to section of GI tract (days)

Non-occupational weekly dosage to various organs, R

<u>Organ</u>	<u>Dose (rem/week)</u>
Whole body	0.01
Bone	0.06
Thyroid	0.6
All other	0.03

Determination of effective energy, ξ

$$\xi = \sum EF(RBE)n \quad \dots (A-3)$$

where

E = energy deposited per disintegration (MeV) (see following discussion)

F = daughter/parent disintegration ratio

= 1 for parent with stable daughter

RBE = relative biological effectiveness

= 1.0 for γ , β^- , β^+ , e^-

= 1.7 for β^- , β^+ , e^- if $E_{\max} \leq 0.03$ MeV

= 10 for α

= 20 for recoil atoms

n = relative damage factor for bone

= 1.0 for all decay in organs other than bone

= 1.0 in bone for γ

= 1.0 in bone for all decay of members of radium chain

= 5.0 in bone for α , β^- , β^+ , and e^-

For β^- , E is given by

$$E = 0.33E_{\max} f \left(1 - \frac{\sqrt{Z}}{50} \right) \left(1 + \frac{\sqrt{E_{\max}}}{4} \right) \quad \dots (A-4)$$

where

$E_{\max}$ = maximum β^- energy (MeV)

f = fraction of disintegrations occurring

Z = atomic number of the β^- emitter

For β^+ , E is given by

$$E = 0.33E_{\max} f \left(1 + \frac{\sqrt{E_{\max}}}{4} \right) + 2(0.51)f \left[1 - \exp(-\sigma\chi) \right] \quad \dots (A-5)$$

The second term arises from the two 0.51 MeV gammas generated in the annihilation.

σ = total coefficient of absorption minus Compton Scattering Coefficient at the energy of the photon (cm^{-1})

χ = effective radius of the organ (cm)

Note: In bone, $n = 5$ applies only to β^+ energy term.

For γ , E is given by

$$E = E_{\gamma} f \left[1 - \exp(-\sigma\chi) \right] \quad \dots (A-6)$$

where

E_{γ} = energy of γ (MeV)

For α , E is given by

$$E = E_{\alpha} f \quad \dots (A-7)$$

where

E_{α} = energy of α (MeV)

For electron capture (EC), E is given by (K&L shells)

$$E = \eta f \left[1 - \exp(-\sigma\chi) \right] \quad \dots (A-8)$$

where

η = binding energy of daughter

Determination of effective half-life, τ_e

For radionuclides experiencing biological and radiological decay:

$$\tau_e = (\tau_r \cdot \tau_b) / (\tau_r + \tau_b) \quad \dots (A-9)$$

where

τ_r = radiological half-life (days)

τ_b = biological half-life (days)

Determination of effective decay constant in GI tract, λ_o

Soluble materials are absorbed into the GI tract only in the small intestine.
For soluble materials,

f_1 = fraction of ingested radionuclide absorbed

λ_o applies to the radiological decay and biological absorption of a nuclide as it travels through the GI tract

$\exp(-\lambda_o t)$ = fraction of the radionuclide remaining in the GI tract at any time, t , after ingestion at time zero

Thus, in the various organs, we have

<u>Organ</u>	<u>Soluble Nuclide</u>	<u>Insoluble Nuclide</u>
Stomach	$\lambda_o = \lambda_r$	$\lambda_o = \lambda_r$
GI (SI)	$\lambda_o = \lambda_r + \lambda_b$	$\lambda_o = \lambda_r$
GI (ULI)	$\lambda_o = \lambda_r$	$\lambda_o = \lambda_r$
GI (LLI)	$\lambda_o = \lambda_r$	$\lambda_o = \lambda_r$

λ_o is obtained by using

$$(1 - f_1) = \exp(-\lambda_o \tau_{SI}) \quad \dots (A-10)$$

where

τ_{SI} = residence time in SI (0.167 days)

and applying the proper λ_o equation in the proper GI section.

Alternately, the fraction $(1 - f_1)$ of material remaining in the GI tract after passing through the small intestine can be employed.

For calculations of MPC in the GI tract, multiply Equation A-2 by the factor below, for the organ of interest:

<u>Organ</u>	<u>Soluble Nuclide</u>	<u>Insoluble Nuclide</u>
GI (SI)	1	1
GI (ULI)	$(1 - f_1)^{-1}$	1
GI (LLI)	$(1 - f_1)^{-1}$	1

The preceding equations were used in the following calculations to determine the $MPC_{w,\infty}$ values for the required nuclides. Data for the calculations was obtained from ICRP-2⁽¹⁰⁾ and the Isotopes Handbook.⁽²¹⁾

$$f_1 = 2 \times 10^{-3}$$

$$\tau_r = 5.84 \times 10^8 \text{ days}$$

Decays

β^- (0.56 MeV) 100%

Soluble Nuclide Data

<u>Organ</u>	<u>τ_b (days)</u>	<u>f_w</u>	<u>m (gm)</u>	<u>r (cm)</u>
Total Body (TB)	180	2×10^{-3}	70,000	30
Kidney	120	6×10^{-5}	300	7
Liver	270	2×10^{-4}	1,700	10
Spleen	540	4×10^{-6}	150	7
Bone	450	6.4×10^{-4}	7,000	5

Effective Energy

<u>Organ</u>	ξ <u>(MeV)</u>
TB	0.211
Kidney	0.211
Liver	0.211
Spleen	0.211
Bone	1.06
GI (LLI)	0.211

<u>MPC_{w,∞}</u>		
<u>Organ</u>	<u>Soluble or Insoluble</u>	<u>MPC_{w,∞}</u> <u>(μCi/ml)</u>
TB	S	8.1×10^{-3}
Kidney	S	5.2×10^{-3}
Liver	S	3.9×10^{-3}
Spleen	S	8.7×10^{-3}
Bone	S	3.5×10^{-3}
GI (LLI)	S	7.2×10^{-5}
GI (LLI)	I	7.2×10^{-5} } minimum

$$f_1 = 0.1$$

$$\tau_r = 2.67 \times 10^8 \text{ day}$$

Decays

β^+	(1.17 MeV)	85%
γ	(1.12 MeV)	4%
γ	(1.81 MeV)	100%
γ	(2.96 MeV)	0.3%
EC	($\eta = 0.0013$)	11%

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Soluble Nuclide Data

<u>Organ</u>	τ_b (days)	f_w	m (gm)	r (cm)
TB	550	0.1	70,000	30
Spleen	500	2×10^{-4}	150	7

Effective Energy

<u>Organ</u>	ξ (MeV)
TB	1.96
Spleen	0.897
GI (LI)	0.77

<u>Organ</u>	<u>Soluble or Insoluble</u>	$\frac{MPC_{w,\infty}}{(\mu Ci/ml)}$
TB	S	5.7×10^{-6}
Spleen	S	4.4×10^{-5}
GI (LLI)	S	2.19×10^{-5}
GI (LLI)	I	1.97×10^{-5} minimum

$$f_1 = 1 \times 10^{-4}$$

$$\tau_r = 3.3 \text{ days}$$

Decays

β^+	(0.90 MeV)	22%
γ	(0.91 MeV)	100%
γ	(1.71 MeV)	1%
γ	(1.62 MeV)	0.12%
EC	($\eta = 0.017$)	77%

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Soluble Nuclide Data

<u>Organ</u>	τ_b (days)	f_w	m (gm)	r (cm)
TB	450	1×10^{-4}	70,000	30
Bone	1000	3.6×10^{-5}	7,000	5
Kidney	900	2×10^{-6}	300	7
Spleen	900	6×10^{-7}	150	7
Liver	320	7×10^{-6}	1,700	10

Effective Energy

<u>Organ</u>	ξ (MeV)
TB	0.784
Bone	0.701
Kidney	0.316
Spleen	0.316
Liver	0.398
GI (LI)	0.257

<u>Organ</u>	<u>Soluble or Insoluble</u>	$MPC_{w,\infty}$ ($\mu\text{Ci/ml}$)
TB	S	2.40
Bone	S	4.46
Kidney	S	3.82
Spleen	S	6.36
Liver	S	4.90
GI (LLI)	S	6.6×10^{-5}
GI (LLI)	I	6.6×10^{-5}

} minimum

$$f_1 = 0.05$$

$$\tau_r = 14 \text{ days}$$

Decays

γ	(0.15 MeV)	100%
γ	(0.16 MeV)	100%

Soluble Nuclide Data

<u>Organ</u>	τ_b (days)	f_w	m (gm)	r (cm)
TB	35	0.05	70,000	30
Bone	100	0.02	7,000	5
Thyroid	70	5×10^{-6}	20	3
Liver	70	5×10^{-4}	1,700	10
Prostate	35	8×10^{-4}	40	3

Effective Energy

<u>Organ</u>	ξ (MeV)
TB	0.169
Bone	0.242
Thyroid	0.023
Liver	0.072
Prostate	0.023
GI (LI)	0.038

<u>Organ</u>	<u>Soluble or Insoluble</u>	$\text{MPC}_{w,\infty}$
		$\text{MPC}_{w,\infty}$ ($\mu\text{Ci/ml}$)
TB	S	7.3×10^{-3}
Bone	S	6.3×10^{-3}
Thyroid	S	7.87
Liver	S	0.11
Prostate	S	5.7×10^{-3}
GI (ULI)	S	8.6×10^{-4}
GI (ULI)	I	8.17×10^{-4}
GI (LLI)	S	4.32×10^{-4}
GI (LLI)	I	4.10×10^{-4} minimum

$^{119m}_{50}\text{Sn}$

$$f_1 = 0.05$$

$$\tau_r = 245 \text{ days}$$

Decays

γ	(0.06 MeV)	100%
γ	(0.02 MeV)	100%

Soluble Nuclide Data

<u>Organ</u>	τ_b (days)	f_w	m (gm)	r (cm)
TB	35	0.05	70,000	30
Bone	100	0.02	7,000	5
Thyroid	70	5×10^{-6}	20	3
Liver	70	5×10^{-4}	1,700	10
Prostate	35	8×10^{-4}	40	3

Effective Energy

<u>Organ</u>	ξ <u>(MeV)</u>
TB	0.072
Bone	0.079
Thyroid	0.033
Liver	0.051
Prostate	0.033
GI (LI)	0.041

MPC_{w,∞}

<u>Organ</u>	<u>Soluble or Insoluble</u>	MPC _{w,∞} <u>(μCi/ml)</u>
TB	S	5.6×10^{-3}
Bone	S	3.3×10^{-3}
Thyroid	S	1.18
Liver	S	3.2×10^{-2}
Prostate	S	1.31×10^{-3}
GI (LLI)	S	3.89×10^{-4}
GI (LLI)	I	3.70×10^{-4} minimum

APPENDIX B

F FACTORS FOR INHALATION OF RADIONUCLIDES

The dose delivered to any organ per curie inhaled (F factors) can be calculated, based on the equations of Reference 10.

An equation was developed for such a calculation for organs except the gastrointestinal tract by D. F. Bunch.⁽¹⁸⁾ Equations were also developed here for the GI tract.

Most of the nomenclature used here is defined and discussed in Appendix A. Any newly introduced factors are discussed in this appendix.

For all organs except the GI tract

$$D = \frac{f_a \xi \tau_e}{m} \left[1 - \exp\left(-\frac{(50)(365)(0.693)}{\tau_e}\right) \right] \left(\frac{1.6 \times 10^{-6} \times 3.2 \times 10^{15}}{0.693 \times 10^2} \right) \dots (B-1)$$

where

D = dose (rem/curie inhaled)

f_a = fraction inhaled to organ of interest

1.6×10^{-6} erg/MeV

3.2×10^{15} disintegrations/day per curie

10^2 erg/gm per rad

(50 x 365) days in 50 years

For the stomach

Multiply Equation B-1 by 0.5 and use

$\tau_e = \tau_r$ = radiological half-life

t = residence time in stomach (1/24 day)

For the GI tract (SI, ULI, LLI)

$$D = \frac{f_a \xi \tau}{m} \exp(-\lambda_o t) \left(\frac{1.6 \times 10^{-6} \times 3.2 \times 10^{15}}{2 \times 10^2} \right) \dots (B-2)$$

TABLE B-1
F FACTORS FOR INHALATION

Radionuclide	F factor (rem/Ci inhaled) T_o	Critical Organ	Soluble or Insoluble Radionuclide
$1H^3$	$2.06 \times 10^{2*}$	Body water	S
$4Be^{10}$	2.34×10^5	Lungs	I
$6C^{14}$	1.82×10^3	Fat	S
$11Na^{22}$	7.91×10^5	Lungs	I
$11Na^{24}$	4.39×10^4	GI (LLI)	I
$13Al^{26}$	1.89×10^6	Lungs	S
$20Ca^{45}$	3.68×10^5	Bone	S
$21Sc^{46}$	2.92×10^5	Lungs	I
$21Sc^{47}$	1.36×10^4	GI (LLI)	I
$21Sc^{48}$	4.48×10^4	GI (LLI)	I
$25Mn^{54}$	1.04×10^4	GI (LLI)	I
$26Fe^{55}$	7.46×10^3	Spleen	S
$30Zn^{65}$	1.11×10^5	Lungs	I
$40Zr^{89}$	1.84×10^4	GI (LLI)	I
$40Zr^{95}$	2.03×10^5	Lungs	I
$50Sn^{117m}$	8.31×10^3	Lungs	I
$50Sn^{119m}$	3.79×10^4	Lungs	I

*An additional equal dose is received through skin absorption.

Factor of 2 in denominator for radiation impinging from one side only.

See Appendix A for determination of τ , λ_o , and t .

The critical organ for an inhaled dose was determined by: (1) examining the MPC in air for the minimum value (for radionuclides listed in Reference 10, or (2) calculating the MPC's in air for all selected organs (soluble and insoluble materials) and determining which organ was critical, based on allowed dosage.

The dose per curie inhaled will be calculated first for radionuclides listed in Reference 10.

Critical Organ: Body water (soluble - HTO or T₂O)

$$f_a = 1.0$$

$$\xi = 0.01 \text{ MeV}$$

$$\tau_e = 12 \text{ days}$$

$$m = 4.3 \times 10^4 \text{ gm}$$

$$F = 2.06 \times 10^2 \text{ rem/Ci inhaled}$$

Critical Organ: Fat (soluble - CO₂)

$$f_a = 0.38$$

$$\xi = 0.054 \text{ MeV}$$

$$\tau_e = 12 \text{ days}$$

$$m = 10^4 \text{ gm}$$

$$F = 1.82 \times 10^3 \text{ rem/Ci inhaled}$$

Critical Organ: Lung (insoluble)

$$f_a = 0.125$$

$$\xi = 0.80 \text{ MeV}$$

$$\tau_e = (120 \times 949)/(120 + 949) = 107 \text{ days}$$

$$m = 10^3 \text{ gm}$$

$$F = 7.91 \times 10^5 \text{ rem/Ci inhaled}$$

Critical Organ: GI (LLI) (insoluble)

$$f_a = 0.625$$

$$\xi = 1.0 \text{ MeV}$$

$$\tau = 18 \text{ hr} = 0.75 \text{ days}$$

$$m = 150 \text{ gm}$$

$$t = 13 \text{ hr} = 0.54 \text{ days}$$

$$\lambda_o = 1.11 \text{ day}^{-1}$$

$$F = 4.39 \times 10^4 \text{ rem/Ci inhaled}$$

Critical Organ: Bone (soluble)

$$f_a = 0.50$$

$$\xi = 0.43 \text{ MeV}$$

$$\tau_e = 162 \text{ days}$$

$$m = 7 \times 10^3 \text{ gm}$$

$$F = 3.68 \times 10^5 \text{ rem/Ci inhaled}$$

Critical Organ: Lung (insoluble)

$$f_a = 0.125$$

$$\xi = 0.64 \text{ MeV}$$

$$\tau_e = (120 \times 84)/(120 + 84) = 49.4 \text{ days}$$

$$m = 10^3 \text{ gm}$$

$$F = 2.92 \times 10^5 \text{ rem/Ci inhaled}$$

Critical Organ: GI (LLI) (insoluble)

$$f_a = 0.625$$

$$\xi = 0.19 \text{ MeV}$$

$$\tau = 0.75 \text{ days}$$

$$m = 150 \text{ gm}$$

$$t = 0.54 \text{ days}$$

$$\lambda_o = 0.202 \text{ day}^{-1}$$

$$F = 1.36 \times 10^4 \text{ rem/Ci inhaled}$$

Critical Organ: GI (LLI) (insoluble)

$$f_a = 0.625$$

$$\xi = 0.69 \text{ MeV}$$

$$\tau = 0.75 \text{ days}$$

$$m = 150 \text{ gm}$$

$$t = 0.54 \text{ days}$$

$$\lambda_o = 0.385 \text{ day}^{-1}$$

$$F = 4.48 \times 10^4 \text{ rem/Ci inhaled}$$

Critical Organ: GI (LLI) (insoluble)

$$f_a = 0.625$$

$$\xi = 0.13 \text{ MeV}$$

$$\tau_e = 0.75 \text{ days}$$

$$m = 150 \text{ gm}$$

$$t = 0.542 \text{ days}$$

$$\lambda_o = 2.29 \times 10^{-3} \text{ days}^{-1}$$

$$F = 1.04 \times 10^4 \text{ rem/Ci inhaled}$$

Critical Organ: Spleen (soluble)

$$f_a = 6 \times 10^{-3}$$

$$\xi = 0.0065 \text{ MeV}$$

$$\tau_e = 388 \text{ days}$$

$$m = 150 \text{ gm}$$

$$F = 7.46 \times 10^3 \text{ rem/Ci inhaled}$$

Critical Organ: Lungs (insoluble)

$$f_a = 0.125$$

$$\xi = 0.15 \text{ MeV}$$

$$\tau_e = (120 \times 244)/(120 + 244) = 80.4 \text{ days}$$

$$m = 10^3 \text{ gm}$$

$$F = 1.11 \times 10^5 \text{ rem/Ci inhaled}$$

Critical Organ: Lungs (insoluble)

$$f_a = 0.125$$

$$\xi = 0.52 \text{ MeV}$$

$$\tau_e = (120 \times 65)/(120 + 65) = 42.2 \text{ days}$$

$$m = 10^3 \text{ gm}$$

$$F = 2.03 \times 10^5 \text{ rem/Ci inhaled}$$

We now calculate F factors for those radionuclides not listed in Reference 10.

$$f_1 = 2 \times 10^{-3}$$

$$\tau_r = 5.84 \times 10^8 \text{ days}$$

Decays

β^- (0.56 MeV) 100%

<u>Organ</u>	<u>τ_b (days)</u>	<u>f_a</u>	<u>m (gm)</u>	<u>r (cm)</u>
Total Body (TB)	180	0.25	7×10^4	30
Kidney	120	7.5×10^{-3}	3×10^2	7
Liver	270	2.5×10^{-2}	1.7×10^3	10
Spleen	540	5×10^{-4}	1.5×10^2	7
Bone	450	8×10^{-2}	7×10^3	5
Lung (I)	120	0.125	1.0×10^3	10
GI (LLI) (S)	0.75*	0.50	1.5×10^2	5
GI (LLI) (I)	0.75*	0.625	1.5×10^2	5

* τ for GI

<u>Organ</u>	<u>ξ (MeV)</u>
TB	0.211
Kidney	0.211
Liver	0.211
Spleen	0.211
Bone	1.06
Lung	0.211
GI (Large intestine) (LI)	0.211

<u>Organ</u>	<u>F Factor (rem/Ci inhaled)</u>
TB	1.0×10^4
Kidney	4.68×10^4
Liver	6.19×10^4
Spleen	2.81×10^4
Bone	4.03×10^5
Lung (I)	2.34×10^5 [Critical organ (allowed bone dosage higher)]
GI (LLI) (S)	1.35×10^4
GI (LLI) (I)	1.69×10^4

$$f_1 = 0.10$$

$$\tau_r = 2.67 \times 10^8 \text{ day}$$

<u>Decays</u>		
β^+	(1.17 MeV)	85%
γ	(1.12)	4%
γ	(1.81)	100%
γ	(2.96)	0.3%
EC	($\eta = 0.0013$)	11%

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<u>Organ</u>	<u>τ_b</u> (days)	<u>f_a</u>	<u>m</u> (gm)	<u>r</u> (cm)
TB	550	0.3	7×10^4	30
Spleen	500	6×10^{-4}	1.5×10^2	7
Lungs (S)	600	4×10^{-2}	1×10^3	10
Lungs (I)	120	0.125	1×10^3	10
GI (LLI) (S)	0.75^*	0.50	1.5×10^2	5
GI (LLI) (I)	0.75^*	0.625	1.5×10^2	5

* τ for GI

<u>Organ</u>	<u>ξ</u> (MeV)
TB	1.96
Spleen	0.897
Lung	1.06
GI (LI)	0.77

<u>Organ</u>	<u>F Factor</u> (rem/Ci inhaled)
TB	3.41×10^5
Spleen	1.32×10^5
Lung (S)	1.89×10^6 (Critical organ)
Lung (I)	1.18×10^6
GI (LLI) (S)	4.44×10^4
GI (LLI) (I)	6.16×10^4

$$f_1 = 1 \times 10^{-4}$$

$$\tau_r = 3.3 \text{ days}$$

Decays

β^+	(0.90 MeV)	22%
γ	(0.91)	100%
γ	(1.71)	1%
γ	(1.62)	0.12%
EC	($\eta = 0.017$)	77%

<u>Organ</u>	τ_b (days)	f_a	m (gm)	r (cm)
TB	450	0.25	7×10^4	30
Bone	1000	9×10^{-2}	7×10^3	5
Kidney	900	5×10^{-3}	3×10^2	7
Spleen	900	1.5×10^{-3}	1.5×10^2	7
Liver	320	2×10^{-2}	1.7×10^3	10
Lung (I)	120	0.125	1×10^3	10
GI (LLI) (S)	0.75*	0.50	1.5×10^2	5
GI (LLI) (I)	0.75*	0.625	1.5×10^2	5

* τ for GI

<u>Organ</u>	ξ (MeV)
TB	0.784
Bone	0.701
Kidney	0.316
Spleen	0.316
Liver	0.398
Lung	0.398
GI (LI)	0.257

<u>Organ</u>	<u>F Factor</u> <u>(rem/Ci inhaled)</u>
TB	6.77×10^2
Bone	2.19×10^3
Kidney	1.28×10^3
Spleen	7.68×10^2
Liver	1.13×10^3
Lung (I)	1.15×10^4
GI (LLI) (S)	1.46×10^4
GI (LLI) (I)	1.84×10^4 (Critical organ)

$f_1 = 0.05$

$\tau_r = 14 \text{ days}$

Decays

γ	(0.15)	100%
γ	(0.16)	100%

<u>Organ</u>	<u>τ_b</u> <u>(days)</u>	<u>f_a</u>	<u>m</u> <u>(gm)</u>	<u>r</u> <u>(cm)</u>
TB	35	0.28	7×10^4	30
Bone	100	8×10^{-2}	7×10^3	5
Thyroid	70	2.8×10^{-5}	2×10^1	3
Liver	70	2.8×10^{-3}	1.7×10^3	10
Prostate	35	4.4×10^{-4}	4×10^1	3
Lung (I)	120	0.125	1×10^3	10
GI (LLI) (S)	0.75*	0.50	1.5×10^2	5
GI (LLI) (I)	0.75*	0.625	1.5×10^2	5

* τ for GI

<u>Organ</u>	ξ (MeV)
TB	0.169
Bone	0.242
Thyroid	0.023
Liver	0.072
Prostate	0.023
Lung	0.072
GI (LI)	0.038

<u>Organ</u>	F Factor (rem/Ci inhaled)
TB	5×10^2
Bone	2.50×10^3
Thyroid	2.8×10^1
Liver	1.03×10^2
Prostate	1.87×10^2
Lung (I)	8.31×10^3 (Critical organ)
GI (LLI) (S)	2.25×10^3
GI (LLI) (I)	2.96×10^3

$$f_1 = 0.05$$

$$\tau_r = 245 \text{ days}$$

Decays

γ	(0.06 MeV)	100%
γ	(0.02)	100%

<u>Organ</u>	τ_b (days)	f_a	m (gm)	r (cm)
TB	35	0.28	7×10^4	30
Bone	100	8×10^{-2}	7×10^3	5
Thyroid	70	2.8×10^{-5}	2×10^1	3
Liver	70	2.8×10^{-3}	1.7×10^3	10
Prostate	35	4.4×10^{-4}	4×10^1	3
Lung (I)	120	0.125	1×10^3	10
GI (LLI) (S)	0.75*	0.50	1.5×10^2	5
GI (LLI) (I)	0.75*	0.625	1.5×10^2	5

* τ for GI

<u>Organ</u>	ξ (MeV)
TB	0.072
Bone	0.079
Thyroid	0.033
Liver	0.051
Prostate	0.033
Lung	0.051
GI (LI)	0.041

<u>Organ</u>	F Factor (rem/Ci inhaled)
TB	6.51×10^2
Bone	4.74×10^3
Thyroid	1.86×10^2
Liver	3.38×10^2
Prostate	8.21×10^2
Lung (I)	3.79×10^4 (Critical organ)
GI (LLI) (S)	2.49×10^3
GI (LLI) (I)	3.28×10^3

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