Safety Aspects of Activation Products in a Compact Tokamak Fusion Power Plant

**H. J. Willenberg
W. E. Bickford**

October 1978

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Richland, Washington 99352

ABSTRACT

Neutron activation of materials in a compact tokamak fusion reactor has been investigated. Results of activation product inventory, dose rate, and decay heat calculations in the blanket and injectors are presented for a reactor design with stainless steel structures. Routine transport of activated materials into the plasma and vacuum systems is discussed. Accidental release of radioactive materials as a result of liquid lithium spills is also considered.

EXECUTIVE SUMMARY

The use of 316 stainless steel as a structural material for a deuterium-tritium tokamak fusion reactor is reviewed to determine the implications of activation products on power plant safety. Activation product inventories, dose rates, and decay heats are described for a helium-cooled, stainless steel blanket surrounding a compact tokamak plasma.

The neutron flux was calculated for a one-dimensional, cylindrical blanket model using the ANISN computer code, and activation product inventories at various times were calculated with the GLUCKE activation code. The total activity of a 1700 MWt blanket after five years of full power operation is 2.1×10^9 Ci. The activity rises extremely rapidly after initial startup, reaching 0.7×10^9 Ci after only one day of full power operation. A substantial fraction of this activity (48% of the one-day total and 20% of the five-year total) is due to activation products of manganese, molybdenum, and silicon. In terms of biological hazard potential, cobalt and manganese isotopes make up 93% of the total radioactivity.

Radiation dose rates at the first wall reach 3×10^6 rem/hr during the first days of full power operation. Dose rates are around 1×10^6 rem/hr in the neutral beam injector, and 3×10^5 rem/hr at the surface of the blanket shield. These dose rates will make contact maintenance on the nuclear island of a tokamak power plant impossible for many years after reactor shutdown.

Under the conditions studied, radioactive decay heat would require cooling of the first wall after shutdown. Assuming adiabatic heating, the first wall decay heat of $0.2 \text{ W}\cdot\text{cm}^{-3}$ would cause it to heat up to 750°C within 17 minutes following a loss of flow. Heating rates in the rest of the blanket are considerably lower and active cooling is probably not needed. The total decay heat falls to 0.2% of full power within one day.

Sputtering and blistering of a bare stainless steel first wall will result in activation products being transported to the plasma and impurity control system. With reasonably efficient ($> 90\%$) magnetic divertors, the

maximum activity permanently entrained on collectors is expected to be about 10^5 Ci for the compact tokamak considered here. This activity level should be considered in reactor design and maintenance, particularly for flowing liquid lithium collection schemes which may transport the activity to other locations. Penetration of high Z impurities from the plasma into the vacuum pumping system can be eliminated by effective baffle design and the long mean free paths and high sticking probability expected for sputter disposition. In the case of vacuum vessel failure, the maximum gamma radionuclide inventory in the plasma potentially available for release could result in airborne concentrations near maximum permissible concentrations in the reactor building. This source term can be reduced significantly by highly efficient divertors ($\sim 99\%$), and will likely be reduced further in the event of vessel failure by the initial influx of air with subsequent cooling and deposition of activations products in the vessel.

In the highly unlikely event of a lithium fire near the reactor, some activated materials will become volatile. At the flame temperature of the liquid lithium reaction with air, water, or concrete, the primary constituents of steel have significant vapor pressures, while aluminum, copper, and lead are near or above their boiling points. Chemical attack of lithium and several of its molten compounds on structural materials may enhance the production of volatile activation products in the form of "smoke" particles.

Understanding the mechanisms by which activation products are formed will allow development of measures to prevent or reduce their potential impact. Areas in which additional research is suggested are:

- investigation of radiation exposures to personnel
- analysis of other candidate structural materials
- determining effects of controlling impurities in stainless steel and using other types of steel
- transporting activation products away from the nuclear island.

Results of these studies will contribute to understanding of fusion power plants and facilitate technological development to ensure safe power plant operation.

CONTENTS

ABSTRACT	iii
EXECUTIVE SUMMARY	v
LIST OF FIGURES	ix
LIST OF TABLES	xi
I. INTRODUCTION	1
II. REFERENCE DESIGN	5
REACTOR DESIGN.	5
STRUCTURAL AND COOLANT MATERIALS.	7
III. BLANKET ACTIVATION	9
OUTER BLANKET ACTIVATION	9
Neutron Flux Calculation.	10
Activation Product Calculation.	13
BIOLOGICAL HAZARD POTENTIAL	22
DOSE RATE	24
COMPARISON WITH LARGE REACTOR DESIGN	25
ALTERNATE STRUCTURAL MATERIALS	27
IV. NEUTRAL BEAM INJECTORS	31
V. VACUUM SYSTEM ACTIVITY	37
VACUUM SYSTEM IMPURITIES	37
IMPURITY FLOW MODEL	40
RESULTING VACUUM SYSTEM ACTIVITY.	46
CONCLUSIONS.	50
VI. RELEASE OF ACTIVATION PRODUCTS IN LITHIUM FIRES.	53

LITHIUM PROPERTIES AND REACTIONS.	53
Lithium-Nitrogen Reaction	55
Lithium-Oxygen Reaction	55
Lithium-Carbon Dioxide Reaction	56
Lithium-Concrete Reaction	56
Lithium-Water Reaction	56
Lithium-Hydrogen Reaction	57
Lithium Reactions With Metals and Refractory Materials.	57
ADIABATIC FLAME TEMPERATURES	57
RADIATION EFFECTS.	58
VAPOR PRESSURES OF NEUTRON ACTIVATION PRODUCTS AND THEIR COMPOUNDS	60
LITHIUM ACCIDENT CONDITIONS	64
SUMMARY	65
VII. COMPONENT ACTIVITY AFTER SHUTDOWN	67
DOSE RATES	70
DECAY HEAT	76
VIII. DIRECTIONS FOR FURTHER RESEARCH.	79
REFERENCES.	81
APPENDIX A - REVIEW OF ACTIVATION CODES IN USE FOR FUSION REACTOR DESIGNS.	A-1
BIBLIOGRAPHY	Biblio-1
DISTRIBUTION	Distr-1

FIGURES

1	Toroidal Cross-Section of a Blanket Module	6
2	Blanket Neutronics Model	10
3	Blanket Neutron Flux.	11
4	Blanket Neutron Flux Spectra by Zone	12
5	Total Blanket Activity by Zone	19
6	Blanket Activity During Operation, by Nuclide	21
7	Biological Hazard Potential of Blanket.	23
8	Blanket Activity During Operation with Alternate Structural Materials.	27
9	Injector Streaming Model	33
10	Injector Activation vs. Time	35
11	Vacuum Impurity Flow Diagram	41
12	Plasma Radionuclide Activities vs. Time	47
13	Collector Radionuclide Activities vs. Time	49
14	Blanket Activity After Shutdown Following 5 Years of Operation	68
15	First Wall Activity After Shutdown	69
16	Estimated Contact Dose Rates at the Blanket Shield.	71
17	Estimated Contact Dose Rate at the First Wall	73
18	Estimated Contact Dose Rate at the Neutral Beam Injector after 10 Years Operation	74
19	Estimated Dose Rate at the Divertor.	75
A-1	Activation-Decay Chain for an Isomeric Group.	A-6
A-2	GLUCKE Library Reactions for Activation Ti Through Zn and Zr Through Ru.	A-8

A-3	Neutronics and Photonics Program System	A-9
A-4	Structure of the Program AKTIV	A-10
A-5	Calculational Scheme Used in UWMAK Studies	A-11
A-6	Calculational Procedure Flowchart for Modified ORIGEN System	A-12
A-7	Information Flow Diagram for the JAERI Code ACTIVE.	A-16

TABLES

1	Reactor Parameters	5
2	Material Composition of Nuclear Grade 316 Stainless Steel.	7
3	Activation Reactions in GLUCKE Library.	17
4	Activity Concentrations at Shutdown.	19
5	Contact Dose Rates from Shield and First Wall (rem/hr) at the Time of Shutdown for 1 Day and 5 Year Continuous Operation	24
6	Comparison of Major Nuclide Contributions to the First Wall Activity Estimated by the GLUCKE code, after 10 Years of Operation at 5000 MWt (no impurities), and UWMAK-II Results	26
7	Radioactive Properties of Various Blanket Materials	28
8	Reactor Operating Parameters	42
9	Assumed First Wall Particle Fluxes and Sputtering Yields	43
10	Assumed Blistering Parameters	45
11	Potential Air Volumes Required for MPC.	48
12	Lithium Reactions, Enthalpies, and Free Energies	54
13	Lithium Nitride - Free Energy of Formation	55
14	Selected Elements - Melting Points, Boiling Points, and Vapor Pressures	60
15	Selected Oxides, Nitrides and Carbides - Melting Points, Boiling Points, and Vapor Pressures.	61
16	Decay Heat Production Rate.	77
17	Time Required for Adiabatic Heating to 750°C Following Shutdown.	78
A-1	Summary of Fusion Reactor Design Study Activation/Dose Calculations	A-3

SAFETY ASPECTS OF ACTIVATION PRODUCTS IN A COMPACT TOKAMAK FUSION POWER PLANT

I. INTRODUCTION

Magnetic fusion power plants based on the deuterium-tritium fuel cycle will produce large numbers of high-energy neutrons. Although no radioactive materials are produced directly by the fusion reaction itself, neutrons which do not escape the nuclear island will be absorbed in blanket and structural materials, some producing long-lived radioactive activation products.

The primary safety concern associated with activation products is radiation exposure. Fusion power plants will be designed so that exposure to personnel is very low during normal operation. However, personnel performing maintenance operations will be exposed to a radiation environment caused by activation products. Disposal and/or recycle of activated materials could result in some release to the environment. Under extreme accident conditions, radioactive materials may be released from their normal locations and present radiation hazards to plant personnel or the general public.

An investigation is in progress at Pacific Northwest Laboratory (PNL) of safety issues resulting from the presence of activation products in magnetic fusion power plants. Early identification of potential safety hazards and their causes will provide a basis for reducing the associated risks during the fusion power plant design process.

The risks associated with neutron activation products depend to a large degree on the materials selected for the blanket structures. A number of structural materials have been proposed, including steel,⁽¹⁻⁴⁾ niobium,⁽⁵⁾ molybdenum,⁽⁶⁾ and aluminum.⁽⁷⁾ Use of any of these materials will require a substantial research and development program to characterize their properties in the radiation, thermal and magnetic environment of a fusion blanket, and

to develop and demonstrate fabrication techniques and materials compatibility. Of all the structural materials which have been proposed, stainless steel has by far the most mature nuclear technology associated with it. Use of stainless steel would therefore be an attractive option from the viewpoint of required materials technology development. Aluminum and vanadium activation rates are much lower than those for steel, especially in the period of one week to several years after shutdown.⁽⁸⁾ Aluminum is also inexpensive and easy to fabricate. Both vanadium and aluminum suffer from poor high-temperature properties. Molybdenum and niobium have better high-temperature properties, but also have high activation cross-sections, many of which produce very long-lived radionuclides.

For the reasons stated above, and because experimental fusion devices will likely be constructed of stainless steel, stainless steel was selected as the first structural material to be investigated. Later studies will focus on alternative materials if warranted, but an initial investigation of activation of steel will certainly have applications for near-term experimental facilities.

Following sections of this report describe the reference design and structural materials (Section II), and characterize activation products in several reactor systems. Section III describes the activation found in the outer blanket, and addresses associated biological hazard and dose rates. Results obtained for the reference design are compared with the larger UWMAK-II⁽²⁾ reactor, and with results obtained assuming other structural materials. Activation products found in neutral beam injectors are estimated in Section IV. Section V describes the activity in the vacuum system: sources of impurities and a flow model are established, and calculations are made based on the impurities and flow assumed. Releases of activation products in lithium fires are estimated in Section VI, after supporting descriptions of lithium reactions, adiabatic flame temperatures, radiation effects on lithium reaction mechanisms and vapor pressures of activation products are made. Reactor components remain

radioactive after the power is shut off. Activity and dose rates for times after shutdown are described in Section VII, and the effects of decay heat are addressed. Section VIII presents recommendations for further research.

II. REFERENCE DESIGN

Reflecting the current national magnetic fusion program focus, the reference design selected as the basis for the preliminary assessment of safety aspects of activation products is a compact tokamak reactor operating on a deuterium-tritium fuel cycle, where tritium is bred in liquid lithium of natural isotopic abundance. The structural material is 316 stainless steel, and high-pressure helium is used as the coolant.

REACTOR DESIGN

The reference design closely resembles a recent demonstration power plant design developed at Oak Ridge National Laboratory by Steiner, et al.,⁽³⁾ which is referred to here as the Steiner Demo. The major difference between the current reference design and the Steiner Demo is the choice of coolant; the former uses helium and the latter a commercial salt mixture. In all other aspects this study uses the Steiner Demo as the reference design. The reactor parameters are summarized in Table 1.

TABLE 1. Reactor Parameters

Major Radius	6.0 m
Minor Radius	1.5 m
Plasma Elongation Factor	1.6
Plasma Volume	426 m ³
First Wall Area	474 m ²
Average Electron Density	$1.7 \times 10^{20} \text{ m}^{-3}$
Average Electron Temperature	14.1 keV
Average Ion Temperature	13.1 keV
Toroidal Beta	0.15
Impurity Level, Z_{eff}	1.07
Neutron Wall Loading	$3.0 \text{ MW} \cdot \text{m}^{-2}$
Average Thermal Power	1700 MW

The contour blanket option of the Steiner Demo report⁽³⁾ was selected for this investigation. This blanket consists of 648 wedge-shaped modules; a toroidal cross-section of a single module is shown in Figure 1. Each module has an operating life of five years. Helium coolant enters the blanket at 400°C and exits at 500°C. The liquid lithium circulates slowly through the blanket, and bred tritium is extracted externally.

Plasma heating is accomplished by neutral beam injection which is considered here to be perpendicular to the plasma toroidal axis. Bundle divertors are utilized for plasma impurity control. It is assumed for now that the reactor building is filled with dry air at atmospheric pressure.

FIGURE 1. Toroidal Cross-Section of a Blanket Module
(from Reference 3)

STRUCTURAL AND COOLANT MATERIALS

Stainless steel seems to be the most likely structural material for early fusion reactors. Besides the iron, chromium, and nickel of which pure stainless steel is composed, actual reactor-grade steels contain a number of impurities. Since the major objective of this study is to identify the key activation products, the most significant impurities in reactor grade steel were considered. The elemental composition assumed is shown in Table 2. The values are the maximum values of molybdenum and manganese allowable in nuclear grade Type 316 stainless steel,⁽⁹⁾ and typical trace values of silicon, niobium, cobalt, copper, vanadium, and aluminum.⁽¹⁰⁾

TABLE 2. Material Composition of Nuclear Grade 316 Stainless Steel

<u>Element</u>	<u>Weight Percent</u>
Fe	61.1
Cr	18.0
Ni	15.0
Mo	2.7
Mn	2.0
Si	0.7
Co	0.2
Cu	0.15
V	0.05
C	0.03
Nb	0.01
Al	0.01

Several coolants have been suggested for tokamak power plants, primarily liquid lithium,^(1,5,6) molten salts⁽³⁾ and helium.^(2,4,7) Liquid lithium has excellent heat transfer properties, but magnetohydrodynamic effects due to the metal flowing through magnetic fields increase the pumping power requirements and may affect coolant piping integrity during major magnetic

disturbances. Molten salt can be used at low pressures, but little is known about the irradiation stability, and the reaction of some molten salts with liquid lithium, graphite, and organic materials may make them undesirable from a safety viewpoint.⁽³⁾ Helium has excellent chemical compatibility and radiation stability, but requires high pressure to make it an attractive heat transfer medium.

High-pressure helium was chosen as the coolant material for the reference design in this study because of its corrosion resistance, chemical, thermal and nuclear properties, and the broad experience associated with its use.

III. BLANKET ACTIVATION

The model and assumptions used to evaluate the rate at which the blanket will be activated are outlined. Methods used to calculate neutron flux and activation product inventory are described. Activation products for the reference blanket are then determined. This section also describes the biological hazard potential of activation products and dose rates. The activity of the reference design is compared with that for a larger tokamak, UWMAK-II.

Most of the neutrons produced in the reactor are absorbed in the blanket. The blanket module investigated in this study was illustrated in Figure 1. The blanket structures, coolant tubes, and shield are stainless steel. The interior of the blanket is filled with liquid natural lithium. The blanket and the detached first wall are cooled by high-pressure helium. Graphite reflectors are used. The graphite and lithium were assumed to be pure. No sorbed gases were considered.

OUTER BLANKET ACTIVATION

The activation product inventories were determined by first calculating the neutron flux throughout the outer blanket and then integrating the activation rates of the various nuclides. For this purpose a one-dimensional model of the outer blanket was assumed. In this model the entire toroidal cross-section is assumed to be composed of modules which are identical to that of Figure 1. This does not represent the geometry of the inner blanket at all, but it is a fair representation of the outer blanket. The toroidal cross-section is modeled as a series of concentric annuli, each with uniform composition. The composition and radii of the zones are shown in Figure 2.

The average plasma radius is 190 cm. The first wall is composed of circular stainless steel tubes filled with helium. Helium is treated as a void in this analysis, so the first wall is modeled as a ring, 3 cm thick, which is 15.7% stainless steel by volume and 84.3% void. There are three

breeding blanket zones separated by graphite reflectors. The reflectors are considered to be pure carbon. The blanket zones are treated as homogeneous mixtures of the various constituents. The inner blanket zone is 76% by volume liquid lithium of natural abundance, 4% stainless steel, and 20% void. The middle blanket zone has more steel and less void due to relaxed heat transfer requirements, and the outer blanket zone has much more steel and less lithium and void. The shield is considered to be 100% stainless steel.

FIGURE 2. Blanket Neutronics Model

Neutron Flux Calculation

The neutron flux was calculated using the discrete ordinates method. The ANISN computer code⁽¹¹⁾ was used in cylindrical geometry. The blanket was treated as an infinitely long cylinder with circular cross-section, and

the one-dimensional neutron transport equation was solved with a $P_3 - S_8$ approximation. The neutron spectrum was divided into 30 energy groups.

The total neutron flux is shown in Figure 3. It can be seen that the attenuation in the blanket is gradual. The total flux at the first wall is 1.0×10^{15} neutrons $\cdot$ cm $^{-2}\cdot$ s $^{-1}$, while the total flux at the outer surface of the shield is 2.2×10^{13} neutrons $\cdot$ cm $^{-2}\cdot$ s $^{-1}$. Although no other shield is specifically identified in the ORNL reference design,⁽³⁾ the magnitude of this flux would necessitate one.

FIGURE 3. Blanket Neutron Flux

The energy-dependent neutron fluxes are shown in Figure 4. It is seen that the spectrum peaks at 14 MeV for the inner zones and has a large resonance-energy population in all zones. The thermal flux is small in the inner regions where the macroscopic lithium cross-section is high, but increases greatly behind the first reflector.

It was assumed that the fluxes determined for the initial composition remain constant throughout the blanket lifetime. This implies that transmutations do not greatly alter the neutronics properties. To support this assumption, the transmutation rates of all the nuclides in the blanket were extrapolated to a ten-year operation time. None of the elemental inventories of the primary constituents changed by more than 1%. The amount of manganese present increased by 5% and the vanadium trace impurity increased by 40%.

FIGURE 4. Blanket Neutron Flux Spectra by Zone

Activation Product Calculation

The GLUCKE computer code⁽¹²⁾ was used to calculate the activation product inventories in the blanket. This code reads the 30-group neutron fluxes calculated by ANISN, processes these with activation cross-section and decay data, and calculates the activation rate of each nuclide. The calculation method is as follows:

Define N_p as the total number of atoms of type p in zone ℓ . The rate of change of N_p is:

$$\frac{dN_p}{dt} = (\text{production rate}) - (\text{decay rate}) - (\text{burnup rate}) \quad (1)$$

Atoms can be produced by activation or decay. The production rate of p atoms due to neutron reactions with nuclide m at point j is:

$$n_m \sum_{i=1}^{30} \phi_{ji} \sigma_{i,m \rightarrow p},$$

where i refers to energy group,

n_m is the number density of nuclide species m ,

ϕ_{ji} is the i^{th} group neutron flux at point j , and

$\sigma_{i,m \rightarrow p}$ is the neutron cross section of the reaction m to p at energy i .

The zone is divided into radial intervals, labeled j , each with volume ΔV_j . The production rate of p atoms due to decay of m atoms is

$$n_m \lambda_{m \rightarrow p},$$

where

$\lambda_{m \rightarrow p}$ is the decay constant of m into p.

The burnup rate due to reactions of type $p \rightarrow m$ is

$$n_p \sum_i \phi_{ji} \sigma_{i, p \rightarrow m}$$

and the decay rate of p atoms is

$$n_p \lambda_{p \rightarrow m}.$$

Adding these terms over all radial intervals in zone ℓ , the rate of change of N_p in zone ℓ is:

$$\begin{aligned} \frac{dN_p}{dt} = \sum_m \left\{ n_m(t) \left[\sum_{j \in \ell} \left(\Delta V_j \sum_{i=1}^{30} \phi_{ji} \sigma_{i, m \rightarrow p} \right) + \Delta V_\ell \lambda_{m \rightarrow p} \right] \right. \\ \left. - n_p(t) \Delta V_\ell \left[\frac{\sum_{j \in \ell} \left(\Delta V_j \sum_{i=1}^{30} \phi_{ji} \sigma_{i, p \rightarrow m} \right)}{\Delta V_\ell} + \lambda_{p \rightarrow m} \right] \right\}, \quad (2) \end{aligned}$$

where

$$\Delta V_\ell \equiv \sum_{j \in \ell} \Delta V_j \text{ and } N_p(t) = n_p(t) \Delta V_\ell$$

The notation $\sum_{j \in \ell}$ indicates summation of all points j which fall within zone ℓ .

To simplify, we define the terms:

$$R_1(t) \equiv \sum_m n_m(t) \left[\sum_{j \in \ell} \left(\Delta V_j \sum_{i=1}^{30} \phi_{ji} \sigma_{i, m \rightarrow p} \right) + \Delta V_\ell \lambda_{m \rightarrow p} \right] \quad (3)$$

and

$$R_2 \equiv \sum_m \left[\frac{\sum_{j \in \ell} \left(\Delta V_j \sum_{i=1}^{30} \phi_{ji} \sigma_{i, p \rightarrow m} \right)}{\Delta V_\ell} + \lambda_{p \rightarrow m} \right] . \quad (4)$$

Now

$$\frac{dN_p}{dt} = R_1(t) - R_2 N_p(t). \quad (5)$$

Since R_1 is a function of $n_m(t)$, equation 5 is really a set of coupled ordinary linear differential equations. Numerical solutions to this by matrix methods are discussed in Appendix A. A simple solution to this equation can be obtained if one neglects the burnup of radioactive nuclides ignores transmutation of stable nuclides. This decouples the set of ordinary differential equations and the solution for the density of a species during reactor operations is therefore,

$$N_p(t) = N_p(0) e^{-R_2 t} + \frac{R_1(t=0)}{R_2} (1 - e^{-R_2 t}) \quad (6)$$

Following shutdown of the reactor after constant power operation for a period of time T , the density is:

$$N_p(t') = N_p(t) e^{-R'_2 t'} + \frac{R'_1}{R'_2} (1 - e^{-R'_2 t'}) \quad (7)$$

where

t' is the time after shutdown

$$R'_1 = \Delta V_\ell \sum_m n_m(T) \lambda_{m \rightarrow p}$$

$$R'_2 = \lambda_{p \rightarrow m}$$

Since these solutions do not properly account for burnup of stable and radioactive species nor treat chain decay correctly, they are approximations and the accuracy of the results must be checked after the computation has been completed. This was done and the transmutation results have been described above. Cross sections for burnup of radioactive species are available only for thermal capture of a very few radionuclides, so radionuclide burnup was ignored. To test the validity of this assumption, the maximum value of $\sigma \phi$ in the blanket (just outside the outer reflector, where the thermal flux is highest) was compared to the decay constant λ for those radionuclides for which σ is available. The only species for which $\sigma \phi > 0.001 \lambda$ are ^{58}Co (2.1% burnup), $^{59\text{m}}\text{Co}$ (0.8%), ^{59}Ni (1.9% burnup in 5 years), ^{63}Ni (0.5% in 5 years), ^{99}Tc (0.4% in 5 years), and ^{94}Nb (0.3% in 5 years).

The 30 energy group cross sections were taken from the barn book.⁽¹³⁾ In evaluating which reactions to include, the following rules were established to reduce the number of reactions and nuclides to a manageable number:

- a) 10% of the nuclide should be transmuted in 10 years. That is, for a flux of 10^{15} neutrons $\cdot \text{cm}^{-2} \cdot \text{s}^{-1}$

$$\sigma_{\text{peak}} > 0.3 \text{ barns, or} \quad (8a)$$

- b) the product contributes 0.5% of the total atoms in 10 yr. That is, for $\phi = 10^{15}$ neutrons $\cdot \text{cm}^{-2} \cdot \text{s}^{-1}$

$$N_i \sigma \geq \frac{0.005 N_t}{\phi t} = 10^{27} \frac{\text{atom-barn}}{\text{m}^3}, \text{ or} \quad (8b)$$

- c) if the product is radioactive, its concentration in 10 yr will be greater than 0.001% (i.e., 10 ppm). That is

$$N_i \sigma \geq \frac{10^{-5} N_t}{\phi t} = \frac{8 \times 10^{32}}{t} \quad (8c)$$

where

t is the lesser of 10 years or the half-life of the product nuclide.

Using these criteria 68 reactions were retained in the GLUCKE library. These reactions are listed in Table 3 and they are later shown in Appendix A, Figure A.2, represented as the Chart of the Nuclides.

TABLE 3. Activation Reactions in GLUCKE Library

${}^6\text{Li}(n,t){}^4\text{He}$	${}^{53}\text{Cr}(n,\gamma){}^{54}\text{Cr}$	${}^{61}\text{Ni}(n,\gamma){}^{62}\text{Ni}$
${}^6\text{Li}(n,n'd){}^4\text{He}$	${}^{54}\text{Cr}(n,\gamma){}^{55}\text{Cr}$	${}^{62}\text{Ni}(n,\gamma){}^{63}\text{Ni}$
${}^7\text{Li}(n,n't){}^4\text{He}$	${}^{55}\text{Mn}(n,2n){}^{54}\text{Mn}$	${}^{64}\text{Ni}(n,\gamma){}^{65}\text{Ni}$
${}^7\text{Li}(n,2n){}^6\text{Li}$	${}^{55}\text{Mn}(n,\gamma){}^{56}\text{Mn}$	${}^{63}\text{Cu}(n,2n){}^{62}\text{Cu}$
${}^7\text{Li}(n,\gamma){}^8\text{Li}$	${}^{54}\text{Fe}(n,\alpha){}^{51}\text{Cr}$	${}^{63}\text{Cu}(n,p){}^{63}\text{Ni}$
${}^9\text{Be}(n,2n){}^8\text{Be}$	${}^{54}\text{Fe}(n,n'p){}^{53}\text{Mn}$	${}^{63}\text{Cu}(n,\gamma){}^{64}\text{Cu}$
${}^{10}\text{B}(n,\alpha){}^7\text{Li}$	${}^{54}\text{Fe}(n,2n){}^{53}\text{Fe}$	${}^{65}\text{Cu}(n,2n){}^{64}\text{Cu}$
${}^{10}\text{B}(n,t){}^8\text{Be}$	${}^{54}\text{Fe}(n,p){}^{54}\text{Mn}$	${}^{65}\text{Cu}(n,\gamma){}^{66}\text{Cu}$
${}^{10}\text{B}(n,n'd){}^8\text{Be}$	${}^{54}\text{Fe}(n,\gamma){}^{55}\text{Fe}$	${}^{93}\text{Nb}(n,2n){}^{92}\text{Nb}$
${}^{11}\text{B}(n,\alpha){}^8\text{Li}$	${}^{56}\text{Fe}(n,n'p){}^{55}\text{Mn}$	${}^{93}\text{Nb}(n,\gamma){}^{94}\text{Nb}$
${}^{12}\text{C}(n,n'\alpha){}^8\text{Be}$	${}^{56}\text{Fe}(n,2n){}^{55}\text{Fe}$	${}^{92}\text{Mo}(n,n'p){}^{91}\text{Nb}$
${}^{14}\text{N}(n,p){}^{14}\text{C}$	${}^{56}\text{Fe}(n,p){}^{56}\text{Mn}$	${}^{92}\text{Mo}(n,2n){}^{91}\text{Mo}$
${}^{27}\text{Al}(n,2n){}^{26}\text{Al}$	${}^{56}\text{Fe}(n,\gamma){}^{57}\text{Fe}$	${}^{92}\text{Mo}(n,p){}^{92}\text{Nb}$
${}^{28}\text{Si}(n,\alpha){}^{25}\text{Mg}$	${}^{57}\text{Fe}(n,\gamma){}^{58}\text{Fe}$	${}^{95}\text{Mo}(n,\gamma){}^{96}\text{Mo}$
${}^{28}\text{Si}(n,p){}^{28}\text{Al}$	${}^{58}\text{Fe}(n,\gamma){}^{59}\text{Fe}$	${}^{96}\text{Mo}(n,\gamma){}^{97}\text{Mo}$
${}^{35}\text{Cl}(n,\gamma){}^{36}\text{Cl}$	${}^{59}\text{Co}(n,2n){}^{58}\text{Co}$	${}^{97}\text{Mo}(n,\gamma){}^{98}\text{Mo}$
${}^{51}\text{V}(n,\gamma){}^{52}\text{V}$	${}^{59}\text{Co}(n,p){}^{59}\text{Fe}$	${}^{98}\text{Mo}(n,\gamma){}^{99}\text{Mo}$
${}^{50}\text{Cr}(n,n'p){}^{49}\text{V}$	${}^{59}\text{Co}(n,\gamma){}^{60}\text{Co}$	${}^{100}\text{Mo}(n,2n){}^{99}\text{Mo}$
${}^{50}\text{Cr}(n,p){}^{50}\text{V}$	${}^{58}\text{Ni}(n,n'p){}^{57}\text{Co}$	${}^{100}\text{Mo}(n,\gamma){}^{101}\text{Mo}$
${}^{50}\text{Cr}(n,\gamma){}^{51}\text{Cr}$	${}^{58}\text{Ni}(n,p){}^{58}\text{Co}$	${}^{204}\text{Pb}(n,2n){}^{203}\text{Pb}$
${}^{52}\text{Cr}(n,2n){}^{51}\text{Cr}$	${}^{58}\text{Ni}(n,\gamma){}^{59}\text{Ni}$	${}^{204}\text{Pb}(n,\gamma){}^{205}\text{Pb}$
${}^{52}\text{Cr}(n,p){}^{52}\text{V}$	${}^{60}\text{Ni}(n,p){}^{60}\text{Co}$	${}^{206}\text{Pb}(n,2n){}^{205}\text{Pb}$
${}^{52}\text{Cr}(n,\gamma){}^{53}\text{Cr}$	${}^{60}\text{Ni}(n,\gamma){}^{61}\text{Ni}$	

Many other parameters are readily determined once the activation product inventories are known. These parameters include activities in Curies, activity concentration, biological hazard potential, and radioactive decay heat.

The results obtained for neutron activation of the reference outer blanket are described in the remainder of this section. Since ANISN and GLUCKE are one-dimensional codes, the plasma and blanket were modeled as uniform cylinders. The neutron source strength is 1.68×10^{17} neutrons $\times$ cm^{-1} -s, which corresponds to a power of $0.473 \text{ MW} \times \text{cm}^{-1}$, or an average neutron wall loading of $2.87 \text{ MW} \times \text{m}^{-2}$. Since the major radius is 6 m, the total thermal power is 1783 MW during operation. With a 95% duty factor, the average thermal power is 1700 MW. The replacement period for blanket modules is five years.

The total activity during reactor operation is shown in Figure 5. After initial startup the blanket activity rises very rapidly to values around 10^9 Ci. After only one day of full power operation the activity is already 0.7×10^9 Ci. Even after the first 20-minute burn the activity is 0.17×10^9 Ci. After the first day of operation the rate of increase of activation slows down considerably. The total activity is 0.95×10^9 Ci after one month, 1.56×10^9 Ci after one year, and 2.06×10^9 Ci at the time of blanket replacement after five years of operation.

The concentrations after five years are shown in Table 4; the first wall has by far the highest concentration of activity. Figure 5 shows that most of the activity is produced in the first wall and inner blanket region as expected, since the flux is highest and most energetic there. It is significant to note, however, that activation of the shield is substantial. At shutdown the activity of the shield is 0.25×10^9 Ci. Zones 6 and 8 are graphite and therefore have much lower activities. The activity concentrations show a much greater range of values than the zone totals.

TABLE 4. Activity Concentrations at Shutdown

Zone	Activity (Ci)	Volume (m ³)	Concentration (Ci x cm ⁻³)
3. First Wall	5.85×10^8	15.0	38.9
5. Inner Blanket	8.03×10^8	184.0	4.36
7. Middle Blanket	1.56×10^8	72.6	2.15
9. Outer Blanket	2.63×10^8	34.5	7.63
10. Shield	2.54×10^8	63.3	4.01
Total	2.06×10^9	466.8 ^(a)	4.41 ^(a)

(a) Includes volume of graphite zones.

FIGURE 5. Total Blanket Activity by Zone

The calculated contributions of each significant radionuclide to the total blanket activity are shown in Figure 6. The most dominant radionuclide in terms of total activity is ^{56}Mn ; 54% of the ^{56}Mn is produced from the ^{55}Mn additive in steel and the remainder from ^{56}Fe . Since ^{56}Mn has a short half-life (2.58 hr), its activity saturates very quickly after startup. One day after startup ^{56}Mn contributes 77% of all the activity in the blanket. After five years of operation the activity due to ^{55}Fe approaches that of ^{56}Mn . The ^{55}Fe decays by electron capture with no gamma emissions, so it is relatively innocuous. The three radioactive cobalt isotopes, ^{57}Co , ^{58}Co and ^{60}Co contribute a total of 4.74×10^8 Ci after five years--about as much activity as ^{55}Fe but with a much greater biological significance. These nuclides are produced by (n,2n) and (n, γ) reactions with the ^{59}Co impurity and by (n,p) reactions with the nickel. Production of ^{57}Co is solely by the $^{58}\text{Ni}(n,n'p)$ reaction, ^{58}Co production is 99% by $^{58}\text{Ni}(n,p)$ and 1.3% by $^{59}\text{Co}(n,2n)$, while 27% of all ^{60}Co is produced by neutron capture in ^{59}Co and 73% by $^{60}\text{Ni}(n,p)$ reactions. It is worth noting that 69% of the ^{60}Co produced in the shield is due to neutron capture in the cobalt impurity. This is due to the neutron thermalization in the outer blanket regions and the large (37 barn) thermal cross-section of ^{59}Co . Other activation products resulting in activities greater than 5×10^7 Ci after five years are ^{51}Cr produced from ^{50}Cr and ^{52}Co , ^{54}Mn produced from ^{54}Fe and ^{55}Mn , and ^{99}Mo produced from ^{98}Mo and ^{100}Mo .

The blanket activity during the first day of operation is dominated by radionuclides with short half-lives. Ninety percent or more of the activity during the first day results from four activation products: ^{56}Mn , ^{52}V , ^{28}Al and ^{101}Mo ; 54% of the ^{56}Mn is produced from manganese, 99% of the ^{52}V is produced from ^{52}Cr , all of the ^{28}Al is produced from silicon, and all of the ^{101}Mo is produced from molybdenum. This means that 30% of all the activity at ten minutes after startup results from steel impurities or additives other than iron, nickel, and chromium. Two hours after startup this value rises to 44%, mainly due to ^{56}Mn , and at the end of the first day of full power operation 48% of the blanket activity is due to steel impurities and additives. Out of a total of 7.06×10^8 Ci, 2.95×10^8 Ci is

FIGURE 6. Blanket Activity During Operation, by Nuclide

due to the $^{55}\text{Mn}(n,\gamma)^{56}\text{Mn}$ reaction, 1.79×10^7 Ci is due to the $^{28}\text{Si}(n,p)^{28}\text{Al}$ reaction, 1.30×10^7 Ci is due to the $^{100}\text{Mo}(n,\gamma)^{101}\text{Mo}$ reaction, and 1.16×10^7 Ci is due to $^{98}\text{Mo}(n,\gamma)^{99}\text{Mo}$ and $^{100}\text{Mo}(n,2n)^{99}\text{Mo}$.

The fraction of the total activity caused by impurities decreases after the first few days. After five years of full power operation, 20% of the total activity is due to impurities and additives.

BIOLOGICAL HAZARD POTENTIAL

The use of activity as a measure of radiation hazard fails to address the fact that some radionuclides represent a greater biological impact than others. The relative hazards of various radionuclides can be described by their Biological Hazard Potential (BHP), which is defined as the volume of air required to dilute the radioactive source to the maximum permissible concentration in unrestricted areas. The BHP is calculated as the sum of the ratios of the blanket activity due to each nuclide to the maximum permissible concentration (MPC) of the nuclide as given in 10 CFR 20 Appendix B.

$$\text{BHP} = \sum_i \frac{A_i}{(\text{MPC}_{\text{air}})_i}$$

where

A is the activity of nuclide i , and
 $(\text{MPC}_{\text{air}})_i$ is the maximum permissible concentration of nuclide i in unrestricted air.

The BHP of the blanket as a function of time is shown in Figure 7. The dominant nuclide is ^{56}Mn at early times, but when the blanket is replaced after five years of operation, ^{58}Co and ^{54}Mn contribute 70% of the total BHP. The combined total of cobalt and manganese isotopes is responsible for 93% of the total BHP, with ^{55}Fe and molybdenum isotopes adding another 6%.

FIGURE 7. Biological Hazard Potential of Blanket

DOSE RATE

Another relevant measure of safety hazard besides activity or BHP is dose rate, because it is a measure of biological hazard of radiation fields to which personnel might actually be exposed. The contact dose rates were estimated for various activation products near the shield and near the first wall. For an infinite slab the dose rate is:⁽¹⁴⁾

$$\phi = DF \frac{S_v}{2\mu_s} [1 - E_2(\mu_s h)]$$

where DF is the gamma dose-to-flux conversion factor⁽¹⁵⁾, S_v is the volumetric gamma source strength, μ_s is the macroscopic total gamma cross-section, h is the slab thickness, and E_2 is an exponential integral function. Buildup factors were also considered and found to have a minor impact. The contact dose rate at the edge of the first wall and shield after one day and after five years of full power operation are tabulated by nuclide in Table 5.

TABLE 5. Contact Dose Rates from Shield and First Wall (rem/hr) at the Time of Shutdown for 1 Day and 5 Year Continuous Operation

Nuclide	First Wall		Shield	
	1 day	5 yrs	1 day	5 yrs
⁵⁴ Mn	1.3×10^3	5.6×10^5	2.6×10^1	1.1×10^4
⁵⁶ Mn	2.6×10^6	2.6×10^6	2.7×10^5	2.7×10^5
⁵¹ Cr	4.0×10^2	1.6×10^4	7.9×10^0	3.2×10^2
⁵² V	5.5×10^5	5.5×10^5	1.0×10^4	1.0×10^4
⁵⁷ Co	1.8×10^2	6.8×10^4	2.8×10^0	1.1×10^3
⁵⁸ Co	1.1×10^4	1.2×10^6	3.1×10^2	3.2×10^4
⁶⁰ Co	1.0×10^2	1.4×10^5	7.0×10^0	9.3×10^3
²⁸ Al	1.5×10^5	1.5×10^5	3.6×10^3	3.6×10^3
Total	3.3×10^6	5.3×10^6	2.8×10^5	3.4×10^5

The dose rate at both the first wall and shield is dominated by the presence of ^{56}Mn for short and long operation periods. Other nuclides of lesser interest to the dose rates for early operations are ^{52}V and ^{28}Al . For later operations ^{58}Co , ^{54}Mn , ^{52}V and ^{60}Co are significant at the shield, while ^{58}C , ^{54}Mn and ^{52}V are significant at the first wall.

COMPARISON WITH LARGE REACTOR DESIGN

Comparison of the blanket activity associated with the reference tokamak design with the blanket activity of an earlier conceptual design provides useful information. UWMAK-II⁽²⁾ is a large tokamak design with stainless steel structures, helium coolant, and solid tritium breeding materials. A one-dimensional neutronics calculation was performed using the ANISN code, and activation calculations were performed with GLUCKE. To provide a comparison of results with the calculations performed for the UWMAK-II design, all materials were considered to be without impurities or additives.

The UWMAK-II blanket modules were divided into two parts: an inner blanket region to be replaced every two years; and an outer blanket with a ten-year lifetime. The GLUCKE code was run for a two-year operation time with an average thermal power of 3760 MW (80% plant factor, 94% duty factor) to determine most activation properties. One case was run, however, for ten years operation to provide a check on GLUCKE's results as compared to those of the UWMAK-II report. A one minute after shutdown following ten years operation, GLUCKE predicts a total blanket activation of 5.43×10^9 Ci. The prediction of the UWMAK-II report is 6.22×10^9 Ci. The discrepancy occurs because the UWMAK-II calculation assumed a 100% plant factor. The activity of the first wall determined by GLUCKE at shutdown following ten years operation is 3.24×10^9 Ci. Table 6 compares the predicted contributions of the major isotopes using 3.24×10^9 Ci. The GLUCKE code for the Steiner-Demo to those predicted for UWMAK-II.

TABLE 6. Comparison of Major Nuclide Contributions to the First Wall Activity Estimated by the GLUCKE code, after 10 Years of Operation at 5000 MWt (no impurities), and UWMAK-II Results

<u>Nuclide</u>	<u>GLUCKE no Impurities for Steiner-Demo</u>	<u>UWMAK-II</u>
^{55}Fe	43%	50%
^{56}Mn	14	14
^{58}Co	13	7
^{57}Co	11	25
^{51}Cr	10	10
Total	$4.31 \times 10^9 \text{ Ci}^{(a)}$	$3.43 \times 10^9 \text{ Ci}$

Scaled to a 100% plant factor and 100% duty factor for comparison for an operating life of 10 years. The total activation shown in the table for the GLUCKE results is larger than the $3.24 \times 10^9 \text{ Ci}$ in order to compare the two cases on a common ground.

Comparing the UWMAK-II design to the compact tokamak design used in this report, the total activity at time of replacement (after two years for UWMAK-II and five years for the ORNL design) is $4.1 \times 10^9 \text{ Ci}$ for UWMAK-II and $2.1 \times 10^9 \text{ Ci}$ for the reference design. Normalizing the activity in terms of power output, UWMAK-II activity is 0.82 Ci/W at time of replacement while the compact design is $1.05 \text{ Ci} \times \text{W}^{-1}$. In comparing these values it is significant that impurities were not considered in the UWMAK calculation. If impurities had been neglected in the compact tokamak, the calculated activity at shutdown would have been $0.83 \text{ Ci} \times \text{W}^{-1}$ -- essentially the same as the larger design. Clearly, the activity concentration of the compact design is much higher at shutdown than that of the UWMAK because of the difference in

physical size. Viewed in terms of waste management, however, the blanket activity to be replaced annually is $0.49 \text{ Ci} \times \text{W}^{-1}\text{-yr}^{-1}$ for UWMAK-II and $0.24 \text{ Ci} \times \text{W}^{-1}\text{-yr}^{-1}$ for the modified Steiner-Demo design used as the reference for the current study. This difference is not a result of the smaller physical size of the latter; rather, it reflects the longer blanket lifetime assumed for the Steiner-Demo design.

ALTERNATE STRUCTURAL MATERIALS

A number of structural materials have been proposed besides stainless steel. Among these are aluminum, vanadium, titanium, and alloys of molybdenum or nickel. A review of radioactivity levels with most of these materials has been presented by Vogelsang.⁽¹⁶⁾ For his analysis, he calculated the neutron flux throughout the outer blanket of UWMAK-I.⁽¹⁾ Using this neutron flux, he determined the activity of the stainless steel blanket described in UWMAK-I. The activity for alternate materials was calculated by straight substitution of structural materials with identical volume and geometry. The neutron flux was assumed to be identical for the different materials and no impurities were considered. Vogelsang's results are shown in Figure 8.

FIGURE 8. Blanket Activity During Operation with Alternate Structural Materials

The activity of Nb-1Zr and TZM (molybdenum-based alloy with 0.45% Ti and 0.1% Zr) is considerably higher than that of 316 stainless steel. After two year's operation, Nb-1Zr is about five times as radioactive, and TZM four times, as steel. The V-Ti alloy considered (80% V, 20% Ti) is somewhat more active than steel, while aluminum is somewhat less after one month of operation. Of all the materials considered, the rate at which blanket activity builds up for early operation time is lowest for the 316 stainless steel. After one day, all materials except steel have at least half of their life-time activity, while after ten days all but steel have reached 90% of their 30-year level. For steel these levels are 40% and 50% respectively.

Vogelsang's results for 316 Stainless Steel show an activity level that is about 20% lower after five years than was calculated for this study. With his assumption of no impurities, this is consistent with our results.

Other radioactivity properties of interest are afterheat and biological hazard potential. Vogelsang's results for these parameters are shown in Table 7. All values are given at shutdown following two years' operation in UWMAK-I (5000 MWt).

TABLE 7. Radioactive Properties of Various Blanket Materials(17)

<u>Alloy</u>	<u>Specific Activity (Ci x W⁻¹)</u>	<u>First Wall Activity Concentration (Ci x cm⁻³)</u>	<u>Afterheat (% of Operating Power)</u>	<u>(km³^{BHP} air x W⁻¹)</u>
316 SS	1.0	100	0.9	0.23
TZM	3.9	125	1.5	0.35
V-20 Ti	1.2	27	1.6	0.036
2024 Al	0.6	44	1.2	0.071
Nb-1Zr	5.1	158	0.5	0.040

Vogelsang's results for stainless steel are 10-20% higher than our own, reflecting his neglect of impurities. Molybdenum has high values of all properities after two years: total activity, first wall activity, afterheat, and BHP. Vanadium has low first wall activity and low BHP, but high afterheat, which is also true for aluminum. Niobium has very high activity levels, but very low afterheat and biological hazard potential.

Several other authors have investigated activation of fusion blankets with alternate materials. Iida, Seki, and Ide⁽¹⁷⁾ examined activation of a molybdenum blanket in the JAERI reactor.⁽¹⁸⁾ They concluded that, because of the short-lived nuclides ^{101}Mo , ^{101}Tc , ^{99}Mo , $^{99\text{m}}\text{Tc}$, and ^{91}Mo , the activity reaches 90% of its 2-year value within 10 days after the start of operation. The activity level after two years is 5.4×10^9 Ci, or $2.7 \text{ Ci} \times \text{W}^{-1}$. This is somewhat lower than Vogelsang indicates, but substantially higher than the steel blanket considered here.

Steiner⁽¹⁹⁾ calculated the activity and decay power of a niobium blanket. Two years after startup, the activity is $2.0 \text{ Ci} \times \text{W}^{-1}$, considerably lower than Vogelsang's results. The decay heat at that time is 0.65% of operating power.

Powell, et al⁽²⁰⁾ considered activation of aluminum structures with commercial-grade 1199 aluminum, including impurities. The inventory of ^{23}Na , produced by the $^{27}\text{Al} (n, \alpha)$ fast neutron reaction, is 1.9 or 1.2×10^9 Ci for a 1 GWe reactor, depending on the reactor design. Assuming thermal efficiencies of 28% and 44%, respectively, this is 6.7 or $2.7 \text{ Ci} \times \text{W}^{-1}$. The half-life of ^{24}Na is 15 hours, so that one day after shutdown, the activity is 4×10^3 Ci and falling rapidly.

The General Atomic Demonstration Power Reactor Design⁽⁴⁾ uses a nickel-based alloy as a structural material. The blanket used Inconel 718 and the shield is stainless steel. The total activity after two years is 2.3×10^9 Ci, or $1.4 \text{ Ci} \times \text{W}^{-1}$, about 25% higher than that of the steel blanket.

IV. NEUTRAL BEAM INJECTORS

The calculation of neutron flux used in earlier chapters assumed one-dimensional geometry. Although this is a reasonable assumption for calculating activity in most of the blanket, it does not apply to penetrations. Vacuum vessel penetrations are necessary for neutral beam injection heating, vacuum ports, and plasma diagnostics. They provide pathways for neutrons and photons to pass unattenuated through the blanket structure. As a result of streaming through penetrations, several reactor components are exposed to considerably more radiation than would be predicted using one-dimensional calculations. Among these are the neutral beam injectors, toroidal field coils adjacent to the injector ports, and vacuum cryopanel.

Several authors have investigated neutron streaming through tokamak penetrations. Abdou⁽²¹⁾ performed three-dimensional Monte Carlo calculations of the neutron flux within penetrations of the ANL-EPR design.⁽²²⁾ The internal structure of the injector was represented as a 5-cm thick stainless steel end cap to quantify the level of streaming into the beam injectors and vacuum pumps. Abdou calculated fluxes within the beam duct, penetration shields, and toroidal field coils for a variety of penetration and shield configurations. In all cases of 85-cm diameter ducts, he calculated a neutron leakage outside the bulk shield of greater than 0.35% of all the neutrons produced. This represents a very strong neutron source for activation of adjacent reactor components and reactor building atmosphere.

Santoro⁽²³⁾ estimated the activation level of toroidal field coils, neutral beam injectors, and the beam calorimeter in TFTR. He used the one-dimensional transport code ANISN, in cylindrical geometry. The cylindrical axis was the plasma minor axis, and the internal structure was represented by fourteen radial zones, alternating between iron and void. This approach tends to overestimate the high-energy, direct streaming flux while ignoring the neutron flux which has been moderated and attenuated in the blanket or duct walls. The specific activity of the unshielded toroidal field coil of TFTR calculated in this manner is about $0.005 \text{ Ci} \times \text{cm}^{-3}$ after eight hours of continuous operation. Most of this activity stems from the presence of ^{64}Cu

and ^{62}Cu . This activity results in an unshielded dose rate of $84 \text{ rem} \times \text{hr}^{-1}$ at shutdown at 1 m from the face of the unshielded TF coil. The specific activity of the innermost zone of the neutral beam injector is $2.24 \times 10^{-3} \text{ Ci} \times \text{cm}^{-3}$. The dose rate at shutdown after eight hours is $288 \text{ rem} \times \text{hr}^{-1}$ at a distance of 2 m from the front surface and 1 m to the side of the injector.

The two-dimensional discrete-ordinates code TWOTRAN-GG⁽²⁴⁾ was used by Ide, et al⁽²⁵⁾ to investigate neutron streaming effects in penetrations in the JAERI Demonstration Reactor.⁽¹⁸⁾ The D-T neutron flux at the first wall of this reactor design is $5.711 \times 10^{13} \text{ n} \times \text{cm}^{-2} \times \text{s}^{-1}$. At the outer edge of the injection ports, the neutron flux of energy over 0.1 MeV is about $10^{13} \text{ n} \times \text{cm}^{-2} \times \text{s}^{-1}$. Ide calculated the induced activity of the injector components and piping with the code ACTIVE.⁽²⁶⁾ At shutdown following one year of operations, the specific activity of the neutral beam injector is $3.6 \text{ Ci} \times \text{cm}^{-3}$, of which 89% is due to copper isotopes, especially ^{64}Cu , and 11% is due to the 316 stainless steel.

The neutron radiation deep inside the injector port has three sources: neutron flux attenuated in the blanket; direct streaming of neutrons produced in the plasma; and penetration-assisted streaming. The first source would be present even if there were no penetration; neutrons entering the blanket are scattered and absorbed by blanket materials, resulting in a moderated, attenuated neutron flux throughout the entire blanket region. High-energy neutrons ($\sim 14 \text{ MeV}$) produced in the plasma can have a direct line-of-sight pathway to the injector internals. The resulting neutron flux is essentially a Doppler-broadened 14.1 MeV spectrum. Neutrons which make only a few collisions with blanket materials near the duct walls and then react with the injector materials may still have higher energies than the blanket flux and are much less attenuated.

The activation level of the neutral beam injectors in the reference tokamak design is estimated in the following discussion. The model used to estimate the neutron flux includes direct streaming and blanket-attenuated fluxes.

The internal structure of the injector is represented as a slab of material at the end of a duct of length d . The duct is assumed to be perpendicular to the plasma axis, which is modeled as a right circular cylinder of radius R . The duct cross section is modeled as a square with length x . With this model, the probability that a neutron born within the line-of-sight of the injector, represented by the shaded area of the plasma in Figure 9, reaches the injector is

$$\frac{x^2}{4\pi d^2} \text{ if } d \gg R.$$

For a neutron source density S in the plasma, the rate at which neutrons are produced within the line-of-sight region is

$$W = 2Sx \int_{-\frac{x}{2}}^{\frac{x}{2}} \sqrt{R^2 - y^2} dy = Sx \left(x \sqrt{R^2 - \frac{x^2}{4}} + 2R^2 \sin^{-1} \frac{x}{2R} \right).$$

The flux at the neutral beam injector due to direct neutron streaming is thus, assuming $d \gg R$,

$$\phi = \frac{W}{4\pi d^2} = \frac{Sx}{4\pi d^2} \left(x \sqrt{R^2 - \frac{x^2}{4}} + 2R^2 \sin^{-1} \frac{x}{2R} \right).$$

FIGURE 9. Injector Streaming Model

For the reference case, we assume that

$$\begin{aligned}x &= 50 \text{ cm}, \\d &= 292 \text{ cm}, \\r &= 190 \text{ cm}, \\ \text{and } S &= 1.48 \times 10^{12} \text{ n} \cdot \text{cm}^{-2} \cdot \text{s}^{-1}.\end{aligned}$$

The direct streaming neutron flux inside the injector is then

$$\phi = 1.31 \times 10^{12} \text{ n} \cdot \text{cm}^{-2} \cdot \text{s}^{-1}.$$

Superimposed on this flux is that due to neutrons attenuated in the blanket. Treating the penetration as a perturbation in the blanket geometry, the attenuated flux is that calculated by ANISN at the outer surface of the blanket. The spectrum was shown in Figure 4. This total neutron flux is $8.20 \times 10^{13} \text{ n} \cdot \text{cm}^{-2} \cdot \text{s}^{-1}$. This value is still much higher than that of the streaming neutrons, but the high energy nature of the latter will affect the activation characteristics.

The activation product inventory of the neutral beam injectors calculated in this manner is shown in Figure 10. After ten years of full power operation, the injector activity is $6.1 \times 10^4 \text{ Ci}$. For the first few months this activity is strongly dominated by ^{56}Mn ; ^{55}Fe becomes substantial after several years.

The contact dose rate at the plate representing the neutral beam injector immediately after shutdown is $0.98 \times 10^6 \text{ rem} \cdot \text{hr}^{-1}$ after one day of full power operation, of which $0.92 \times 10^6 \text{ rem} \cdot \text{hr}^{-1}$ is due to ^{56}Mn decay. After one year of operation, the dose rate is $1.2 \times 10^6 \text{ rem} \cdot \text{hr}^{-1}$, and the dose rate increases only slightly from this value after ten years of operation. These values have strong implications for safe reactor design. Careful radiation shielding analyses must be performed to ensure the safety of the plant personnel performing maintenance on the neutral beam injectors or any adjacent equipment.

FIGURE 10. Injector Activation vs. Time

V. VACUUM SYSTEM ACTIVITY

Design studies on tokamak fusion reactors using the D-T fuel cycle have indicated that intense irradiation of the first wall will result in erosion and activation product transport in the vacuum system. To provide the necessary containment and shielding, all radionuclide inventories must be identified both in physical form and location. The mass transport of these materials through reactor systems makes prediction of radiation levels in the event of required maintenance or prediction of inventories available for accidental release quite difficult.

Ideally, radioactive material removed from the first wall would be confined to as few systems as possible to reduce contamination for maintenance purposes or to reduce the potential for accidental release. However, the plasma physics considerations may dictate the diversion of impurities from the vacuum vessel, possibly compromising containment. In the major reactor design studies done to date, such as the UWMAK-I through -III series,^(1,2,6) the deposition of material removed from the first wall is not discussed in detail; only gross removal inventories are predicted for a wall in radioactive equilibrium. The efficiency of impurity collection systems is generally given for hydrogen only, leading some authors to believe that the first wall erosion products will be transported out of the vacuum vessel and collected by the vacuum pumps in designs such as UWMAK-I. This is not necessarily the case and a more detailed discussion of the vacuum system and impurities involved is required.

VACUUM SYSTEM IMPURITIES

In the vacuum vessel of a tokamak reactor, a number of mechanisms are available for introducing activated materials into the system. The major source of impurities will be the vacuum wall itself. In today's physics experiments outgassing and desorption of surface layers dominate the influx of impurities.⁽²⁷⁾ However, with longer burn times and higher wall loadings expected during reactor operating conditions, the actual erosion of the first wall will dominate impurity influx in fusion reactors. The phenomena of interest will be physical sputtering, chemical sputtering, blistering and evaporation.

Impurities may also originate from leaks in the vacuum system around pumps, seals, etc. Such leaks are expected to be negligible during normal operations. Fuel injection systems such as pellet injectors are assumed to be free of impurities; however, neutral beam injectors may introduce impurities from particle sputtering on grids and other surfaces along the beam line. No real information is available on impurities from leaks and injection systems,⁽²⁸⁾ so we assume that sputtering, blistering and evaporation dominate. Direct ejection of radioactive recoils from wall material also will not be considered.

The reference design under consideration will determine which sources of impurities are applicable. For example, with stainless steel first walls, physical sputtering and blistering due to particle bombardment will be the phenomena of interest under normal operating conditions. Accidental plasma dumps to the first wall will not be considered. Reference values for sputtering and blistering will be established after an impurity flow model is developed.

Once removed from the first wall, the transport and final deposition of material will depend on the specific impurity involved as well as the physical conditions which exist in the main torus, divertor region and vacuum system. This will include the geometry of the system for high Z impurities, as will be discussed below.

The transport and deposition of metal alloy constituents removed by erosion will be generically different from the transport of volatile compounds or gases released from the wall. Gases are expected to be pumped from the vacuum vessel regardless of cavity temperatures. Compounds which are volatile under conditions in the reactor cavity may remain in the gaseous phase and be pumped out through the system, but will likely condense on piping surfaces or the outer chevrons of cryopumps designed to retain hydrogen.⁽⁶⁾ In any event, the activation products of stainless steel are not associated with volatile compounds. The metallic first wall constituents are chosen for the extremely low vapor pressures under the expected operating conditions ($\sim 500^\circ\text{C}$). The wall will be highly conditioned during startup and operation to remove gases and contaminated layers with techniques being used now in machines like ORMAK.⁽²⁹⁾ Under high vacuum conditions ($\sim 10^{-4}$ to 10^{-5} torr) these surfaces will likely promote excellent adhesion for sputter deposition of materials removed and later transported back to the wall.

The deposition of material back to the vacuum wall during operation is expected to be at a grazing angle of incidence, as was found in ORMAK⁽³⁰⁾ where small protrusions in the wall shielded the liner from deposits. Deposition of impurities streaming from the plasma in the absence of magnetic fields at the end of the burn is also expected to be a line-of-sight phenomenon due to the long mean free paths and high sticking probabilities expected. In laboratory sputtering of materials with discharge gas pressures in the 10^{-3} to 10^{-4} torr range, mean free paths are on the order of 4 cm.⁽³¹⁾ At the lower tokamak operating pressures and higher particle energies, much larger mean free paths can be expected in the reactor. For these reasons, and because the exhaust ports can be shielded and/or baffled, no high Z metallic impurities are expected to be transported far into the exhaust manifolds. For designs relying on cryo-pumping surfaces located within the vacuum vessel, the exhaust manifolds leading to backup pumps or tritium cleanup systems can actually be valved off, assuring deposition in the vacuum vessel. At the end of plasma burn a tokamak reactor would likely include a cooldown cycle by impurity injection, followed by a current interrupt with subsequent loss of confinement. The dump to the first wall will cause additional sputtering but this material is expected to again deposit out. The cooldown cycle will have to be tailored to preclude significant evaporation as well as thermal stresses caused by the plasma dump.

All larger reactor designs to date include a magnetic divertor to channel a significant fraction of the ionized impurities out of the main vacuum vessel to some type of collection bed. A flowing lithium collector was used in UWMAK-I and -II, and was further refined in UWMAK-III, to reduce backstreaming of impurities to the plasma. The divertor efficiencies are projected as 90% and 99% respectively. The collection system used in UWMAK-III is designed to provide an efficiency of 99.9% using sputter deposition on particle bombardment plates. The ports leading field lines to the plates are also designed to reduce the likelihood of backstreaming of resputtered material.

For the collection systems relying on sputter deposition, the eroded first-wall material would be entrained permanently in a region requiring no routine maintenance, located behind the reactor shielding along the first wall. However, for designs using a flowing lithium surface, the eroded first-wall

material will be transported to some cleanup system. The resulting radiation fields caused by crud buildup in pipes, etc., will be design dependent and so only gross inventories will be identified.

In the event of vacuum vessel failure, the radionuclide inventory in the plasma may be transported in part into the reactor containment building. The exact fraction of the inventory cannot be determined in this paper, therefore the full inventory will be used as a potential upper bound to assess the impact on building containment, aerosol blowdown systems, etc.

The above discussion on wall removal and deposition will now be incorporated into a simple impurity flow model to estimate radionuclide inventories in the plasma and on collection surfaces.

IMPURITY FLOW MODEL

Several impurity flow models for tokamaks have been developed recently; the most detailed are models by Stacey, et al.⁽³²⁾ and by Sucov.⁽³³⁾ These models were developed primarily to determine the effect of impurities on plasma performance. In this paper radionuclide inventories in the plasma and on the collector plates are at issue. The models will therefore be changed slightly to produce the impurity flow diagram shown in Figure 11.

The flow diagram depicts an impurity source at the vacuum wall introducing material into a plasma scrapeoff region. Here the magnetic divertor fields can sweep impurities into the divertor with a probability, P_D . The remaining impurities move into the plasma.

Once in the divertor, the ionized impurities have three options:

1. they may impact a collector plate with probability P_C ,
2. they may enter the vacuum system manifolds (highly unlikely) with probability P_V , or
3. they may simply sweep through the divertor with probability $(1 - P_C - P_V)$.

Depending on the design, secondary sputtering and reemission are not completely eliminated, and so a permanent entrainment probability, P_e , is shown. For

impurities reaching the plasma, a leakage term back into the scrapeoff region is assumed. This source term is assumed to be the ratio of impurity density and confinement time in the plasma. However, the differential equations in this case will be consistently solved for impurity inventory rather than densities. The source term shown is then the ratio of impurity inventory to confinement time, in atoms per second. This material can then be transported to the divertor, or appear as a flux in the vacuum wall ($1 - P_D$), resulting in additional self-sputtering.

FIGURE 11. Vacuum Impurity Flow Diagram

The model is simplified in that the same divertor efficiency is used for impurities coming in from the walls ("shielding") or leaking out from the plasma ("unloading"). For impurity terms the model will further be simplified by assuming that:

1. transmutation effects on the chemical composition of alloys over the reactor lifetime will not be considered, and
2. burnup of activation products once generated in the vacuum wall will not be considered.

Although the scrapeoff region and divertor will undoubtedly have some characteristic residence time, it will be assumed that particles are immediately passed through these regions with no time delay.

The time-dependent differential equations generated by this model have been solved for the buildup of radionuclides in the plasma during each burn and in the collectors over a 30-year lifetime.

The reactor operating characteristics chosen to run the model are shown in Table 8. Because of the variation in wall areas among designs, the wall area of the reference design (Steiner Demo) will be normalized to 1000 m². This is consistent with the trend towards smaller tokamak designs, and allows for simple scaling and comparison of final results.

TABLE 8. Reactor Operating Parameters (1,2,6,34)

	Burn Time (sec)	Down Time (sec)	Duty Factor	Plant Factor	Wall Area (m ²)	Ion Confinement Time Tp (sec)	Divertor Efficiency (%)	14.1 MeV Neutron Wall Loading (MW·m ⁻²)	14.1 MeV Neutron Current (cm ⁻² ·s ⁻¹)	Total Flux n·cm ⁻² ·s ⁻¹
UWMAK-I	5400	390	0.927	0.80	2830	14.2	90	1.25	5.5 x 10 ¹³	4.34 x 10 ¹⁴
UWMAK-II	5400	330	0.942	0.80	2830	8.28	99	1.16	5.1 x 10 ¹³	3.5 x 10 ¹⁴
UWMAK-III	1800	100	0.947	(?)	1600	3.33	99.9	2.0	8.8 x 10 ¹³	9.0 x 10 ¹⁴
Princeton FPP	6000	180	0.97	(?)	1563	3.83	(?)	1.76	7.8 x 10 ¹³	8.66 x 10 ¹⁴
This Paper	1200	60	0.95	0.80	1000	1 - 10	90-99 ^(a)	3	1.3 x 10 ¹⁴	9.83 x 10 ¹⁴

(a) Collector Efficiency - 95-99..

To determine a wall erosion rate the particle currents to the first wall and their mean energies must be chosen. Based on previous design estimates, the assumed values with sputtering yields and resulting erosion rates for stainless steel are given in Table 9, where S is the sputtering yield in atoms per incident particle. A return flux for Fe ions will be dependent on total sputtering and blistering erosion and on divertor efficiencies. Ion sputtering yields are from the review by Stacey, et al.⁽³²⁾

TABLE 9. Assumed First Wall Particle Fluxes and Sputtering Yields

Particle	Mean Energy (keV)	Flux (cm ⁻² sec ⁻¹)	S	Erosion (mm·yr ⁻¹)
D	20	2×10^{13}	0.018	0.0014
T	20	2×10^{13}	0.028	0.0022
He	20	2×10^{12}	0.20	0.0015
He	~100	2×10^{11}	0.02	0.00002
n	0.1-10 ⁴	5×10^{14}	-	Negligible
n	10 ⁴	2×10^{14}	5×10^{-5}	0.00004
Fe	20	?	3.8	-
				~0.005
				$\sim 1.6 \times 10^{-11} \text{ cm} \cdot \text{s}^{-1}$

The more recent experimental values of fast neutron sputtering are for niobium, estimated by Harling, et al.⁽³⁵⁾ at

$$S_n \leq 1.1 \times 10^{-5}, \text{ (d, t) neutrons}$$

$$S_n \leq 2.5 \times 10^{-5}, \text{ (d, Be) neutrons}$$

where S_n is the number of atoms sputtered per incident neutron. This compares to a theoretical value^(36,37) of

$$S_n \sim 1.5 \times 10^{-5} \text{ atoms/neutron.}$$

Neutron sputtering for other potential first wall materials is lacking. Thomas, et al.⁽³⁸⁾ now use the estimate of

$$S_n = 5 \times 10^{-5} \text{ atoms/fast neutron}$$

for Type 316 stainless steel, vanadium and molybdenum. This value is used here as the best estimate of neutron sputtering into the vacuum system.

Blistering and exfoliation were initially projected as a serious wall erosion problem, especially for helium bombardment. The estimates of wall erosion for stainless steel in UWMAK-I assumed that new layers of blistered material continued to exfoliate under irradiation. However, more recent work has identified a number of mechanisms with the potential for reducing or eliminating blistering beyond the initial layer.

Several investigators have reported that after formation and sputtering removal of the original blister layer, no subsequent blistering is observed. The particle bombardment fluence necessary to achieve this condition is referred to as the cutoff dose. It has been shown that the cutoff dose can be reduced by decreasing ion energy or increasing target temperature. The cutoff dose can also be reduced by continuously changing the angle of incidence during bombardment to simulate an isotropic source,⁽³⁹⁾ which would more closely approximate conditions expected in a fusion reactor. It has also been shown that irradiation by a broad energy distribution of He ions can dramatically reduce subsequent blistering from He bombardment in the ≤ 10 keV range.⁽⁴⁰⁾ A majority of the recent blistering and sputtering work has been with niobium; however, the cutoff phenomenon has been observed with Type 304 stainless steel bombarded by 5 keV hydrogen ions.⁽⁴¹⁾

Based on experimental results of cutoff dose⁽²⁷⁾ and blister skin thickness⁽⁴²⁾ for niobium, an assumed blister erosion rate is calculated in Table 10. The total erosion rate estimated by blistering and sputtering can then be used to determine a high Z impurity flux to the wall, and hence an erosion rate due to self-sputtering. With an assumed divertor efficiency of 90% this flux, Φ , becomes

$$\Phi \approx (1 - 0.90)(6.02 \times 10^{23})\left(\frac{7.615}{56}\right)(1.74 \times 10^{-11})$$

$$\approx 1.4 \times 10^{11} \text{ ions} \cdot \text{cm}^{-2} \cdot \text{s}$$

The incremental erosion rate, R, due to self-sputtering is then

$$R = S\phi \frac{M}{6.02 \times 10^{23} \rho} \frac{\text{cm}}{\text{s}}$$

$$= 6.5 \times 10^{-12} \text{ cm} \cdot \text{s}^{-1}$$

where the self-sputtering ratio, atomic mass number and density for stainless steel are assumed to be

$$S = 3.8 \text{ atoms sputtered per incident atom}$$

$$M = 56$$

$$\rho = 7.615 \text{ g} \cdot \text{cm}^{-3}.$$

Erosion by self-sputtering is then assumed to be comparable to D and T ion sputtering. Total erosion of the vacuum side of the first wall is estimated to be $\sim 0.007 \text{ mm} \cdot \text{yr}^{-1}$ ($2.4 \times 10^{-11} \text{ cm} \cdot \text{s}^{-1}$) as opposed to the $0.13 \text{ mm} \cdot \text{yr}^{-1}$ ($\sim 4 \times 10^{-10} \text{ cm} \cdot \text{s}^{-1}$) assumed in UWMAK-I. The difference is largely due to the reevaluation of 14 MeV neutron sputtering and the blistering. The impurity model can now be run with the following wall erosion rates, with the lower value more likely:

$$1 \times 10^{-11} \leq R \leq 1 \times 10^{-10} \text{ cm} \cdot \text{s}^{-1}$$

Plasma confinement times, τ_p , will range from

$$1.0 \leq \tau_p < 10 \text{ sec.}$$

TABLE 10. Assumed Blistering Parameters

	Energies	
	20 keV	100 keV
Thickness (μm)	0.1	0.5
Critical Dose (cm^{-2})	1.5×10^{19}	1×10^{20}
Time to Dose (s)	7.5×10^6	5×10^8
R ($\text{cm} \cdot \text{s}^{-1}$)	1.3×10^{-12}	1×10^{-13}

Divertor and collector plate efficiencies will range from

$$0.90 \leq P_D \leq 0.99$$

$$0.95 \leq P_C \leq 0.99 \quad .$$

RESULTING VACUUM SYSTEM ACTIVITY

As expected, the predicted radionuclide activities in the plasma at the end of each burn are directly related to the divertor efficiency, plasma confinement time and wall removal rate. The activities predicted for a relatively inefficient divertor and long plasma impurity confinement time ($P_D = 0.90$, $\tau_p = 10$ s) are shown in Figure 12 for a wall removal rate of $R = 10^{-11} \text{ cm}\cdot\text{s}^{-1}$.

For the case of $R = 10^{-10} \text{ cm}\cdot\text{s}^{-1}$, the entire series of curves would shift up one order of magnitude. The total inventory of gamma emitters in the plasma at the end of each burn will then be ~ 0.2 Curies after several years of operation. With a lower wall removal rate, more efficient divertor, and shorter plasma confinement time this could go down to 2×10^{-4} Curies. Again, all of these estimates are for a wall area of 1000 m^2 .

The potential air volumes required to reduce a complete release from reactor containment to maximum permissible concentrations (MPC) are shown in Table 11. The activities are those illustrated in Figure 12 for $R = 10^{-11} \text{ cm}\cdot\text{s}^{-1}$, $\tau_p = 10$ sec after 30 years of operation. As can be seen, the air volumes potentially required are not insignificant; however, they are not large compared to the volume of the Steiner Demo containment building ($>10^4 \text{ m}^3$).⁽³⁾ Larger wall removal rates and areas above 1000 m^2 will increase the predicted inventory given; however, this is unlikely considering the trend towards compact tokamaks and efficient divertors ($P_D > 0.90$). Partial trapping of inventories in the vessel itself in the event of vacuum failure will further reduce the aerosol release. It then appears likely that dilution of radionuclides in the reactor hall will be below MPCs. Any subsequent release to the environment would be further reduced by holdup in the containment building.

FIGURE 12. Plasma Radionuclide Activities vs Time

TABLE 11. Potential Air Volumes Required for MPC

Radionuclide	Activity (Ci)	MPC ($\mu\text{Ci}\cdot\text{cm}^{-3}$)	Air Volume (m^3)
^{56}Mn	4×10^{-3}	5×10^{-7}	8×10^3
^{58}Co	3.5×10^{-3}	5×10^{-8}	7×10^4
^{54}Mn	2×10^{-3}	4×10^{-8}	5×10^4
^{60}Co	3×10^{-4}	9×10^{-9}	3.3×10^4
^{57}Co	3×10^{-3}	2×10^{-7}	1.5×10^4
^{55}Fe	1.2×10^{-2}	9×10^{-7}	1.3×10^4
^{52}V	1×10^{-3}	1×10^{-6}	1×10^3

The predicted inventory of radionuclides in the divertor collection system is very insensitive to the assumed machine parameters. This is due to the relatively long machine burn times (~ 1000 s) compared to plasma confinement times (1 to 10 s). Because plasma impurities are likely to be swept out sometime before the end of burn, the predicted inventories with $P_D = 0.99$ are only slightly larger than at $P_D = 0.90$. The only real variables of importance then are the wall activation and wall removal rates.

In Figure 13 the predicted collector inventories are shown for radionuclides of interest. In addition, the ^{55}Fe activity is shown with wall removal rates ranging from $10^{-11} \text{ cm}\cdot\text{s}^{-1} < R < 10^{-10} \text{ cm}\cdot\text{s}^{-1}$. The most likely predicted gamma activities for all radionuclides in the collector region are then $\sim 2 \times 10^5$ Curies for $R = 10^{-11} \text{ cm}\cdot\text{s}^{-1}$. Activity increases by one order of magnitude to $\sim 2 \times 10^6$ Curies for $R = 10^{-10} \text{ cm}\cdot\text{s}^{-1}$. Again this is for a wall area of 1000 m^2 .

For a liquid lithium collection system the erosion products would be transported to some cleanup system such as the zirconium foils used in UWMAK designs. Crud would also build up in pipes. Therefore, the actual collector inventory would be subject to some maintenance schedule. For example, a wall removal rate of $10^{-11} \text{ cm}\cdot\text{s}^{-1}$ would result in ~ 24 kg of erosion products per year per

FIGURE 13. Collector Radionuclide Activities vs. Time

1000 m² of wall area with the activities shown in Figure 13. The vacuum wall will eventually come into radioactive equilibrium presenting a constant source to the collector; however, the maintenance schedule may preclude equilibrium in the collector. That is, the radionuclide inventory, N, will be

$$\frac{dN'}{dt} = C - \lambda N'$$

or

$$N = C' (1 - e^{-\lambda t})$$

where C' is in fact the equilibrium collector radionuclide inventory predicted in Figure 13. For example, with a maintenance schedule of one year, important radionuclides would have the following fraction of predicted collector equilibrium activity:

$$^{60}\text{Co} - 0.123$$

$$^{55}\text{Fe} - 0.226$$

$$^{54}\text{Mn} - 0.555$$

$$^{57}\text{Co} - 0.607$$

$$^{58}\text{Co} - 0.970$$

Replacement of the first wall structures every five years, as in the Steiner Demo, would further reduce the inventory of the longer-lived radionuclides delivered to the collector system, since these radionuclides would not yet have reached equilibrium activity in the stainless steel first wall.

CONCLUSIONS

The erosion rate of a stainless steel first wall in a compact tokamak fusion reactor has been reviewed, based on recent experimental results in materials radiation damage. The high energy 14 MeV neutron fluence to the wall is no longer seen as a major source of erosion, with the best estimates of sputtering yield now put at $S_n \approx 5 \times 10^{-5}$ atoms/neutron. In addition, a number of mechanisms have been identified to reduce or eliminate surface blistering due to ion (He, D, T) bombardment. With a set of first wall particle fluxes based

on previous reactor designs, the wall erosion on the vacuum side is now estimated at $10^{-11} \text{ cm}\cdot\text{s}^{-1} < R < 10^{-10} \text{ cm}\cdot\text{s}^{-1}$ with values near $10^{-11} \text{ cm}\cdot\text{s}^{-1}$ being more likely. This compares to an estimate of $\sim 4 \times 10^{-10} \text{ cm}\cdot\text{s}^{-1}$ used in the UWMAK-I study.

As for the subsequent transport and deposition of activation products in the vacuum system, it is suggested that the divertor collection system and exhaust manifolds can be designed to effectively eliminate the transport of high Z gamma emitters into the mechanical vacuum pump system. For the high Z elements in stainless steel, no compound and/or phase has been identified which would allow transport deep into the exhaust manifolds towards the diffusion pumps, roots pumps, etc. Metal walls and collection surfaces are expected to provide for excellent sputter deposition in the divertor region or in the main torus vacuum vessel itself. High Z stainless steel impurities still in the plasma at the end of the burn are expected to dump to the first wall and deposit when magnetic confinement is lost. Experiments to identify any bulk or elemental redistribution of material around the torus have yet to be performed.

With vacuum deposition, eroded material swept from the plasma by divertor action and permanently entrained on a collector system in the divertor region will no longer be available for accidental release, and will not be in a location to expose workers during routine maintenance. The radiation hazards presented by a flowing liquid lithium system which subsequently transports activated erosion products out of the divertor region to some impurity cleanup system will be design dependent and must be evaluated individually. For a wall area of 1000 m^2 the activity is expected to reach $\sim 2 \times 10^5$ Curies after several years of operation. This is for a wall removal rate of $10^{-11} \text{ cm}\cdot\text{s}^{-1}$. Collector inventories will be very insensitive to divertor efficiencies ($>90\%$) and impurity confinement times ($\sim 10 \text{ s}$) when long machine burns ($\sim 1000 \text{ s}$) are considered. Predicted inventories are directly related to wall removal rate, with $10^{-10} \text{ cm}\cdot\text{s}^{-1}$ proposed as an upper bound giving $\sim 2 \times 10^6$ Curies gamma activity in equilibrium.

In the case of vacuum vessel failure, the maximum gamma-emitting radionuclide inventory in the plasma potentially available for release was found to be ~ 0.2 Curies for a wall area of 1000 m^2 . This inventory easily reduces

several orders of magnitude for low wall erosion rates and high efficiency (~99%) divertors. In the event of a catastrophic failure of the vacuum vessel, this potential source term would likely be reduced further by the initial influx of air into the vessel with some subsequent deposition in the vessel. More work will be required to identify actual release values and transport mechanisms into the reactor containment building; however, it is likely that in the event of release, airborne radionuclide concentrations in the reactor hall will be below MPCs. It is suggested that further holdup by the containment building would further reduce releases to the environment to insignificant levels.

VI. RELEASE OF ACTIVATION PRODUCTS IN LITHIUM FIRES

Some form of lithium is required in all fusion reactors which breed tritium. It may be present as a solid compound or liquid metal, either of natural isotopic abundance or enriched in ^6Li to take advantage of the large (940 barns) thermal neutron cross-section for tritium production ($^6\text{Li}(n,\alpha)^3\text{H}$). Thermal properties of lithium have led to its consideration as the primary coolant in a number of reactor design studies. In the reference design considered here, the blanket vessel is filled with liquid lithium. Heat generated by neutron reactions in the lithium and steel structures is transferred out of the blanket by high-pressure helium flowing through steel coolant pipes. The liquid lithium metal is circulated slowly to extract tritium.

Liquid lithium reacts vigorously with water and readily with air or concrete. Its chemical activity, similar to that of liquid sodium, has caused some concern over the consequences of accidental spills. The liquid lithium inventory is expected to be around 100 tons. This could present a significant fire hazard in the reactor building in the event of lithium release.

This study considers the possibility of such a release as a result of a lithium fire within a magnetic fusion reactor containment vessel. Descriptions of lithium reactions that could occur in the fusion reactor are followed by discussions of the effects of adiabatic flame temperatures, radiation and vapor pressures of neutron activation products and their compounds on lithium reactions. There is little experimental evidence in this area and much of what is reported is based on predicted behavior. All conditions and constraints utilized in the determination of the predicted results are explicitly expressed in the text.

LITHIUM PROPERTIES AND REACTIONS

Lithium is one of the most reactive elements known, yet good reliable information concerning its chemical behavior is often lacking. The most comprehensive compilation of its physical properties and interactions with other

materials is a document authored by Ballif, et al.⁽⁴³⁾ For the purposes of this study the chemical reactions are of predominant interest.

Table 12 is a partial list of the known reactions of lithium, their enthalpies and free energies at 25°C.⁽⁴⁴⁾

Lithium reacts with all known molecular gases. Reaction of lithium with air may be considered to consist mainly of reaction with nitrogen, oxygen, and carbon dioxide. These reactions and their products are included in Table 12.

TABLE 12. Lithium Reactions, Enthalpies, and Free Energies⁽⁴⁵⁾

Reactions	Enthalpy $\Delta H^\circ_{25^\circ\text{C}}$ (kcal/mole)	Free Energy $\Delta G^\circ_{25^\circ\text{C}}$ (kcal/mole)
$2 \text{ Li(c)} + 1/2 \text{ O}_2\text{(g)} \rightarrow \text{Li}_2\text{O(c)}$	-142.650	-133.950
$2 \text{ Li(c)} + \text{O}_2\text{(g)} \rightarrow \text{Li}_2\text{O}_2\text{(c)}$	-151.9	-138.1
$\text{Li(c)} + 1/2 \text{ H}_2\text{(g)} + 1/2 \text{ O}_2\text{(g)} \rightarrow \text{LiOH(c)}$	-116.589	-105.676
$\text{Li(c)} + 3/2 \text{ H}_2\text{(g)} + \text{O}_2\text{(g)} \rightarrow \text{LiOH} \cdot \text{H}_2\text{O(c)}$	-188.926	-163.437
$\text{Li(c)} + \text{H}_2\text{O(l)} \rightarrow \text{LiOH(c)} + 1/2 \text{ H}_2\text{(g)}$	- 48.7	- 48.99
$3 \text{ Li(c)} + 1/2 \text{ N}_2\text{(g)} \rightarrow \text{Li}_3\text{N(c)}$	- 47.500	- 37.300
$\text{Li(c)} + 1/2 \text{ H}_2\text{(g)} \rightarrow \text{LiH(c)}$	- 21.61	- 16.72
$2 \text{ Li(c)} + 3/2 \text{ CO}_2\text{(g)} \rightarrow \text{Li}_2\text{CO}_3\text{(c)} + 1/2 \text{ C(c)}$	-148.6	-128.4
$2 \text{ Li(c)} + 3\text{CO(g)} \rightarrow \text{Li}_2\text{CO}_3\text{(c)} + 2\text{C(c)}$	-210.45	-171.38
$2 \text{ Li(c)} + 2\text{C(c)} \rightarrow \text{Li}_2\text{C}_2\text{(c)}$	- 14.2	--
$2 \text{ Li(c)} + \text{Mo(c)} + 2\text{O}_2\text{(g)} \rightarrow \text{Li}_2\text{MoO}_4\text{(c)}$	-365.25	--
$\text{Li(c)} + \text{Al(c)} + 2\text{H}_2\text{(g)} \rightarrow \text{LiAlH}_4\text{(c)}$	- 24.67	--
$\text{Li(c)} + \text{Sn(c)} \rightarrow \text{LiSn(c)}$	- 16.8	--
$\text{Li(c)} + \text{Pb(c)} \rightarrow \text{LiPb(c)}$	- 14.6	--
$3 \text{ Li(c)} + 2\text{Sb (c)} \rightarrow \text{Li}_3\text{Sb}_2\text{(c)}$	- 43.5	--

Lithium-Nitrogen Reaction

Reaction of lithium and nitrogen occurs at room temperature with catalytic amounts of water present. Observation of the reaction of molten lithium and dry nitrogen indicates that the reaction rate is strongly temperature dependent. The free energy of formation of lithium nitride has been calculated as a function of temperature,⁽⁴⁴⁾ and these results are given in Table 13. Basically these results indicate that the lithium-nitrogen reaction becomes less thermodynamically favorable with increasing temperature.

TABLE 13. Lithium Nitride - Free Energy of Formation

<u>Temperature (°C)</u>	<u>ΔG° (kcal/mole)</u>
25	-37.3
527	-20.1
1227	+ 3.8

Predicted adiabatic flame temperatures of lithium-air or lithium-concrete reactions exceed the temperature where the lithium-nitrogen reaction is thermodynamically favorable. This does not indicate, however, that a lithium-nitrogen reaction will not occur in a lithium fire. The lithium-nitrogen reaction probably would not occur at the flame front in a lithium-air fire. However, nitrogen diffusion through this zone into the cooler molten lithium pool should allow rapid formation of lithium nitride.

Lithium nitride is a very reactive material, and it has been stated⁽⁴³⁾ that no metal or ceramic material has been found to be resistant to attack by the molten compound.

Lithium-Oxygen Reaction

Lithium reacts with oxygen to form lithium oxide, Li_2O , under most conditions, although the peroxide species, Li_2O_2 , is slightly favored thermodynamically. The ignition temperature of lithium in oxygen has been cited as 630°C ⁽⁴⁵⁾ although this is probably not an exact figure. Reactions of lithium

appear to be a complex function of moisture content, impurity concentrations, diffusion rates, surface to mass ratio of lithium and other undetermined factors. The significant fact is that the lithium-oxygen reaction is thermodynamically favorable at both the temperatures of a lithium spill and the calculated adiabatic flame temperatures.

It is expected that the lithium-oxygen reaction will predominate to the extent of the limiting reagent in any lithium-air reaction envisioned. Once formed, lithium oxide is a very reactive compound and may be expected to attack most refractory materials.⁽⁴³⁾

Lithium-Carbon Dioxide Reaction

Lithium reacts with carbon dioxide, as shown in Table 12, to form lithium carbonate, Li_2CO_3 . As in the case of sodium, carbon dioxide will also react with the alkali metal oxide forming the carbonate species to the extent of the limiting reagent of the reaction.

Lithium-Concrete Reaction

Reaction of lithium with concrete is essentially a lithium-water reaction, the water being present in the concrete as water of hydration. Lithium can also react with other components of concrete but such reactions are not as thermodynamically favorable as the lithium-water reaction.

Predictions of most lithium-concrete interactions have been based on the more available sodium-concrete reaction data. There does exist, however, some recent experimental evidence on lithium-concrete reactions.⁽⁴³⁾ This evidence indicates lithium reduces silicates to silicides, and that the quantity of heat released is a function of the type of concrete. Definitive studies of lithium-concrete reactions do not appear to be available.

Lithium-Water Reaction

Lithium reacts with water to form lithium hydroxide and hydrogen in the presence of both air and nitrogen atmospheres. It has been reported⁽⁴⁶⁾ that the lithium-water reaction occurs preferentially with respect to the lithium-nitrogen reaction. In a fire involving both lithium-air and lithium-concrete reactions, reaction with concrete will continue even if all the available atmospheric oxygen is consumed.

The release of hydrogen as a product of the lithium-water reaction may lead to further hydrogen-metal reactions or to hydrogen combustion which would increase the overall flame temperature.

Lithium hydroxide is reported to be extremely corrosive, attacking all known metals or refractory materials.⁽⁴³⁾

Lithium-Hydrogen Reaction

Molten lithium reacts with hydrogen to form lithium hydride over the approximate temperature range of 400 to 710°C.⁽⁴⁷⁾ Lithium hydride is a relatively stable compound which can be melted (m.p. = 688°C) without decomposition. Lithium hydride reduces oxides readily and reacts with both metals and ceramics at high temperatures.⁽⁴³⁾

Lithium Reactions With Metals and Refractory Materials^(43-44,48-51)

Molten lithium may be expected to attack almost all known ceramics and insulating materials. The mechanism of attack is formation of the relatively stable lithium oxide, nitride or carbide species. Molten lithium may be expected to reduce most metal oxides and to react with the less thermodynamically stable carbides and silicides.

The mechanism for reaction of lithium with most metals and refractory alloys is intergranular attack on precipitated nitrides, carbides or silicides. The resistance to attack by lithium is a function of elemental composition, impurity concentration, prior heat treatment of the material and temperature. An increase in the temperature results in increased attack by lithium.

The result of molten lithium attack on refractory metals or alloys that are candidates for the first wall may therefore be disintegration of structural integrity and production of a high surface area material.

ADIABATIC FLAME TEMPERATURES

Calculated adiabatic flame temperatures of both lithium-air and lithium-concrete reactions have been reported by several authors.⁽⁵¹⁻⁵³⁾ Various assumptions and constraints were chosen and utilized in these calculations, including selection of reactant and product species, pressure, reaction

kinetics, equilibrium processes, and overall adiabatic behavior. Assumption of overall adiabatic behavior is reasonable and produces calculated temperatures which should exceed those which would actually occur. Heat capacities of structural materials have, in general, been neglected; their inclusion will lower predicted temperatures.

The significant points made by these authors are:

- Both lithium-air and lithium-concrete reactions produce flame temperatures of approximately 2400 to 2500K.
- The predicted temperatures are almost identical regardless of the individual assumptions and constraints imposed.
- A significant concentration of extremely reactive atomic oxygen is produced -- approximately 10^{-3} atm for the lithium-air reaction^(52,53) and approximately 10^{-4} atm for the lithium-concrete reaction.⁽⁵³⁾

These calculated adiabatic flame temperatures are dominated by reactions with oxygen and water for the lithium-air and lithium-concrete reactions, respectively. Release mechanisms of neutron activation products, however, depend not only upon the temperatures achieved in a fire but also upon accurate knowledge of reaction products, their concentration, and subsequent reaction pathways. This information is presently not available but should be a major concern and requires further investigation.

One additional aspect which seems to be often neglected or even incorrectly interpreted is the vapor pressure of neutron activation products at the temperatures expected to occur. That temperatures are below the melting or boiling points for materials of concern do not preclude volatile release of neutron activation products.

RADIATION EFFECTS

Previous studies have neglected the effects the high radiation field caused by neutron activation products may have on the chemical reactions possible in a lithium spill and accompanying fire. The question of whether the radiation field contributes to the possible reactions must be examined. Potential reactions of importance - assuming air is present - include:

The G value (molecules produced per 100 MeV energy absorbed) is about 8 for reaction (1) and 4-5 for reaction (2).⁽⁵⁴⁾ Reactions (3 through 5) are only illustrative of the many oxidation reactions that are possible. The net result of reactions (4) and (5), if water is present, is the formation of nitric acid. A primary concern is the reaction of various metals with the nitric acid or with atomic oxygen. Thus, it becomes important to estimate the yields from radiolysis.

The total activity of the reference blanket after five years' operation is about 2×10^9 Ci. The energy released as gamma radiation after shutdown is about 10^{26} eV s⁻¹. Much of this energy will be absorbed in the blanket and in construction materials. For a very crude estimate we will assume that half of the energy is absorbed in the containment building air, 10^5 m³ at 300K. This means that 10^{25} eV is assumed to be absorbed in O₂ and 4×10^{25} eV in N₂. Based on G values of 8 molecules per 100 eV absorbed in O₂ and 5 in N₂, the production of active atoms is 3×10^{24} atoms·s⁻¹, resulting in a concentration of atomic oxygen and nitrogen of about 3×10^{13} atoms·cm⁻³·s⁻¹.

The contribution from β energy is more difficult to estimate since both the amount escaping the stainless steel and the amount absorbed by the air depend on the energy of the particles, which varies from zero to the maximum β energy. For a rough approximation we will assume that for steel (maximum range ~1 mm) half of the β energy of 10^{17} eV cm⁻³ escapes from the outer 0.1 mm layer and is absorbed by the first centimeter of air. This is 5×10^{14} eV cm⁻³ s⁻¹ or approximately 20 times the estimate for γ energy. However, the β effect occurs only in the surface layer of air. Although these estimates are very crude, it seems likely that the maximum production rate of atomic species and hence other active molecules is on the order of 10^{12} to 10^{13} molecules cm⁻³ s⁻¹.

Typical second-order rate constants for atomic oxygen are of the order of $10^{-11} \text{ cm}^3 \cdot (\text{molecule} \cdot \text{s})^{-1}$.⁽⁵⁵⁾ Thus for reactions with a gas at any significant pressure, the half-life is less than a microsecond.

When one notes that the concentration of oxygen molecules in 1 atm air is $10^{18} \text{ moles} \cdot \text{cm}^{-3}$, the production of 10^{13} atoms per second of a species with a half-life of one microsecond does not appear to be significant in an accident situation. However, over a long period of time, the effect of secondary reactions, e.g., corrosion, arising from these active species could be important.

VAPOR PRESSURES OF NEUTRON ACTIVATION PRODUCTS AND THEIR COMPOUNDS

Volatile release of neutron activation products, or their compounds, at the temperatures expected in a lithium-air or lithium-concrete fire is the primary release mechanism of concern. Accurate experimental vapor pressure data for such an analysis are often lacking, but Tables 14 and 15 present data selected for use. The sources of such information are cited, but the values reported should be viewed as approximate in most cases.

TABLE 14. Selected Elements - Melting Points, Boiling Points, and Vapor Pressures

Element (Symbol)	Melting Point ⁽⁵⁵⁾ (K)	Boiling Point ⁽⁵⁵⁾ (K)	Vapor Pressure ⁽⁵⁶⁾ T = 2500K (torr)
Al	933	2793	201
Co	1765	3203	24 ^(a)
Cr	2133	2953	69
Cu	1356	2833	101
Fe	1809	3133	12
Mn	1517	2333	(b)
Mo	2888	4883	4×10^{-4}
Nb	2740	5013	1×10^{-4}
Ni	1728	3188	14
Pb	600	2023	(b)
Ti	1940	3558	2
V	2175	3683	1
Zr	2125	4673	2×10^{-3}

(a) Required extrapolation beyond temperature range cited.
(b) Flame temperature exceeds normal boiling point.

TABLE 15. Selected Oxides, Nitrides and Carbides - Melting Points, Boiling Points, and Vapor Pressures

Compound (Formula)	Melting Point ⁽⁵⁵⁾ (K)	Boiling Point ⁽⁵⁵⁾ (K)	Vapor Pressure ⁽⁵⁷⁾ T = 2500K (torr)
Al ₂ O ₃	2318	3253	2 ^(a)
AlN	>2473 in N ₂	2273 ^(b)	1 at 1760K
Al ₄ C ₃	Stable to 1673	2473 ^(c)	-
CoO	2208	-	-
Co ₂ O ₃	1168 ^(c)	-	-
Co ₃ O ₄	Transforms to CoO, 1173-1223	-	-
CrO ₂	Loses O, 573	-	-
CrO	-	-	-
Cr ₂ O ₃	2708	4273	-
CrO ₃	469	Decomposes	-
CrN	1973 ^(b)	-	1 at 992K
Cr ₃ C ₂	2163	4073	2.47 x 10 ⁻⁵ at 2073K
Cu ₂ O	1508	Loses O, 2073	-
CuO	1599	-	-
Cu ₃ N	573 ^(b)	-	-
FeO	1693	-	-
Fe ₂ O ₃	1838	-	-
Fe ₂ N or Fe ₄ N ₂	473 ^(b)	-	-
Fe ₃ C	2110	-	-
Mn ₃ O ₄	1978	-	-
MnO ₂	Loses O, 808	-	-
Mn ₂ O ₇	279	328 ^(b)	-
MnO	-	-	-
Mn ₂ O ₃	(Questionable Existence)	-	-
Mn ₃ C	-	-	-
MoO ₂	-	-	-
Mo ₂ O ₅	-	-	-
Mo ₂ O ₃	-	-	-
MoO ₃	1068	1428 ^(a)	-

TABLE 15. Selected Oxides, Nitrides and Carbides - Melting Points, Boiling Points, and Vapor Pressures (cont'd)

Compound (Formula)	Melting Point ⁽⁵⁵⁾ (K)	Boiling Point ⁽⁵⁵⁾ (K)	Vapor Pressure ⁽⁵⁷⁾ T = 2500K (torr)
Mo ₂ N	-	-	1 at 592K
NbO ₂	-	-	2 ^(a)
NbO or Nb ₂ O ₂	-	-	-
Nb ₂ O ₅	1733	-	-
Nb ₂ O ₃	2053	-	-
NbN	2846	-	-
NbC	3773	-	1 x 10 ^{-4(a)}
NiO	2263	-	759 ^(a)
NiC	-	-	-
PbO ₂	563 ^(b)	-	-
PbO	1161	-	>760 ^(a)
Pb ₃ O ₄	773 ^(c)	-	-
Pb ₂ O ₃	643 ^(c)	-	-
Pb ₂ O	-	-	-
TiO ₂	2103-2123	2773-3273	3 x 10 ^{-1(a)}
TiO	2023	>3273	-
Ti ₂ O ₃	2403 ^(c)	-	-
TiN	3203	-	1
TiC	3413 ± 90	5093	8 x 10 ^{-4(a)}
VO or V ₂ O ₂	Ignites	-	-
VO ₂	2240	-	-
V ₂ O ₅	963	-	-
V ₂ O ₃	2243	-	-
VN	2593	-	-
VC	3083	4173	2 x 10 ⁻²
ZrO ₂	2973	5273	2 x 10 ⁻⁴
ZrN	3253 ± 50	-	4 x 10 ⁻²
ZrC	3813	5373	~2 x 10 ⁻⁵

(a) Required extrapolation beyond temperature range cited.

(b) Sublimates.

(c) Decomposes.

A flame temperature of 2500K was utilized in all cases. This occasionally required extrapolation of existing vapor pressure information beyond the reported temperature range.

The vapor pressure in mm Hg (torr) may be represented by an equation of the type

$$\log P = -\frac{A}{T} + B + C \log T + 10^{-3} DT$$

where A, B, C, and D are constants for a specified temperature range, and T is the Kelvin temperature of interest.

Table 14 presents the melting and boiling points and vapor pressures at 2500K of selected pure elements. Melting and boiling points were obtained from Reference 55 while the required constants for calculation of the vapor pressures were obtained from Reference 56.

Table 15 gives the melting points, boiling points, and calculated vapor pressures at 2500K for those oxides, nitrides, and carbides of the elements listed in Table 14 for which information was available.

Phosphides, sulfides, and silicides were omitted because their probability of occurrence is low and available information is lacking. Hydrides will not be stable at the flame temperature and will dissociate into the respective metal and hydrogen.

Comparison of calculated elemental vapor pressures obtained from data in Reference 57 often showed large discrepancies with respect to that obtained from Reference 56. Therefore, calculated vapor pressures of the compounds reported in Table 15 should be regarded as only approximate without further confirmation as to their accuracy.

Table 14 indicates that only molybdenum, niobium, and zirconium have vapor pressures of less than 1 torr at the expected flame temperature of 2500K. Titanium and vanadium have calculated vapor pressures at 2500K in the 1 to 10 torr range, while cobalt, chromium, iron, and nickel have vapor pressures of 10 to 100 torr. The calculated vapor pressures of aluminum and copper exceed 100 torr at 2500K, while this temperature exceeds the normal boiling point of both lead and manganese.

Table 15 shows that only a few compounds have a significant vapor pressure. Compounds whose vapor pressures exceed those of the elemental metal are: MoO_3 , NbO_2 , and NiO . Thus, with these exceptions, the major concern for volatile release of neutron activation products is the vapor pressures of the elements. The contribution to this volatilization through thermal decomposition of oxides and other compounds has not been considered.

LITHIUM ACCIDENT CONDITIONS

We have discussed the possible reactions of lithium resulting from a lithium spill, the predicted flame temperatures of both lithium-air and lithium-concrete fires, and the vapor pressures of selected elements and compounds of concern. These results are now applied to three specific atmospheric conditions: 1) when the volume inside the containment vessel is a vacuum, 2) when the reaction is confined to the interior of a containment building, and 3) when the containment is breached so that air may be exchanged between the reactor and the external environment.

1. If the volume inside the containment vessel is under vacuum, then we are limited to a lithium-concrete fire. The mechanism of initiating this fire is not examined, but we assume it exists. The flame temperature is essentially the same as that of the lithium-air fire but is expected to proceed with a slower consumption of lithium. This slower rate of consumption is due to both a limited surface area of the lithium-concrete interface and the requirement that dehydration of the concrete must occur to obtain the necessary reactant, water.

The total system should be expected to be very reducing due to the amount of lithium present and the exclusion of air or oxygen from the reaction zone. The temperatures reached, however, are sufficient to cause considerable transport of neutron activation products as is seen by the vapor pressures reported in Table 14. This transport may be aided significantly by chemical attack of both lithium and lithium hydroxide on structural materials, destroying their structural integrity and producing a high surface area material. Even in the event that the vapor pressures are low, chemical attack may produce particles that can be transported by the "smoke" of the lithium fire.

2. If the containment building is assumed to be in one atmosphere of air, we still have a limiting reagent situation - the quantities of oxygen and nitrogen are less than that of lithium. Again, the total system would be expected to be a reducing one. Chemical attack again is a major concern, especially with the probable occurrence of molten lithium nitride. The mechanisms of release are expected to be the same as in the previous case. In addition, one must not ignore that it is probable that lithium-concrete as well as lithium-air reactions will occur in this situation.

3. When the containment vessel is breached and free exchange of air between the containment vessel and the environment occurs, we have our "worst case". In this event, unlimited air (oxygen and nitrogen) is available, and the maximum consumption of lithium will occur. This implies the flame temperature will exist over a longer time period than in the previous two cases, and that the environment is accessible to volatile activation products, transported either as vapor or condensed in the "smoke."

SUMMARY

Release of neutron activation products as a result of a lithium fire is possible. The vapor pressures of elements of concern at the expected flame temperatures are predicted to be a major mechanism of release. Contributing to this effect are the expected chemical reactions of lithium and its reaction products on structural materials.

VII. COMPONENT ACTIVITY AFTER SHUTDOWN

The radioactivity in a fusion reactor remains high long after reactor shutdown. In the first wall, for example, the specific activity remains above $1 \text{ Ci}\cdot\text{cm}^{-3}$ for about five years after shutdown. This activity presents a number of potential hazards to plant personnel and the general public. The possibility of radioactive releases as a result of accidents such as lithium fires or coolant failures still exists, just as during operation. Afterheat and dose rates remain high long after shutdown. The risks of personnel and public exposure actually increase after shutdown because of maintenance operations and component transportation.

The total blanket activity is shown in Figure 14 for times after reactor shutdown. It is seen that the decay is rather slow--from a total activity of $2.06 \times 10^9 \text{ Ci}$ at shutdown following five years' operation, the blanket takes 52 days to decay to 10^9 Ci , 15 months to decay $5 \times 10^8 \text{ Ci}$, and about 10 years to decay to 10^8 Ci . The key nuclides contributing to this activity can be seen in Figure 15, which shows the first wall activity as a function of time, by nuclide. The first wall activity is dominated by ^{55}Fe , ^{58}Co , ^{57}Co , ^{51}Cr , and ^{54}Mn for the first few years. The same is true of the outer blanket, except that ^{56}Mn is more significant for the first few hours, the differences between regions being primarily a result of the neutron energy spectrum shown in Figure 4. The thermal flux in the outer blanket is about four orders of magnitude larger than in the first wall or inner blanket. Therefore, in the case of ^{56}Mn the thermal $^{55}\text{Mn} (n,\gamma) ^{56}\text{Mn}$ reaction dominates over the fast neutron-induced $^{56}\text{Fe} (n,p) ^{56}\text{Mn}$ reaction (see Table 3) in the outer blanket. The activities of other components, such as neutral beam injectors and plasma divertors, decay from their shutdown value in a similar manner. If these components have longer lifetimes than the first wall, then the dominance of ^{55}Fe is more pronounced. Iron-55, however, decays by electron capture and the internal bremsstrahlung and x-rays are soft. Hence, they were not considered in the dose calculations presented herein.

FIGURE 14. Blanket Activity After Shutdown Following 5 Years of Operation

FIGURE 15. First Wall Activity After Shutdown

DOSE RATES

Maintenance operations need to be carefully examined for potential hazards. Maintenance activities include both scheduled maintenance, such as removal and replacement of the first wall or neutral beam injectors at the end of their useful life, and unscheduled maintenance, such as repair of damaged or defective equipment. In current nuclear power plants, unscheduled maintenance accounts for over 50% of total down time. A fusion power plant is expected to be an even more complex engineering structure with at least an equivalent rate of component failure.

It is instructive to investigate plant dose rates with some background information on allowable exposures. The maximum exposure to plant personnel permitted by current United States Nuclear Regulatory Commission⁽⁵⁸⁾ regulations is:

	Quarterly Limit (rem)
Whole body	1.25
Skin of whole body	7.5
Hands and forearms, feet and ankles	18.75

To ensure compliance with these limits, the maximum dose rate in which an unshielded worker, such as a welder, can work is about 15 rem/hr. At this rate the worker would exceed his quarterly permissible dose after 5 minutes. This includes time to enter and exit the area and set up and remove his equipment. Further, it cannot be assumed that an infinite pool of trained personnel exists to perform reactor maintenance on such complex machinery. Therefore, a more realistic upper limit to the acceptable dose rate is about 0.1 rem/hr. Personnel would be allowed to work at this exposure rate for one week per year, providing the quarterly limit is not exceeded.

With this background, it is obvious that dose rates around the reactor are much too high to permit contact maintenance on many components for years after shutdown. Remote maintenance techniques will be essential to the operation and maintenance of any tokamak which uses steel structures. Contact dose rates at the blanket shield are shown in Figure 16. This is the dose rate to which any unshielded worker performing maintenance on the shield would be

FIGURE 16. Estimated Contact Dose Rates at the Blanket Shield

exposed. Even after five years this dose is still three orders of magnitude above the maximum unshielded working limit. The reference design calls for replacing the blanket modules every five years, which requires removal of the biological shield for the segment to be replaced.

The contact dose rate at the first wall, shown in Figure 17, is even higher than at the shield. Starting from a value of $5.3 \times 10^6 \text{ rem}\cdot\text{hr}^{-1}$ immediately following shutdown after five years' operation, the dose rate falls to $2.0 \times 10^6 \text{ rem}\cdot\text{hr}^{-1}$ after one day, $1.6 \times 10^6 \text{ rem}\cdot\text{hr}^{-1}$ after one month, and $8.3 \times 10^4 \text{ rem}\cdot\text{hr}^{-1}$ after five years. Even after five years the dose rate is much too high to allow contact maintenance, even with extensive removable personnel shielding.

Considering the key nuclides contributing to the dose rates, the first wall and shield show similar patterns. For the first few hours, most of the dose is due to ^{56}Mn decay. Following ^{56}Mn decay, the ^{58}Co dominates for several months. After five years, ^{60}Co is by far the major contributor to the dose rate, followed by ^{54}Mn and, to a much lesser extent, ^{57}Co . Long-lived activity (>10 years), however, may be determined by activation of trace elements not considered in this study which produce very long-lived isotopes (e.g., ^{94}Nb).⁽⁵⁹⁾

The dose rate of the neutral beam injector has been calculated after ten years' operation. This is shown in Figure 18. The dose rates shown are those at the injector at time of replacement, assuming a ten-year lifetime. No shielding of the injector has been considered in calculating the dose rate. At shutdown, the dose rate is $4.9 \times 10^5 \text{ rem}\cdot\text{hr}^{-1}$, after one month it is $1.7 \times 10^5 \text{ rem}\cdot\text{hr}^{-1}$, and after five years it is still $1.1 \times 10^4 \text{ rem}\cdot\text{hr}^{-1}$.

An estimate has been made of the contact dose rate at the collector plate of the plasma divertor. For this estimate, the plate was assumed to be a 3-cm thick plate of stainless steel, exposed to the neutron spectrum of the inner blanket. First wall materials collected were assumed to deposit on a thin surface layer. The calculated dose rates are shown in Figure 19. Out of $4.9 \times 10^7 \text{ rem}\cdot\text{hr}^{-1}$ at shutdown following five years' operation, $3.2 \times 10^7 \text{ rem}\cdot\text{hr}^{-1}$ are due to first wall material that is deposited. This result is not surprising

FIGURE 17. Estimated Contact Dose Rate at the First Wall

FIGURE 18. Estimated Contact Dose Rate at the Neutral Beam Injector after 10 Years Operation

FIGURE 19. Estimated Dose Rate at the Divertor

when one considers that the sputtered first wall materials deposit with a high packing fraction in a thin layer on a collector surface area which is much smaller than the first wall area. After five years the contact dose rate at the collector plate is still $1.1 \times 10^4 \text{ rem}\cdot\text{hr}^{-1}$.

DECAY HEAT

Activation products may present another hazard to safe reactor operation besides radiation. Energy release through radioactive decay may cause key reactor materials to heat up, possibly compromising the materials integrity of systems vital to reactor safety. The activation product concentrations were treated as a beta and gamma source to assess the blanket heating rate. Data were used from Table of Isotopes⁽⁶⁰⁾ to describe the beta and gamma radiation emitted with each radionuclide.

The gamma production was separated into 21 energy groups and the heating rate in the blanket was calculated with the ANISN computer code. For β decay, it was assumed that two-thirds of the Q value of the reaction is shared by the β^- and γ rays, with the remainder being carried off by neutrinos. The β^- energy is deposited at the production site. For electron capture processes it was assumed that the entire Q value is removed by neutrinos. For β^+ decay it was assumed that only two 0.511 MeV photons are produced. The total afterheat at shutdown after five years of full power operation is 23.9 MW, of which 3.8 MW are due to beta decay and 20.1 MW due to gamma decay. One day after shutdown the total blanket afterheat is 3.37 MW, of which 0.57 MW is due to beta decay and 2.80 MW is due to gamma decay. The beta and gamma heating rates by zone are tabulated in Table 16.

An estimate was made of the temperature rise of the blanket due to radioactive afterheat. Each zone was considered with a uniform heating source. Heating was assumed to be adiabatic, with no coolant flow. This model is very conservative, totally neglecting the effects of thermal conduction and radiation. Nevertheless, it provides a lower limit on how soon blanket damage might occur following a loss of coolant flow accident after shutdown.

The rate of temperature rise due to adiabatic heating is

$$\frac{dT}{dt} = \frac{Q}{\rho c_p}$$

TABLE 16. Decay Heat Production Rate

Zone	Time After Shutdown Following 5 Years Operation	
	0	1 Day
3. First Wall	0.211	0.0308
5. Inner Blanket	0.053	0.0078
6. Inner Reflector	0.046	0.0067
7. Middle Blanket	0.025	0.0036
8. Outer Reflector	0.019	0.0028
9. Outer Blanket	0.078	0.0115
10. Shield	0.045	0.0067
Total	23.9 MW	3.37 MW

where Q is the volumetric heating rate, ρ is the mass density of the zone, and c_p is the average specific heat. Exponentially-decreasing decay heating was assumed. Using the data of Table 16, the time constant τ during the first day after shutdown is 12.25 hr. The temperature rise in time t during a loss of flow is then:

$$\Delta T = \frac{Q_0 \tau}{\rho c_p} (1 - e^{-t/\tau}),$$

where Q_0 is the heating rate at shutdown. For the first wall the initial temperature was assumed to be 400°C, and the time required to heat the wall to 750°C was calculated. Since the thermal properties of stainless steel deteriorate rapidly above this temperature, it was taken as a damage threshold. Using an initial heating rate $Q_0 = 0.211 \text{ W/cm}^3$, a smear density $\rho = 1.21 \text{ g}\cdot\text{cm}^{-3}$, and a specific heat $c_p = 0.50 \text{ J}\cdot(\text{g}\cdot\text{deg})^{-1}$, the time required for the first wall to heat up to 750°C is $t = 16.9 \text{ min}$. Therefore, some active cooling of the

first wall after shutdown may be desirable, at least for the first few hours, but afterheat does not appear to be a major problem in terms of material damage, especially when thermal conduction and radiation are considered.

The rest of the blanket was assumed to have an initial temperature of 500°C, and 750°C was taken as the damage threshold. The results are summarized in Table 17, where t is the time to heat up to the threshold. It appears that afterheat does not pose a safety problem anywhere in the blanket away from the first wall.

TABLE 17. Time Required for Adiabatic Heating to 750°C Following Shutdown

Zone	$Q_0 \cdot 10^{-3}$ W·cm ⁻³	$\rho \cdot 10^{-3}$ g·cm ⁻³	C_p J·(g-deg) ⁻¹	t (min)
3. First Wall	0.211	1.21	0.50	17
5. Inner Blanket	0.053	0.534	2.45	111
6. Inner Reflector	0.046	2.25	1.67	457
7. Middle Blanket	0.025	1.11	1.64	391
8. Outer Reflector	0.019	2.25	1.67	∞
9. Outer Blanket	0.078	5.79	0.55	193
10. Shield	0.045	7.62	0.50	481

VIII. DIRECTIONS FOR FURTHER RESEARCH

The objective of this study has been to identify key activation products which are likely to have an impact on safe magnetic fusion reactor operation. Since stainless steel is almost certain to be the primary structural material at least in early experimental fusion reactors, the analysis has focused on activation of stainless steel structures in a compact tokamak. This analysis is one necessary part of an assessment of the safety acceptability of stainless steel as a structural material.

Although it has been shown that the total activation inventory of a steel reactor structure is high, no information has been identified which would preclude the use of stainless steel. No structural activation products exist which are volatile under normal operating conditions. It appears that the activated materials are well-confined to the basic nuclear island structure so that adequate shielding can be provided in reactor design. Under accident conditions, liquid lithium may result in the volatilization of some components of steel. This is probably also true of any other structural material.

Maintenance-related activities will be the dominant source of radiation-environment exposures of personnel. Radiation exposures to maintenance personnel during anticipated and accident conditions should be investigated so that the activities to be performed remotely can be determined and the technology necessary to assure personnel safety can be developed.

Other candidate structural materials should be analyzed and compared with stainless steel. An evaluation of the acceptability of stainless depends partially on the relative characteristics of the alternative materials. In this respect qualitative differences should be stressed. A reduction of total plant inventory or contact dose rate is not by itself an improvement in plant safety. Only if expected releases during accidents or actual maintenance exposures are reduced is a gain realized.

There are several methods available for reducing inventories and dose rates in steel if that would result in a significant safety advantage. It has been seen that the presence of manganese, cobalt, and molybdenum result in a substantial fraction of the total inventory, particularly during the first few

days of operation. The effects of controlling impurities should be investigated, as should the relative merits of using other types of steel. Isotopic tailoring of steel has also been suggested as a means of reducing activation inventory⁽⁶¹⁾ and should be investigated.

Several mechanisms have been identified by which activation products can be transported away from the nuclear island. Two mechanisms which have not yet been reported here are neutron sputtering of activation products in the coolant system and activation of reactor building air. The first may conceivably lead to accumulation of radioactive materials in the coolant piping and heat exchangers, making personnel access to these areas difficult. Air activation may be the limiting factor in determining the waiting period after normal or emergency shutdown before personnel can access the reactor building.

REFERENCES

1. B. Badger, et al., UWMAK-I: A Wisconsin Toroidal Fusion Reactor Design. UWFD-68, University of Wisconsin, Madison, WI, 1974.
2. B. Badger, et al., UWMAK-II: A Conceptual Tokamak Power Reactor Design. UWFD-112, University of Wisconsin, Madison, WI, 1975.
3. D. Steiner, et al., ORNL Fusion Power Demonstration Study: Interim Report. ORNL/TM-5813, Oak Ridge National Laboratory, Oak Ridge, TN, 1977.
4. C. C. Baker et al., Conceptual Design Study of a Noncircular Tokamak Demonstration Fusion Power Reactor. GA-A13992, General Atomic Company, San Diego, CA, 1976.
5. A. P. Fraas, Conceptual Design of the Blanket and Shield Region and Related Systems for a Full Scale Toroidal Fusion Reactor. ORNL-TM-3096, Oak Ridge National Laboratory, Oak Ridge, TN, 1973.
6. B. Badger, et al., UWMAK-III: A Noncircular Tokamak Power Reactor Design. UWFD-150, University of Wisconsin, Madison, WI, 1976.
7. J. R. Powell, Ed., Preliminary Reference Design of a Fusion Reactor Exhibiting Very Low Residual Radioactivity. BNL-19565, Brookhaven National Laboratory, Upton, NY, 1974.
8. D. L. Kummer, Ed., Conference Proceedings: Low Activation Materials Assessment for Fusion Reactors. EPRI ER-328-SR, Electric Power Research Institute, Palo Alto, CA, 1977.
9. ASTM Handbook, 1977.
10. J. J. Heger, "Residual Elements in Stainless Steel - General Characteristics," in Effect of Residual Elements on Properties of Austenitic Stainless Steels. ASTM-STP418, American Society for Testing Materials, 1967.
11. W. W. Engle, A User's Manual for ANISN. K-1693, Oak Ridge Gaseous Diffusion Plant, Oak Ridge, TN, 1967.
12. R. T. Perry, H. Gorenflo, and W. Dänner, INDRA: A Program System for Calculating the Neutronics and Photonics Characteristics of a Fusion Reactor Blanket. IPP4/137, Max-Planck-Institut für Plasmaphysik, Garching, West Germany, 1976.
13. D. I. Garber and R. R. Kinsey, Neutron Cross Sections: Volume II, Curves. BNL-325, Brookhaven National Laboratory, Upton NY, 1976.

14. Theodore Rockwell III, Reactor Shielding Design Manual. D. Van Nostrand Co., Princeton, NJ, 1956.
15. "American National Standard Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," ANSI/ANS-6.1.1-1977 (N666), American Nuclear Society, La Grange Park, IL, 1977.
16. W. F. Vogelsang, "Radioactivity and Associated Problems in Thermonuclear Reactors," in Proc. Second Topical Meeting on the Technology of Controlled Nuclear Fusion. CONF-760935-P4, Richland, WA, September 1976.
17. H. Iida, Y. Seki and T. Ide, "Induced Activity and Dose Rate in a Fusion Reactor with Molybdenum Blanket Structure," J. Nuclear Science and Technology, 14 (11) 836-38, Nov. 1977.
18. K. Sako, et al., "Conceptual Design of the JAERI Fusion Reactor," in Proc. Second Topical Meeting on the Technology of Controlled Nuclear Fusion. CONF-760935-P2, Richland, WA, Sept. 1976.
19. D. Steiner, The Neutron-Induced Activity and Decay Power of the Niobium Structure of a D-T Fusion Reactor Blanket. ORNL-TM-3094, Oak Ridge National Laboratory, Oak Ridge, TN, August 1970.
20. J. R. Powell, et al., "Minimum Activity Blankets for Commercial and Experimental Power Reactors," in Fifth Symposium on Engineering Problems of Fusion Research. Princeton University, Princeton, NJ, November 1973.
21. M. A. Abdou, L. J. Milton, J. C. Jung, and E. M. Gelbard, "Multidimensional Neutronics Analysis of Major Penetrations In Tokamaks," in Proc. Second Topical Meeting on the Technology of Controlled Nuclear Fusion. CONF-760935-P3(1976).
22. W. M. Stacey, Jr., et al., Tokamak Experimental Power Reactor Conceptual Design. ANL/CTR-76-3, Argonne National Laboratory, Argonne, IL, 1976.
23. R. T. Santoro and R. G. Alsmiller, Jr., Neutronics Calculations for the TFTR Neutral Beam Injectors. ORNL/TM-5068, Oak Ridge National Laboratory, Oak Ridge, TN, 1976.
24. K. D. Lathrop and R. W. Brinkley, Theory and Use of the General Geometry TWOTRAN Program. LA-4432, Los Alamos Scientific Laboratory, Los Alamos, NM, 1970.
25. T. Ide, Y. Seki, and H. Iida, "Effects of Neutron Streaming Through Injection Ports on Neutronic Characteristics of a Fusion Reactor," in Proc. Second Topical Meeting on the Technology of Controlled Nuclear Fusion. CONF-760935-P2, Richland, WA, September 1976.

26. K. Ioki, Y. Harada, and N. Asami, Code ACTIVE for Calculation of the Transmutation, Induced Activity and Decay Heat in Neutron Irradiation. JAERI-M 6471, 1976. (in Japanese).
27. B. M. U. Scherzer, R. Behrisch, J. Roth, "Wall Erosion by Physical and Chemical Sputtering and Blistering," in Proc. International Symposium on Plasma Wall Interaction. Pergamon Press, New York, New York, 1977, pp. 355.
28. Tokamak Impurity Report. DOE/ET 0001, U.S. Dept of Energy, Division of Magnetic Fusion Energy, Washington, DC, 1977.
29. R. E. Clausing, L. C. Emerson, L. Heatherly, R. J. Colchin, "Surface Impurities and Clean-up Techniques in Ormak." Journal of Nuclear Materials 63:495-500, 1976.
30. R. J. Colchin, C. E. Bush, G. L. Jahns, J. F. Lyons, M. Murakami, R. V. Neidigh, D. L. Shaeffer, "Impurity and Surface Studies in Ormak." Journal of Nuclear Materials 63:75, 1976.
31. D. Brzezinska, "Application of the Vacuum Technique to the Deposition of Thin Films by Ionic Sputtering." Przemyslowy Instytut Elektroniki 12(3-4):95-101, 1971.
32. W. M. Stacey, Jr., D. L. Smith, J. N. Brooks, P. J. Bertoncini, D. A. Ehst, Impurity Control in Tokamak Reactors. ANL/FPP/TM-91, Argonne National Laboratory, Argonne, IL, 1977.
33. E. W. Sucov, Impurity and Gas Throughput Control for TNS. WFPS-TME-060, Westinghouse Electric Corp., Pittsburgh, PA, 1977.
34. R. G. Mills, et al., A Fusion Power Plant. MATT-1050, Princeton Plasma Physics Laboratory, Princeton, NJ, 1974.
35. O. K. Harling, M. T. Thomas, R. L. Brodzinski, L. A. Rancitelli, "Fast-Neutron Sputtering of Niobium." Journal of Applied Physics. 48(10):4324, 1977.
36. P. Sigmund, "Theory of Sputtering: Sputtering Yield of Amorphous and Polysrystalline Targets." Physical Review. 184(2):383-416, 1969.
37. R. Behrisch, "Neutron Sputtering," Nuclear Instruments and Methods. 132:293-300, 1976.
38. M. T. Thomas, O. K. Harling, R. L. Brodzinski, L. A. Rancitelli, "Fusion Neutron Induced Radioactive Emissions from Surfaces." Journal of Nuclear Materials 63:452, 1976.
39. B. M. U. Scherzer, et al., op. cit., p.369.

40. B. M. U. Scherzer, "Ion-Solid Interaction in Fusion Reactors." Journal of Vacuum Science and Technology 13:(1):424, 1976.
41. R. Behrisch, J. Bohdanský, G. H. Oetjen, J. Roth, G. Schilling, H. Verbeek, "Measurements of the Erosion of Stainless Steel, Carbon and SiC by Hydrogen Bombardment in the Energy Range of 0.5-7.5 keV." Journal of Nuclear Materials 60:321-329, 1976.
42. S. K. Das and M. Kaminsky, "Radiation Blistering in Metals and Alloys," in Radiation Effects on Solid Surfaces. Manfred Kaminsky, ed., American Chemical Society, Washington, DC, 1976, pp. 112-170.
43. J. L. Ballif, D. W. Jeppson, W. W. Yuan, and B. E. Chou, Lithium Literature Review: Lithium's Properties and Interactions. HEDL TC-1000, Hanford Engineering Development Laboratory, Richland, WA, 1978.
44. J. O. Cowles, A. D. Pasternak, Lithium Properties Related to Use as a Nuclear Reactor Coolant. UCRL-50647, Lawrence Radiation Laboratory, Livermore, CA, 1969.
45. C. Tyzak and P. B. Longton, "The Oxidation of Lithium." United Kingdom Atomic Energy Authority Technical Note 131, 1955.
46. M. M. Markowitz, "Alkali Metal-Water Reactions." J. Chem. Educ.: 40(12), 1962, pp. 633-636.
47. Yu I. Ostroushko, et al., Lithium, Its Chemistry and Technology. United States Atomic Energy Commission-tr-4940, Division of Technical Information, Washington, DC, 1962.
48. Atomic Energy Commission, Dept. of Navy, Washington, DC, "Liquid Metals Handbook," 2nd Edition, 1952.
49. R. N. Singh, "Compatibility of Ceramics with Liquid Na and Li." J. of the Am. Ceramic Society, 59(3-4):112-115, 1976.
50. D. L. Smith and K. Natesan, "Influence of Nonmetallic Impurity Elements on the Compatibility of Liquid Lithium with Potential CTR Containment Materials." Nuclear Technology 22, 1974.
51. D. Steiner, "The Technological Requirements for Power by Fusion." Nuclear Science and Engineering 58:107-165, 1975.
52. M. S. Kazimi (ed.), D. A. Dube, R. W. Green, L. M. Lidsky, N. C. Rasmussen, R. W. Sawdye, and J. A. Sefcik, "Aspects of Environmental and Safety Analysis of Fusion Reactors." Progress Report for the Period October 1, 1976, to September 30, 1977, Department of Nuclear Engineering, Massachusetts Institute of Technology, Cambridge, MA, 1977.

53. D. Okrent, W. E. Kastenberg, T. E. Botts, C. K. Chan, W. L. Ferrell, T. H. K. Frederking, M. J. Sehnert, and A. Z. Ullman, "On the Safety of Tokamak-Type, Central Station Fusion Power Reactors." Nuclear Engineering and Design, 39:215-238, 1976.
54. J. W. T. Spinks and R. J. Woods, An Introduction to Radiation Chemistry. Wiley, New York, New York, 1976, p. 218.
55. R. C. Weast (ed.), Handbook of Chemistry and Physics, 45th Edition. The Chemical Rubber Co., Cleveland, OH, 1964.
56. C. J. Smithells (ed.), Metals Reference Book, 5th Edition. Butterworths, London, 1976.
57. A. P. Zefirov (ed.), Thermodynamic Properties of Inorganic Substances; Handbook. Atomizdat, Moscow, pp. 460, 1965 (Russian).
58. Title 10, Code of Federal Regulations, Energy, Part 20, "Standards for Protection Against Radiation," (Revised 12-31-75), U.S. Nuclear Regulatory Commission, Washington, DC.
59. J. J. Stephens, Jr. and R. O. Pohl, "Trace Elements in Reactor Steels: Implications for Decommissioning." Nuclear Eng. Design 47:125, 1978.
60. C. M. Lederer, J. M. Hollander, and I. Perlman, Table of Isotopes, 6th Edition. John Wiley and Sons, New York, NY, 1967.
61. R. W. Conn, K. Okula, and W. Johnson, "Minimizing Long-Term Radioactivity in Fusion Reactors by Isotopic Tailoring." Transactions of the American Nuclear Society, 1977 Annual Meeting, American Nuclear Society, La Grange Park, IL, 1977, pp. 27.

APPENDIX A

REVIEW OF ACTIVATION CODES IN USE FOR FUSION REACTOR DESIGNS

by

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INTRODUCTION

The activation and dose rate calculations presented in this report were made using a number of simplifying assumptions in order to solve the activation and decay equations analytically rather than numerically. Analysis of these results showed that for almost all nuclides this simplified method is accurate to within the desired uncertainty of current calculations and the accuracy of input cross section data. However, in the future, it may be desirable to use more sophisticated codes, particularly for certain nuclides (e.g., ^{58}Co) and structural materials other than 316 stainless steel. Hence a review of calculational methods employed by other researchers and used in major fusion power reactor design studies was undertaken.

In the modifications made by Willenberg to the GLUCKE computer code^(A1) it was necessary to assume no burnup of the parent stable nuclide, $n_m(t)$ in Equation 2. Since burnup cross sections for radioactive species were not available, the R_2 term for the activated products was taken as only the decay constant $\lambda_{p \rightarrow m}$. If these two criteria are relaxed and one allows for decay back to a nuclide that can be activated, the computational methods used are nontrivial and must be done numerically. There presently are at least ten codes in the literature which solve numerically the coupled linear ordinary differential equation set given in Equation 5. These are: DKR,^(A2-A4) AKTIV,^(A5) ACTIVE (Japanese),^(A7,A8) MATEXP,^(A9) ORIGEN,^(A10) TCAP,^(A11,A12) GAMACT,^(A13)

(a) Supported by the U.S. DOE Education Programs Division via NORCUS/University of Washington.

CINDER,^(A14-A16) and INDRA (GLUCKE).^(A1) The ORIGEN and CINDER codes were written for fission reactors, but by proper weighting of the cross section set used they can (and have, in the case of ORIGEN) be used for fusion calculations. The ACT/ACTIVE code is applicable only to inertial confinement fusion because the neutron source is assumed to be a delta function in time (burst) followed by decay. All of the codes except ACTIVE and CINDER accept multiregion-multienery group, one-dimensional neutron fluxes generated by the ANISN code^(A17-A19) and most use the ENDF/B-IV cross section library. THRES-2 is used where isotopic cross sections are needed. The chief differences between the codes are their decay chain libraries and their generation of gamma flux source terms. Each of these codes was originally written for a specific blanket structure, as they each have a specific decay chain library. The DKR, ORIGEN and CINDER codes have more complete libraries than most. Table A-1 lists the reactor design studies to which these codes were applied. Further discussion of each code will appear in the following sections.

There is presently no code which calculates activation, dose rates, etc., for two- and three-dimensional blankets with the precision that will be necessary when specific fusion reactor designs are being finalized. Some work on 2-D neutron flux calculations has been carried out for the neutral beam injectors of the JAERI reactor using TWOTRAN-GG,^(A83) and for the divertor openings of UWMAK-III with DOT.^(A84) Jung at Argonne is modifying the DKR code to accept neutron flux inputs from 2-D and 3-D codes, but this work is in its early stages. Shielding studies for TFTR neutral beam injectors are being done at Oak Ridge by Santoro, et al., using Monte Carlo calculations.^(A67) The flux maps from these could be scaled to power reactor conditions as first estimates of activation and dose rates in commercial designs. This is an important area of investigation as any penetration into the blanket-shield that cannot be adequately shielded will seriously restrict all maintenance activities in the reactor hall and personnel access during shutdown prior to removal of the shield.

Comparison of these calculations for the activation of blanket structural materials with the activation of light water reactor core support structures

**TABLE A-1. Summary of Fusion Reactor Design
Study Activation/Dose Calculations**

Design Study	Structural Material Proposed	Code Used	Comments	References
A. Tokamaks				
UWMAK-I (A20)	316 SS, no impurities Nb-1Zr V-20Ti TZM Al-2024	DKR	No impurities considered Alternative structural materials and isotopic tailoring considered	A36, A37, A38 A46, A47
	316 SS impurities SAP	MATEXP ANISN only	German study of 316 SS Brookhaven Low Activation Alternative Blanket Study	A39, A40 A41-A45
UWMAK-II (A21)	316 SS Nb-1Zr V-20Ti	DKR	No impurities considered	A21
UWMAK-III (A22)	TZM	DKR	No impurities considered	A22
ORNL EPR (A23)	Nb-1Zr Nb, V 304, Nb & Nimonic 105	MATEXP MATEXP MATEXP	No impurities considered No impurities considered No impurities considered	A38, A48, A49 A50-52 A53, A54
ORNL "Steiner" DEMO (A24)	316 SS	ORIGEN ANISN	No impurities considered Reported here with helium coolant	A58 This Report
GA EPR (A13)		GAMACT		A13
GA Noncircular DEMO (A25)	GAMACT	DKR	A25	
TETR (A26)		DKR		A26
ANL-Tokamak EPR (A27)		DKR		A27
Princeton Tokamak (A28)	PE-16, Nb (43Ni, 37Fe, 18Cr)	TCAP		A38, A59
JAERI (A29)		ACTIVE		A8, A66, A87
Culham Conceptual (A30)	PE 16, Nb	MATEXP		A60
Tokamak Reactor Mk-II (CCTR-II)	V-10Cr-20Ti			A61
German Blanket	316 SS V-10Cr-10Ti	AKTIV	Impurities Specifically Included	A62, A63
FINTOR (A31, A32)		ISPRA Code		A64
JET (A33, A34)		DOT		A65
TFTR (A35)		Monte Carlo		A66-A68
B. Other Magnetic Confinement Reactors				
Tandem Mirror Reactor (A69)	Inconel-718 SS Stainless Steel		Activity calculations not reported in detail	
Mirror (A70)		MORSE/ANISN	Preliminary results	
ANL/LASL-Theta Pinch	Nb-1% Zr, V-20% Ti	TR 3A		A55-A57
C. Inertial Confinement Reactors				
SOLASE (A73)	Al-6061 Ti-8Mn Al-1100	DKR/DOSE	Impurities in Al only included	A80
LLL-IFC (A74, A75)		ACT/ACTIVE		A81
LASL-IFC (A76, A77)	Nb Mo V-20Ti 316 SS SAP		not documented	A82
Sandia-EBR (A78)			No data	
HIDE (A79)			No data	

and pressure vessels at the time of decommissioning provides useful information. The calculations for a typical PWR have just been published using the ORIGEN code^(A85) and those for a BWR are expected within a year. In other words, fusion blanket activation and dose estimates using the codes listed in Table A-1 are at least as sophisticated and complete (if not more advanced) as those for structural materials activation in a technology that is now commercial.

Developing detailed activation/dose calculations this early in the design of commercial fusion reactors will help avoid some of the problems that have occurred in fission technology. Maintenance and repair problems occasionally arise in fission power plants which were not considered during reactor design. This has been a major cause of long power outages because neutron activation products may prevent direct contact maintenance and repair. In some cases, the maintenance problems were overlooked because of a lack of detailed analysis of neutron activation.

The problem of radiation exposure during maintenance and repair operations may be magnified for fusion, since major structures with complex configurations such as blankets, shields, and vacuum vessels will need to be periodically replaced on a routine basis. This will greatly increase the effort required for maintenance. Consideration of maintenance, component replacement, and decommissioning is therefore essential in fusion reactor design. Sophisticated analytical techniques are justified for fusion even during preliminary conceptual design studies for commercial power plants.

CODE DATA NEEDS

The primary deficiency in existing data for activation codes which numerically calculate the production and decay of radioactive species is the availability of isotopic, energy-dependent neutron reaction cross sections--not just for the stable nuclides but also for the radioactive species produced. While energy-dependent reaction cross sections are available up to 14 MeV for many stable isotopes, there are almost no measured energy-dependent cross sections for radioactive species (e.g., ⁵⁸Co, ⁵³Mn, ⁵⁹Ni, ⁶³Ni, ⁹⁴Nb). Activation inventories may be reduced by transmutation of long-lived radioactive species,

but until cross sections are available it is difficult to estimate the magnitude of this reduction. Efforts should thus be made in the coming years to include the radioactive species in cross section measurements after assessing which species are most affected by fusion neutrons.^(a)

Radioactive chain decay and isomeric states form a second major area of data needs. Isotopic chains and half-lives are well known and well understood. Fortunately, since one starts with stable nuclides, activation of structural material rarely leads to a chain with more than three species. The simplified model used in Section III neglects chain decay and isomeric states. As long as activation of radioactive species and burnup of stable nuclei are neglected, it is a simple matter to include chain decay. The biggest gap in knowledge today is in the branching ratios for production by fast neutrons of isomeric states (e.g., $^{98}\text{Mo} \rightarrow ^{99\text{m}}\text{Mo}$ or $^{98\text{m}}\text{Mo} \rightarrow ^{99}\text{Mo}$). Some of the standard materials lie in the region of the "islands" of isomers (Nb, Mo, Tc). Fortunately $^{92\text{m}}\text{Nb}$ and $^{93\text{m}}\text{Nb}$ have only microsecond lifetimes and can be neglected in calculations. But if activation of either the long-lived ^{92}Nb or ^{94}Nb (13.6 b thermal cross section) is included branching ratios must be known.^(A48) Figure A-1 shows the niobium decay chain and illustrates this point.

CODE CALCULATIONAL DIFFICULTIES

While differencing of Equation 4 and numerical time-step integration using matrix inversion is straightforward since most of the non-zero elements lie near the diagonal, difficulties are encountered for nuclides which have half-lives shorter than the time step used. To facilitate rapid convergence, short-lived isotopes should be treated separately and the Bateman equation solutions should be used as suggested in the ORIGEN code.^(A10) A review of the accuracy and reliability of the various codes listed in Table A-1 is impossible at this time because no two codes have been used to calculate the

(a) Based on the results presented in Section III, Willenberg estimated that very few radioactive species may be burned up more than a few percent (e.g., ^{59}Ni , ^{58}Co , ^{53}Mn).

FIGURE A-1. Activation-Decay Chain for an Isomeric Group

same case nor are there any experimental data available against which to check the results. ORIGEN and CINDER have been extensively checked for LWR calculations^(A10) and are now accepted for licensing calculations. In the case of fusion calculations, it is not the codes that would cause uncertainty, but the uncertainty in the input cross-section data, some of which are available only from theoretical models or "guesstimates." In many cases the impurity concentrations in the structural materials are also unknown; some of these may contribute significantly to the activity and dose.^(A86)

CODE DESCRIPTIONS

The primary differences between the various activation codes listed in Table A-1 are their reaction libraries and decay chain libraries. Almost universally ANISN and the ENDF cross-section library are used. Hence the chief differences between codes lie in the extent of the activation and decay

libraries. The codes are restricted to the nuclides and reactions contained therein. GLUCKE^(A1) and AKTIV^(A5) are primarily intended for stainless steel. GLUCKE restricts its library to 68 reactions shown in Table 3 and put into "Chart of the Nuclides" in Figure A-2. It and AKTIV are a part of the INDRA Neutronic and Photonic Program System developed by the Max-Planck-Institut für Plasmaphysik.^(A1) The information flow diagrams for this system are shown in Figures A-3 and A-4. Somewhat similar, but less ambitious code diagrams for the DKR^(A2-A4) and fusion ORIGEN package^(A10) are shown in Figures A-5 and A-6. The other codes, with the exception of ACT/ACTIVE^(A5) which is for neutron burst applications such as pellet fusion (inertial confinement), all have similar formats but are specific to the user group's computational package. Hence the discussion below will be limited to differences in the codes from the reaction chain indicated in Table 3 and Figure A-2.

AKTIV

The AKTIV library has 94 reactions for stainless steel, including isomeric states for Ti, Sc, V, Co, Mn, Cr, Ni and Fe. It does not include Al, C, Nb, Cu, Mo and Si impurities contained in GLUCKE or in the studies by Gruber and Schneider^(A61-A63) at the Hahn-Meitner-Institut für Kernforschung which also include O, N, Ti, Ta, S, P, As and B impurities. AKTIV includes neutron reactions with radioactive nuclides ⁴⁹V, ⁵³Mn, ⁵⁴Mn, ⁵⁵Fe, ⁶⁰Co, ⁵⁹Ni and ⁶³Ni, but not ⁵²V, ⁵¹Cr, ⁵⁵Cr, ⁵³Fe, ⁵⁹Fe, ⁵⁷Co, ⁵⁸Co, and ⁶³Ni shown in Figure A-2. AKTIV does not include (n,γ) reactions for ⁵¹V, ⁵⁰Cr, ⁵²Cr, ⁵³Cr, ⁵⁴Cr, ⁵⁶Fe, ⁵⁷Fe, ⁶⁰Ni, and ⁶¹Ni which are in GLUCKE and surprisingly has no reaction for ⁵⁹Co which is the only stable isotope of that impurity in stainless steel. The additional 70 reactions not in the GLUCKE library are primarily those for (n,α), (n,d), (n,n'α), (n,t) reactions not considered in GLUCKE and those with radioactive nuclides. Unfortunately, no computations using AKTIV have been reported to compare with the results in Section III, the results of the DKR code, and those of Gruber and Schneider for stainless steel. While the library of reactions is more extensive, the omissions in AKTIV and the exclusion of impurities may make it not as useful as thought.

FIGURE A-2. GLUCKE Library Reactions for Activation Ti Through Zn and Zr Through Ru

FIGURE A-4. Structure of the Program AKTIV

FIGURE A-5. Calculational Scheme Used in UWMAK Studies

FIGURE A-6. Calculational Procedure Flowchart for Modified ORIGEN System

DKR

The DKR code library used in the UWMAC studies contains many of the same isotopes as the GLUCKE library (additions being ^{13}C , ^{16}O , $^{207,208}\text{Pb}$, $^{182-186}\text{W}$, and ^{94}Mo). Its reaction library is more extensive for each isotope as it includes (n,α) , (n,t) , (n,d) and $(n, n'\alpha)$ reactions, for a total of 121 different reactions. The code has a few isomeric states in the library ($^{60\text{m}}\text{Co}$, $^{99\text{m}}\text{Tc}$, $^{95\text{m}}\text{Nb}$, and $^{93\text{m}}\text{Nb}$). The code uses only linear decay chains to avoid the problem of matrix inversion. A simple recursion relation is thus possible to use when Equation 4 is put in numerical form. This eliminates the possibility of including transmutation of a radionuclide if the daughter is not a member of the chain or if the chain branches to an isomer. The DKR code has been used extensively not only at Wisconsin but at Argonne, and used some at GA. Its calculational procedure is shown in Figure A-5.

ORIGEN

Smith and Parish^(A53,A54) modified the input library cross-section data for the ORIGEN code so that it could do fusion activation calculations. Their calculational procedure is shown in Figure A-6. ORIGEN takes only three neutron energy groups rather than the full ANISN output so it is necessary to provide a weighted one; two; or three-group set. While the library Smith constructed was very limited and included only (n,γ) , $(n,2n)$, $(n,3n)$, (n,α) , $(n,n'\alpha)$, (n,p) , (n,d) and (n,t) with the stable isotopes of Fe, Ni and Cr, and those stable nuclides formed by decay from the initially stable Fe, Ni, and Cr, their solution represents a contribution in displaying the potential of a code which will be available to most researchers. Smith and Parish did not include impurity elements in stainless steel, but ORIGEN has been run for LWR and LMFBR flux distributions for the important impurities and this library can be easily modified to include the fast neutron reactions above the fission spectrum. An alternate but equivalent code is the LASL CINDER code prepared for EPRI. This code will also be widely available in the near future.

INDRA

The INDRA^(A1) code package shown in Figure A-3, which contains the subprogram GLUCKE used in this report, is a complete package which also calculates gamma fluxes (as is ANISN). As it stands, however, INDRA does not contain a subprogram which calculates time-dependent activation. This is the purpose of AKTIV, but as noted AKTIV's library currently does not include all the blanket elements in the INDRA library.

MATEXP

The MATEXP^(A9) code is a general purpose numerical matrix inversion code which solves Equation 4. As such, it has been used in the codes such as ORIGEN and AKTIV. Gruber and Schneider have adapted MATEXP for fusion activation calculations by adding subprograms for data input: ATOM to calculate atom densities, IWAM-1 to interpolate cross sections, and DOSIS to calculate gamma doses. The code documentation, in German, does not indicate how the subprograms are coupled together or neutron energy group data (99

groups), or how the reaction library is kept in the code. A complete map of reactions similar to Figure A-2 is documented but these are displayed by groups and not as a full "Chart of the Nuclides" matrix like Figure A-2. When many reactions are included, the "Chart of Nuclides" flow diagram becomes so complex it is necessary to break it apart this way for clarity. The Gruber and Schneider library contains 240 nuclides from ^3H to ^{183}W . Over 250 reactions with stable and unstable nuclides are included. The library is thus more complete than the other codes and includes most impurities encountered in structural material.

ACTIVE

The Japanese have written a matrix inversion code called ACTIVE^(A87) to calculate blanket, structural and magnet activation rates in the JAERI Demonstration Fusion Reaction (JDFR). The documentation on this code is limited and not all translated to English. The actual code documentation (Reference A7) could not be obtained in the U.S. to review but portions of it are contained in several reports and papers in the open literature^(A8,A66,A83,A87). From these publications the ACTIVE reaction library appears to contain a fairly complete reaction set for the isotopes of stainless steel (Cr, Fe, Ni) without impurities, B_4C , Pb, and concrete-shield materials (no impurities), Cu and Nb normal and superconducting magnets. Of the calculations reviewed, the set of materials included in the JAERI activation and dose modeling of a demonstration power reactor is perhaps the closest to a real design of all the studies (see Reference A87, pp. 18-19) and the numbers calculated compare to those calculated in this report for the total activity, first wall and shield contact dose using GLUCKE. Unfortunately full details of the calculations were not available for review and some inconsistency between the reaction diagrams and equations were noted in the report which gives some details in English (text mostly in Japanese).^(A87)

Figure A-7 shows the information flow diagram for the ACTIVE-ANISN package. Gamma transport is treated by feeding the ACTIVE results into a subprogram, SOURCE which calculates the gamma ray source in the various zones of the reactor.

FIGURE A-7. Information Flow Diagram for the JAERI Code
ACTIVE(A7,A82)

Other Codes

The ACT/ACTIVE code^(A6) was written at Livermore originally for underground neutron testing and PLOWSHARE projects. It has been adapted recently to inertial confinement fusion. ACT calculates the fast neutron flux reactions solving for them using the ID-MEG code. Ground states and isomeric states for $(n,3n)$, $(n,2n)$, (n,γ) , (n,p) , and (n,α) reaction are included. Time steps are taken for less than 2% burn-up of the sum of all isotopes. ACTIVE uses the 2-D thermal flux results from TART. No published data is available on the activation library.

The Princeton Plasma Physics Laboratory has developed an activation code, TCAP (Transmutation Chain Analysis Program) which utilizes MATEXP. The chain library is not documented and the code and library have recently been destroyed with the departure of Dr. Price from Princeton. A code was developed at General Atomic, GAMACT,^(A13) to calculate gamma ray production from ANISN results to then be fed back into ANISN to calculate gamma transport. No documentation on the code has been obtained to date. The code is a point code and being replaced by the DKR code.

Los Alamos has done blanket structural calculations for the Reference Theta-Pinch Reactor (RTPR)^(A55-A57) and developed a code package to fit into the DTF-IV neutron transport code. The activation code (TR3A) itself is not documented in the available literature but some of the activation chains included are in the publications.

Finally, Culham has developed a code for transmutation calculations of fusion blanket structures.^(A60) The code was not available for review and the published literature is sketchy on the activation chains for PE-16 and Nb.

SUMMARY

The transmutation and dose codes available for fusion blanket structural material and shield calculations have been reviewed. While individual codes are centered around standard neutron transport codes such as ANISN, there is a wide diversity in the transmutation and decay chain libraries that have

been developed. An effort now needs to be made to develop a universally-accepted library and to "standardize" the reaction set included. Particularly needed is an effort to obtain an energy-dependent set of cross-sections for the transmutation of selected radionuclides (analogous to the Xe and Sm burnup in fission reactors). This report has indicated that calculations using simplified analytical solutions not including burnup of the parent and daughter nuclides provide reasonably accurate results. However, this is totally dependent on having good input cross-sections, including important reactions and impurities. It is difficult to obtain initial concentration data on impurities as they are highly dependent on the process used to make the structural materials, on the raw materials used, and the techniques employed to control impurities. Manufacturers are reluctant to release data on impurities as they vary significantly from batch to batch and to date no procedure to include trace element analyses as a routine part of the QA has been initiated. This suggests that a research program using a 14 MeV neutron source be initiated to measure trace elements in various proposed structural materials using neutron activation analysis.

BIBLIOGRAPHY

- A1 R. T. Perry, H. Gorenflo and W. Dänner, INDRA: A Program System for Calculating the Neutronics and Photonics Characteristics of a Fusion Reactor Blanket. IPP 4/137, Max-Planck-Institut für Plasmaphysik, Garching, West Germany, 1976.
- A2 T. Y. Sung, DKR: A Radioactivity Calculation Code for Fusion Reactors. UWFD-170, University of Wisconsin, Madison, WI, 1970.
- A3 T. Y. Sung, Radioactivity Calculations in Fusion Reactors. Ph.D Dissertation, Dept. of Nuclear Engineering, University of Wisconsin, Madison, WI, 1976.
- A4 T. Y. Sung and W. F. Vogelsang, Decay Chain Data Library for Radioactivity Calculations. UWFD-171, University of Wisconsin, Madison, WI, 1976.
- A5 H. J. Odenthal, W. Dänner and H. Gorenflo, AKTIV: A Computer Program for Evaluating the Activity, Afterheat and Biological Hazard Potential of Stainless Steel Structures in Fusion Reactor Blankets. IPP 4/154, Max-Planck-Institut für Plasmaphysik, Garching, West Germany, 1977.
- A6 R. M. Leeser, W. E. Alley and J. B. Green, ACT and ACTIVE Codes for Calculating Neutron-Induced Activation. UCRL-51292, Lawrence Livermore Laboratory, Livermore, CA, 1978.
- A7 K. Ioki, Y. Harada and N. Asami, Code, ACTIVE for Calculation of Transmutation, Induced Activity, and Decay Heat in Neutron Irradiation. JAERI-M 6471 (in Japanese), Japan Atomic Energy Research Institute, Tokai, Japan, 1976.
- A8 H. Iida, Y. Seki and T. Ide, "Induced Activity and Dose Rate in a Fusion Reactor with Molybdenum Blanket Structure." J. Nuc. Sci. and Tech. 14:836, 1977.
- A9 M. J. Bell and R. K. Adams, MATEXP, A General Purpose Digital Computer Program for Solving Ordinary Differential Equations by the Matrix Exponential Method. ORNL-TM-1933, Oak Ridge National Laboratory, Oak Ridge, TN, 1977.
- A10 M. J. Bell, ORIGEN-The ORNL Isotope Generation and Depletion Code. ORNL-4628 (May 1973) as contained in RSIC Documents CCC-217 (May 1977) and DCL-38 (Feb 1976). Oak Ridge National Laboratory, Oak Ridge, TN.
- A11 W. G. Price, Jr., Activation and Afterheat in a Fusion Power Reactor. MATT-1006, Princeton Plasma Physics Laboratory, Princeton, NJ, 1973.
- A12 W. G. Price, Jr., "Chapter 10, Neutronics," pp. 164-215 in A Fusion Power Plant (R. G. Mills, ed.). MATT-1050, Princeton Plasma Physics Laboratory, Princeton, NJ, 1974.

- A13 C. C. Baker, et al., Experimental Fusion Power Reactor Conceptual Design Study. EPRI ER-289, General Atomic Co., San Diego, CA, 1976.
- A14 T. R. England, CINDER - A One-Point Depletion and Fission Product Program. WAPD-TM-334 (rev.), Bettis Atomic Power Laboratory, Pittsburgh, PA, 1964.
- A15 T. R. England, W. B. Wilson, and M. G. Stamatelatos, Fission Product Data for Thermal Reactors. Part 1: A Data Set for EPRI-CINDER Using ENDR/B-IV. Los Alamos Scientific Laboratory, Los Alamos, NM, 1976.
- A16 T. R. England, W. B. Wilson and M. G. Stamatelatos, Fission Product Data for Thermal Reactors. Part 2: User's Manual for EPRI-CINDER Code and Data. LA-6746-MS, Los Alamos Scientific Laboratory, Los Alamos, NM, 1976.
- A17 W. W. Engle, A User's Manual for ANISN. K-1693, Oak Ridge Gaseous Diffusion Plant, Oak Ridge, TN, 1967.
- A18 ANISN-W Multigroup One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering. CCC-255, Radiation Shielding Information Center, Oak Ridge, TN, 1975.
- A19 ANISN-CEA, One Dimensional Discrete Ordinates Transport Code. LCC-82 (ORNL-tr-2419 and 2408) (CEA-N-1358 and 1335), Radiation Shielding Information Center, Oak Ridge, TN, 1970.
- A20 B. Badger, et al., UWMAK-I: A Wisconsin Toroidal Fusion Reactor Design. UWFD-68, University of Wisconsin, Madison, WI, 1974.
- A21 B. Badger, et al., UWMAK-II: A Conceptual Tokamak Power Reactor Design. UWFD-112, University of Wisconsin, Madison, WI, 1975.
- A22 B. Badger, et al., UWMAK-III: A Noncircular Tokamak Power Reactor Design. UWFD-150, University of Wisconsin, Madison, WI, 1976.
- A23 A. P. Fraas, Conceptual Design of the Blanket and Shield Region and Related Systems for a Full Scale Toroidal Fusion Reactor. ORNL-TM-3096, Oak Ridge National Laboratory, Oak Ridge, TN, 1973.
- A24 D. Steiner, et al., ORNL Fusion Power Demonstration Study: Interim Report. ORNL/TM-5813, Oak Ridge National Laboratory, Oak Ridge, TN, 1977.
- A25 C. C. Baker, et al., Conceptual Design Study of a Noncircular Tokamak Demonstration Fusion Power Reactor. GA-A13992, General Atomic Company, San Diego, CA, 1976.

- A26 B. Badger, et al., TETR - A Tokamak Engineering Test Reactor to Qualify Materials and Blanket Components for Early DT Fusion Power Reactors. UWFD-191, University of Wisconsin, Madison, WI, 1977.
- A27 W. M. Stacey, Jr., et al., Tokamak Experimental Power Reactor Conceptual Design. ANL/CTR-76-3, Argonne National Laboratory, Argonne, IL, 1976.
- A28 R. G. Mills (ed.), A Fusion Power Plant. MATT-1050, Princeton Plasma Physics Laboratory, Princeton, NJ, 1974.
- A29 K. Sako, T. Tone, Y. Seki, H. Iida, H. Yamato, N. Asami, T. Yamamoto, T. Suzuki, K. Maki, and K. Ioki, "Conceptual Design of the JAERI Demonstration Fusion Reactor." Proc. of the Second Topical Meeting on the Technology of Controlled Nuclear Fusion. CONF-760935, Richland, WA, 1976, pp. 607-618.
- A30 J. T. D. Mitchell, "The Impact of Servicing Requirements on Tokamak Fusion Reactor Design," Int. School of Fusion Reactor Technology, Erice-Trapani, September 1976. Pergamon, New York, 1978.
- A31 The FINTOR Group, FINTOR-I - A Minimum Size Tokamak D-T Experimental Reactor. JRC EURATOM ISPRA Report, 1976.
- A32 E. Bertolini, et al., "Conceptual Design Studies of an Experimental Tokamak Power Reactor." pp. 609-622 in Proc. of the 8th Symposium on Fusion Technology, EURATOM Report EUR5182, 1974.
- A33 P. H. Rebut, "First Description of the Joint European Torus." Proc. of the 8th Symposium on Fusion Technology. EURATOM Report EUR5182, pp. 273-316, 1974.
- A34 The JET Project, Design Proposal for the Joint European Torus, EUR-5515e (1976) and Scientific and Technical Development. EUR-5791e (1977).
- A35 Final Environmental Statement, Tokamak Fusion Test Reactor Facilities, U.S. Energy Research and Development Administration, Washington, DC ERDA-1544, July 1975.
- A36 W. F. Vogelsang, G. L. Kulcinski, R. G. Lott and T. Y. Sung, "Transmutations, Radioactivity and Afterheat in a D-T Tokamak Fusion Reactor." Nucl. Tech. 22:379, 1974.
- A37 W. F. Vogelsang, "Radioactivity and Associated Problems in Thermonuclear Reactors," Proc. of the Second Topical Meeting on the Technology of Controlled Nuclear Fusion. CONF-760935, Richland, WA, 1976, pp. 1303-1318.

- A38 R. W. Conn, T. Y. Sung and M. A. Abdou, "Comparative Study of Radioactivity and Afterheat in Several Fusion Reactor Blanket Designs." Nucl. Tech. 26:391, 1975.
- A39 J. Gruber and J. Schneider, Transmutation und Aktivierung von Stainless Steel 316 SS in Einem Thermischen Fusions-Reaktor-Blanket, Part I, HMI-B 212, Hahn-Meitner-Institut für Kernforschung, Berlin, West Germany, 1976.
- A40 J. Schneider, Transmutation und Aktivierung von Stainless Steel 316 SS in Einem Thermischen Fusions-Reaktor-Blanket, Part II, HMI-B 212, Hahn-Meitner-Institut für Kernforschung, Berlin, West Germany, 1976.
- A41 J. R. Powell, F. T. Miles, A. Aronson and W. E. Winche, Studies of Fusion Reactor Blanket with Minimum Radioactive Inventory and with Tritium Breeding in Solid Lithium Compounds. BNL-18236, Brookhaven National Laboratory, Upton, NY, 1973.
- A42 J. R. Powell, A. Aronson, P. Bezler, F. Miles and W. Winche, "Minimum Activity Blankets Using Aluminum Structure." in Proc. of First Topical Meeting on the Technology of Controlled Nuclear Fusion. CONF-740402-P1, San Diego, CA, 1974, pp. 533-547.
- A43 J. R. Powell, F. T. Miles, A. Aronson, W. E. Winche and P. Bezler, "Minimum Activity Blankets for Commercial and Experimental Power Reactors." Fusion Reactor Design Problems, IAEA, Vienna, 1974, pp. 243-395.
- A44 R. Benenati, et al., "BNL Minimum Activity Aluminum Blanket Designs." Proc. of the Magnetic Fusion Energy Blanket and Shield Workshop, Vol. II, ERDA-76/117-2, CONF-760343, Upton, NY, 1976, pp. 266-362.
- A45 J. R. Powell, (Ed.), Preliminary Reference Design of a Fusion Reactor Blanket Exhibiting Very Low Residual Radioactivity. BNL-19565, Brookhaven National Laboratory, Upton, NY, 1974.
- A46 R. W. Conn, K. Okula and W. Johnson, "Minimizing Long-Term Radioactivity in Fusion Reactors by Isotopic Tailoring." Trans. Am. Nucl. Soc. 27:27, 1977.
- A47 R. W. Conn, K. Okula and A. W. Johnson, "Minimizing Radioactivity and Other Features of Elemental and Isotopic Tailoring of Materials for Fusion Reactors." preprint, Dept. of Nuclear Engineering, University of Wisconsin, Madison, WI, May 19, 1978.
- A48 D. Steiner, The Neutron-Induced Activity and Decay Power of the Niobium Structure of a D-T Fusion Reactor Blanket. ORNL-TM-3094, Oak Ridge National Laboratory, Oak Ridge, TN, 1970.

- A49 D. Steiner, "Nuclear Performance of Vanadium as a Structural Material in Fusion Reactor Blankets." ORNL-TM-4353, Oak Ridge National Laboratory, Oak Ridge, TN, 1973.
- A50 D. Steiner, "The Nuclear Performance of Vanadium as a Structural Material in Fusion Reactor Blankets." Nucl. Fusion 14:33, 1974.
- A51 D. Steiner and A. P. Fraas, "Preliminary Observations on the Radiological Implications of Fusion Power." Nuclear Safety 13:5, 1972.
- A52 M. L. Williams, R. T. Santoro and T. A. Gabriel, "The Calculated Performance of Various Structural Materials in Fusion Reactor Blankets." Nucl. Tech. 29:384, 1976.
- A53 R. D. Smith, Structure Activation in Fusion Devices. Masters Thesis, Department of Mechanical Engineering, The University of Texas at Austin, May 1977.
- A54 R. D. Smith and T. A. Parish, "Structure Activation in an Experimental Power Reactor." Proc. 7th Symposium on Engineering Problems in Fusion Research, IEEE 77CH1267-4-NPS, New York, NY, 1977, pp. 1718-1722.
- A55 D. J. Dudziak and R. A. Krakowski, "Radioactivity Induced in a Theta-Pinch Fusion Reactor." Nucl. Tech. 25:32, 1975.
- A56 J. E. Draley, V. A. Maroni, T. A. Coultas and R. A. Krakowski, "Environmental Impact Study of a Reference Theta-Pinch Reactor (RTPR)." Proc. of the First Topical Meeting on the Technology of Controlled Nuclear Fusion. CONF-740402-P1, San Diego, CA, 1974, pp. 564-577.
- A57 D. J. Dudziak and R. A. Krakowski, "A Comparative Analysis of D-T Fusion Reactor Radioactivity and Afterheat." Proc. of First Topical Meeting on the Technology of Controlled Nuclear Fusion. CONF-740402-P1, San Diego, CA, 1974, pp. 548-563.
- A58 J. W. Nigg and J. N. Davidson, "The Induced Activity and Decay Power of the Structure of a Stainless Steel Fusion Reactor Blanket." Proc. First Topical Meeting on Technology of Controlled Nuclear Fusion. San Diego, CA, 1974, pp. 578-587.
- A59 W. G. Price, Jr., "Activation Hazards from Fusion Neutrons," Proc. of the 5th Symposium on Engineering Problems of Fusion Research. IEEE Pub. No. 73CH0843-3-NPS, New York, NY, 1974, pp. 217-221.
- A60 S. Blow, "Some Features of the Behavior of Structural Materials in a Possible Fusion Reactor Blanket." J. Brit. Nucl. Eng. Soc. 11:371, 1972.

- A61 J. Gruber, "Transmutation and Activity in a Thermal Blanket," Proc. of the 6th Symposium on Engineering Problems of Fusion Research. IEEE Pub. 75CH1097-5-NPS, New York, NY, 1975, pp. 1192-1196.
- A62 J. Gruber, "Evaluation of the Activity Levels in Fusion Reactor Blankets." Hahn-Meitner-Institut für Kernforschung Report BC-69/HMI-B-202, May 1977.
- A63 F. K. Altenheim, H. Andresen, J. Gruber, W. Lutze and H. Migge, "Beiträge zum Fusions Reactor Program." pp. 77-83 in Wissenschaftlicher Ergebnisbericht 3, 1975, HMI-B 199 Hahn-Meitner-Institut für Kernforschung Berlin, West Germany, 1975.
- A64 F. Farfaletti-Casali, F. Peter and P. Rocco, "Radioactivity and Afterheat of FINTOR and Related Problems of Maintenance and Waste Disposal." Proc. of the 9th Symposium on Fusion Technology. Pergamon Press, New York, NY, 1976, pp. 453-458.
- A65 H. J. Brockmann and U. Ohlig, "Activation from D-T Fusion Neutrons and Resulting Dose Rates in JET." Proc. of the 9th Symposium on Fusion Technology, Pergamon Press, New York, NY., 1976, pp. 629-634.
- A66 Y. Seki, et al., Induced Activity of the Copper Coil in D-T Burning Experiment. JAERI-M 5862 (in Japanese) Japan Atomic Energy Research Institute, Tokai, Japan, 1974.
- A67 R. T. Santoro, J. S. Tang, R. G. Alsmiller, Jr. and J. M. Barnes, "Monte Carlo Analysis of the Effects of a Blanket-Shield Penetration in the Performance of a Tokamak Fusion Reactor." Nucl. Tech. 37:65, 1978.
- A68 R. T. Santoro, V. C. Baker and J. M. Barnes, "Neutronics and Photonics Calculations for the Tokamak Experimental Power Reactor." Nucl. Tech. 37:274, 1978.
- A69 R. W. Moir, et al., Preliminary Design Study of the Tandem Mirror Reactor (TMR). UCRL-52302, Lawrence Livermore Laboratory, Livermore, CA, 1977.
- A70 T. H. Batzer, et al., Conceptual Design of a Mirror Reactor for a Fusion Engineering Research Facility (FERF). UCRL-51617 Lawrence Livermore Laboratory, Livermore, CA, 1977.
- A71 R. A. Krakowski, F. L. Ribe, T. A. Coultas and A. J. Hatch, An Engineering Design of a Reference Theta-Pinch Reactor, Vol. I. ANL 8019/LA 5336, Argonne National Laboratory, Argonne, IL, 1974.
- A72 T. A. Coultas, J. E. Draley, V. A. Maroni and R. A. Krakowski, An Engineering Design of a Reference Theta-Pinch Reactor-RTPR, Vol. II: An Environmental Impact Study. ANL-8019/LA5336, Argonne National Laboratory, Argonne, IL, 1974.

- A73 R. W. Conn, et al., SOLASE, A Conceptual Laser Fusion Reactor Design. UWFDM-220, University of Wisconsin, Madison, WI, 1977.
- A74 W. R. Meier and J. A. Maniscalco, Reactor Concepts for Laser Fusion. UCRL preprint 79654, Proceedings AIChE, Nov. 13-17, New York, NY, 1977.
- A75 J. A. Maniscalco, J. Hovingh and R. R. Buntzen, "A Development Scenario for Laser Fusion," Trans. Am. Nucl. Soc. 22:66, 1975.
also see J. Maniscalco, J. Brink, R. Buntzen, J. Hovingh, W. Meier, M. Monsler and P. Walker, Civilian Applications of Laser Fusion, UCRL-52349, Lawrence Livermore Laboratory, Livermore, CA, 1977.
- A76 L. A. Booth, D. A. Freiwald, T. G. Frank and F. T. Finch, "Prospects of Generating Power with Laser-Driven Fusion," Proc. IEEE 64:1460, 1976.
- A77 L. A. Booth and T. G. Frank, Commercial Applications of Inertial Confinement Fusion. LA-6838-MS, Los Alamos Scientific Laboratory, Los Alamos, NM, 1977.
- A78 S. G. Varnado, J. L. Mitchiner and G. Yonas, Civilian Applications of Particle-Beam-Initiated Inertial Confinement Fusion Technology. SAND77-0516, Sandia Laboratories, Albuquerque, NM, 1977.
- A79 L. W. Smith, (ed.), Proceedings of the Heavy Ion Fusion Workshop. BNL-50769, Brookhaven National Laboratory, Upton, NY, 1977.
- A80 S. I. Abdel-Khalik, R. W. Conn, G. A. Moses, E. T. Cheng, E. M. Larsen, E. G. Lovell, C. W. Maynard, I. N. Sviatoslavsky and W. G. Wolfer, "Engineering Design Aspects for SOLASE," Trans. Am. Nucl. Soc. 27:37, 1977.
- A81 M. S. Singh, Radiological Safety Design Considerations for a Laser-Fusion Facility. Preprint UCRL-78580 for Proceedings of the 5th International Conference on Reactor Shielding, April 18-22, 1977.
- A82 T. G. Frank, D. J. Dudziak and E. L. Heck, "Laser-Fusion-Reactor-Induced Radioactivity and Afterheat." Trans. Am. Nucl. Soc. 21:464, 1974.
- A83 T. Ide, Y. Seki and H. Iida, "Effects of Neutron Streaming Through Injection Ports on Neutronic Characteristics of a Fusion Reactor." Proc. of the Second Topical Meeting on the Technology of Controlled Nuclear Fusion. CONF-760935-P1, Richland, WA, 1976.
- A84 F. R. Mynatt, et al., The DOT-III Two-Dimensional Discrete Ordinates Transport Code. ORNL-TM-4280, Oak Ridge National Laboratory, Oak Ridge, TN, 1973.

- A85 R. I. Smith, G. J. Konzek and W. E. Kennedy, Jr., Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station. US NRC NUREG/CR-0130, Pacific Northwest Laboratory, Richland, WA, 1978, Chapter 7 and Appendix C.
- A86 J. J. Stephens, Jr. and R. O. Pohl, "Trace Elements in Reactor Steels: Implications for Decommissioning." Nucl. Eng. and Design 47:125, 1978.
- A87 H. Iida, Y. Seki and T. Ide, Induced Activity and Dose Rates in the JAERI Experimental Fusion Reactor (JEFR). JAERI-M 6639, Japan Atomic Energy Research Institute, Tokai, Japan, 1977.

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