Assessment of Radiological Releases to the Environment from a Fusion Reactor Power Plant

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OAK RIDGE NATIONAL LABORATORY
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ASSESSMENT OF RADIOLOGICAL RELEASES TO THE ENVIRONMENT
FROM A FUSION REACTOR POWER PLANT

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ABSTRACT

This report summarizes the expected tritium and activation-product inventories and presents an assessment of the potential radiological releases from a fusion reactor power plant, hypothetically located at the Oak Ridge National Laboratory. Routine tritium releases and the resulting dose assessment are discussed. Uncertainties associated with the conversion of tritium gas to tritium oxide and the global tritium cycling are evaluated. The difficulties of estimating releases of activated materials and the subsequent dose commitment are reviewed.

INTRODUCTION

The development of fusion reactors to produce electrical power continues to have major importance in the plans of the Department of Energy. As the engineering aspects of these reactors proceed, so must the awareness of health and environmental problems likely to be encountered. While designs of fusion reactors continue to change so rapidly that such impacts can not be estimated accurately, anticipated problem areas can be identified, and this is the intent of the study presented here. This paper identifies the major radioisotopes that will be produced in a fusion reactor power plant, the quantities involved, the possibilities for release, and the associated radiological dose evaluations. The uncertainties involved in each of these areas are highlighted.

SUMMARY

Radioisotope Inventories

• There will be 10^7-10^8 Ci of tritium in a typical fusion reactor power plant.

• Large amounts (several thousand Ci/MW_t) of activation products will also be produced and be present in components of these fusion devices.

Routine Releases

• The only radioactive substance that will be released in measurable quantities during routine operation of a fusion reactor power plant is tritium.

• Most reference designs predict the release of tritium to be 1 to 10 Ci/day; this amount is calculated assuming the use of proposed tritium barriers on the steam generators.

• For a continuous tritium oxide release of 1 Ci/day from a hypothetical fusion reactor located at the Oak Ridge National Laboratory (ORNL) site, the resultant whole-body dose is 3×10^{-3} mrem/yr to the closest resident.

• When using a global hydrologic model, it has been shown that of all possible discharge modes, release of tritium into surface waters results in the highest percentage of tritium uptake in man.

Uncertainties in Calculations of Dose from Tritium

• The uncertainty of radiological dose due to oxidation and exchange is relatively unimportant in the global model.

• In estimating over-all man-rem for a release of tritium, the Gaussian model predicts an 80-km radius dose commitment from a point source to be 40 times greater than the prediction of a more uniform release of an equivalent amount of tritium via the global model; specific regional model adaptation is needed.

• The individual dose uncertainty is due to: (1) Source term---what are the relative fractions of HT and HTO released from the reactor site to the atmosphere; and (2) HTO/HT risk factor---will doses to the dead skin layer continue to be considered the most important pathway, or will absorption and internal irradiation take a more important role in determining the dose resulting from an exposure to tritium gas.

  - The first uncertainty is very large with the possible fraction of tritium as water vapor varying from just above 0% to nearly 100%.

  - The second uncertainty may be as much as 15,000/200= 75 (Data of Pinson and Langham vs ICRP recommendation) on the side of over estimation if external irradiation is not an important pathway for exposure from HT. If ICRP recommendations are followed, then whenever the HTO/HT fraction is greater than 0.005 (1/200), the dose from HTO is the controlling factor. On the other hand, if the skin dose is unimportant relative to internal dose, then when the HTO/HT fraction is greater than 0.00007 (1/15,000), the dose from HTO is the controlling factor.

Accidental Releases

- Accidents have the potential to release large quantities of tritium and some smaller quantities of activation products.

- Malfunction of the magnets, loss-of-coolant, steam generator leak or break, and lithium fires are among the accident possibilities.

- Lithium fires probably create the greatest accident hazard for a fusion power plant. Many uncertainties exist in estimating the release of corrosion products that are present in the lithium during combustion.

- Most of the corrosion products are nonvolatile and in the event of a fire would be expected to "plate out." Some materials, such as V_2O_5 and MoO_3, are volatile at expected temperatures; however, formation of oxygen-containing corrosion products is not too likely since Li_2O is even more stable than Na_2O.

- If all 3×10^7 Ci of tritium contained in the UWMAK-I reactor blanket and coolant is released as the oxide at a height of 50 m, the resultant dose commitment to the nearest resident to ORNL would be 1 rem.

- A hazard evaluation of the activated structural material, shows that ^{60}Co appears to be the worst isotope found in the 316 stainless steel structure; ^{89}Sr for the Nb-1% Zr structure; and [45]Ca for the V-20% Ti structure.

Uncertainties in the Evaluation of the Consequences of Accidental Releases for Liquid Lithium Systems

- Many uncertainties exist in the thermodynamic conditions of the lithium system.

- Behavior of the lithium during a fire.

- Percentage of the lithium burned during a fire.

- The temperature variation with depth of liquid lithium during a fire.

● Mechanisms by which activated corrosion products may escape during a fire are complex and not well understood. This includes: corrosion product species, generation, transport, distribution in coolant, and particle deposition and removal.

RADIOISOTOPE INVENTORIES

Tritium

Various assessments and discussions of fusion power have previously been published by others,[1-8] and in all cases deuterium and tritium are assumed to be the fuel for the first generation of fusion reactors. These fuels will be used in the reaction:

$$D + T \rightarrow {}^4He(3.5 \text{ MeV}) + n(14.1 \text{ MeV}) . \tag{1}$$
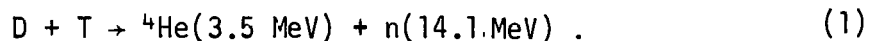
Large quantities of tritium will, therefore, be present in various parts of a magnetic fusion reactor (MFR), including the fuel storage and processing system, the fuel injection system, the plasma recovery system, and the breeding blanket and reprocessing systems. In addition to the operational inventory, the reactor will contain up to 10 kg of tritium in storage per 1000 MW_e. Tritium inventories for various initial reactor designs are presented in Table 1.[9,10]

An important aspect of tritium fuel is that only the initial fuel charge will require being transported to the MFR plant. After start-up, only the nonradioactive materials, deuterium and lithium, are required to fuel the plant. Subsequent tritium fuel, which is produced through neutron capture in deuterium and lithium, will be handled in a closed cycle at the power plant.

Activation Products

During normal operation of a fusion reactor, tritium will most likely be the major radionuclide present in reactor effluents and the working environment. Under abnormal conditions, during major maintenance periods, or after normal shutdown and subsequent dismantling, other

Table 1. Important parameters of various original reference designs

	ORNL[a]	PPPL[b]	UWMAK-I[c]	θ-Pinch[d]	Mirror[e]
Net Power (MW$_e$)	520	2000	1500	4100	200
Tritium Inventory in Blanket and Coolant (Ci)	3.8 x 10^6	3.8 x 10^6	3.8 x 10^7	9.3 x 10^6	9.6 x 10^6
Tritium Inventory in Plant (Ci)	5.8 x 10^7	5.4 x 10^7	(9.4-14) x 10^7	5.7 x 10^7	1.9 x 10^7
Primary Coolant	Li	He	Li	Li	Li
Structure Material	Nb - 1% Zr	PE - 16[f]	316 SS[g]	Nb - 1% Zr	SS
Breeding Material	Li	Flibe[h]	Li	Li	Li
14-MeV Neutron Wall Load (MW/m^2)	0.4	1.8	1.3	1.9	1.9
Total Flux Density (n/cm^2-sec)	1.5 x 10^{14}	8.7 x 10^{14}	4.0 x 10^{14}	2.7 x 10^{14}	5.2 x 10^{14}
Total Fluence at Replacement (n/cm^2)	3.0 x 10^{22}	1.0 x 10^{23}	3.0 x 10^{22}	1.7 x 10^{23}	Unknown
Replacement Interval (yrs)	10	5	2	2	NA

[a]Oak Ridge National Laboratory reference design.

[b]Princeton Plasma Physics Laboratory reference design.

[c]University of Wisconsin Tokamak reference design.

[d]Argonne National Laboratory - Los Alamos Scientific Laboratory Theta-Pinch reference design.

[e]Lawrence Livermore Laboratory reference design.

[f]Nimonic PE-16 approximate composition: 43% Ni, 35% Fe, 17% Cr, 3% Mo, 1% Ti, 1% Al.

[g]Type 316 stainless steel approximate composition: 62% Fe, 1% Si, 18% Cr, 14% Ni, 3% Mo, 2% Mn.

[h]FLIBE is a eutectic mixture of the salts LiF and BeF_2.

radioisotopes may be of major concern. Activation products are created from the transmutation of structural and containment materials by the 14 MeV neutrons emitted in the deuterium-tritium (D-T) reaction (See Eq. 1). A list of proposed materials that will be irradiated for the initial reference designs is presented in Table 1 along with the calculated neutron fluences at the initial containment wall.[10] In addition to alteration of the mechanical and physical properties caused by transmutations, the inner wall materials will also experience swelling due to the production of He and H_2 gases in the metals[11] and will require periodic replacement. Structural design and material selection will dictate the frequency for replacement; estimated replacement intervals for different MFR schemes range from 2 to 10 years (see Table 1).[10]

The materials used in a fusion reactor blanket will obviously dictate the types and amounts of activated material that are produced. To demonstrate this fact, the radioactivity associated with various blankets for a 10-year operating period[11-13] of the UWMAK-I reference design is shown in Table 2.

The UWMAK-I was chosen for this analysis, since it is a conceptually well-advanced tokamak reactor, and the tokamak is believed by many to be the first MFR concept that will achieve commercial operation.[14] The UWMAK-I is a 5000-MW_t device with a lithium-cooled blanket utilizing Type 316 stainless steel as the first wall and the structure in the blanket. While no existing materials are thought to be able to tolerate the high neutron-wall loading assumed for the UWMAK-I design, stainless steel is presently considered to be the most promising choice for use in the initial demonstration phase of commercial fusion

Table 2. Major radioactive isotopes in UWMAK-I
for the materials considered

Blanket	Radioisotope	$T_{1/2}$	Activity at Shutdown (Ci/kW_t)
316 Stainless Steel	^{55}Fe	2.60 yr	140
	^{56}Mn	2.58 hr	42
	^{54}Mn	303 day	24
	^{57}Co	270 day	10
	^{58}Co	71.3 day	29
	^{60}Co	5.26 yr	4.7
	^{51}Cr	27.8 day	26
	^{57}Ni	1.5 day	1.1
	^{49}V	330 day	0.67
Nb-1% Zr	^{92m}Nb	10.16 day	154
	^{93m}Nb	13.6 yr	3.1
	^{95m}Nb	3.75 day	50
	^{95}Nb	35 day	41
	^{89}Sr	3.27 day	37.7
	^{89}Zr	52.7 day	1.01
	^{90}Y	2.67 day	3.42
V-20% Ti	^{45}Ca	165 day	2.48
	^{46}Sc	83.9 day	1.87
	^{47}Sc	3.43 day	1.58
	^{48}Sc	1.83 day	12.1
	^{45}Ti	3.09 hr	0.12

power development.[15] Two alternate structural materials, Nb-1% Zr and V-20% Ti, are also analyzed as they are considered to be the two primary back-up systems. Molybdenum and aluminum alloys have also been suggested for use as structural material, but presently they are not considered primary candidates for first-generation fusion reactors.[13]

It is calculated that over 80% of the radioactivity in the 316 stainless steel results from five isotopes: ^{55}Fe, ^{56}Mn, ^{58}Co, ^{51}Cr, and ^{54}Mn. The decay of the radioactivity in the stainless steel blanket is dominated by ^{55}Fe($T_{\frac{1}{2}}$ = 2.6 yr) and ^{60}Co($T_{\frac{1}{2}}$ = 5.3 yr). The Nb-1% Zr blanket produces many unique radionuclides of niobium, and this blanket has the largest total amount of radioactivity. The V-20% Ti system contains the least amount of radioactivity; ^{45}Ca is the only relatively long-lived isotope in this blanket.

The radioactivity in these studied blankets as a function of time of operation is shown in Fig. 1.[11] A common feature of the activated blankets is that the total inventory of radioactivity increases rapidly after startup. Decay of the radioactivity following a shutdown is given in Fig. 2;[11] the predominant feature is the relatively slow decay with time. The Nb-1% Zr blanket presents the greatest problem with long-term radioactive waste; the principal contributor being ^{94}Nb($T_{\frac{1}{2}} \cong$ 20,000 yr).

ORNL-DWG 77-17731

RADIOACTIVITY (10^9 Ci)

V-20% Ti + SS
Nb-1% Zr + SS
TYPE 316 STAINLESS STEEL

20

18

16

14

12

10

8

6

4

2

0

10min   1.67h   16.7h   1week   10week   2yr   20yr
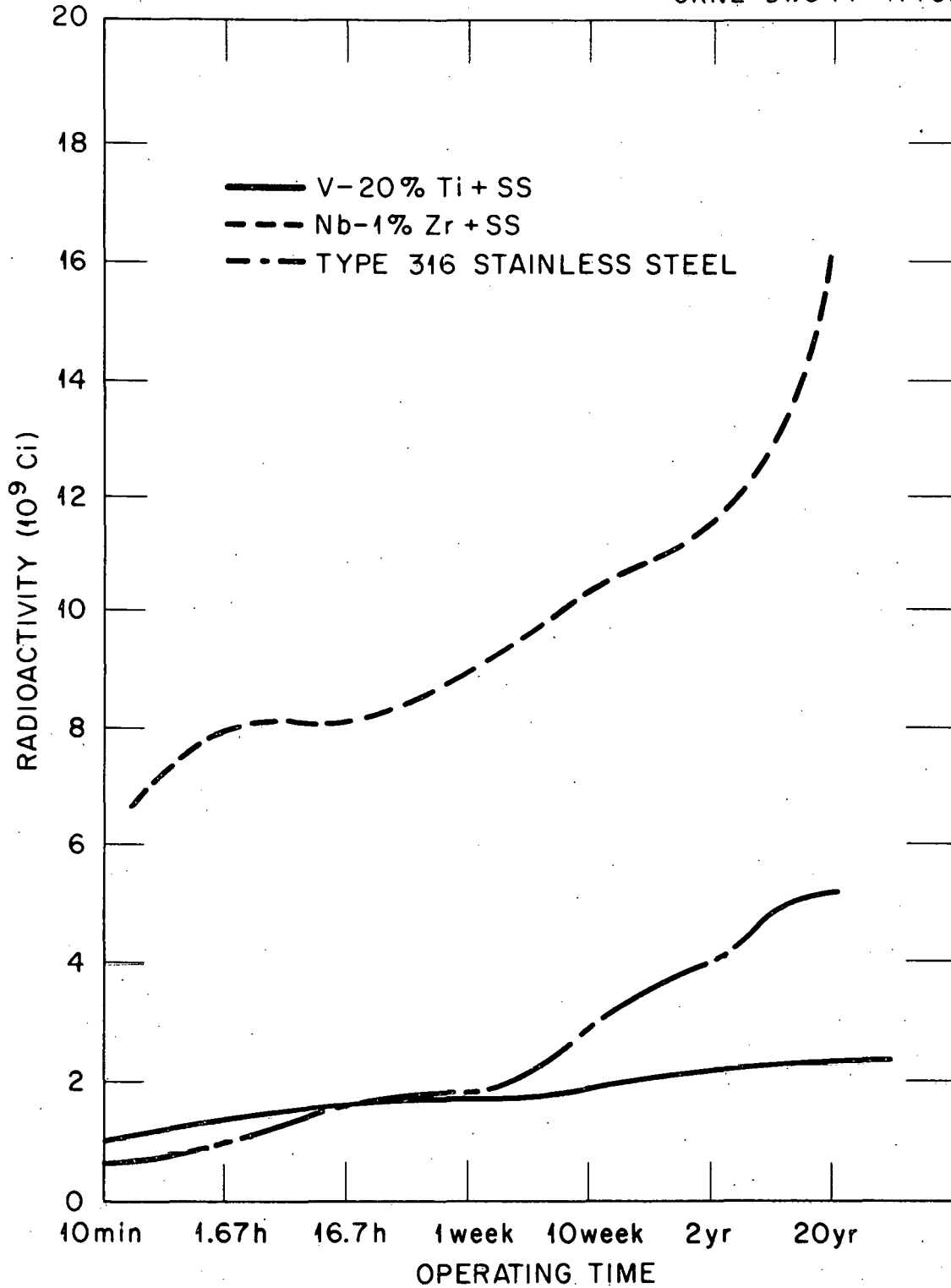
OPERATING TIME

Fig. 1. Radioactivity at shutdown in UWMAK-I for
Type 316 stainless steel, Nb-1% Zr, and
V-20% Ti systems vs operating time at
5000 MW_t.

ORNL-DWG 77-17730
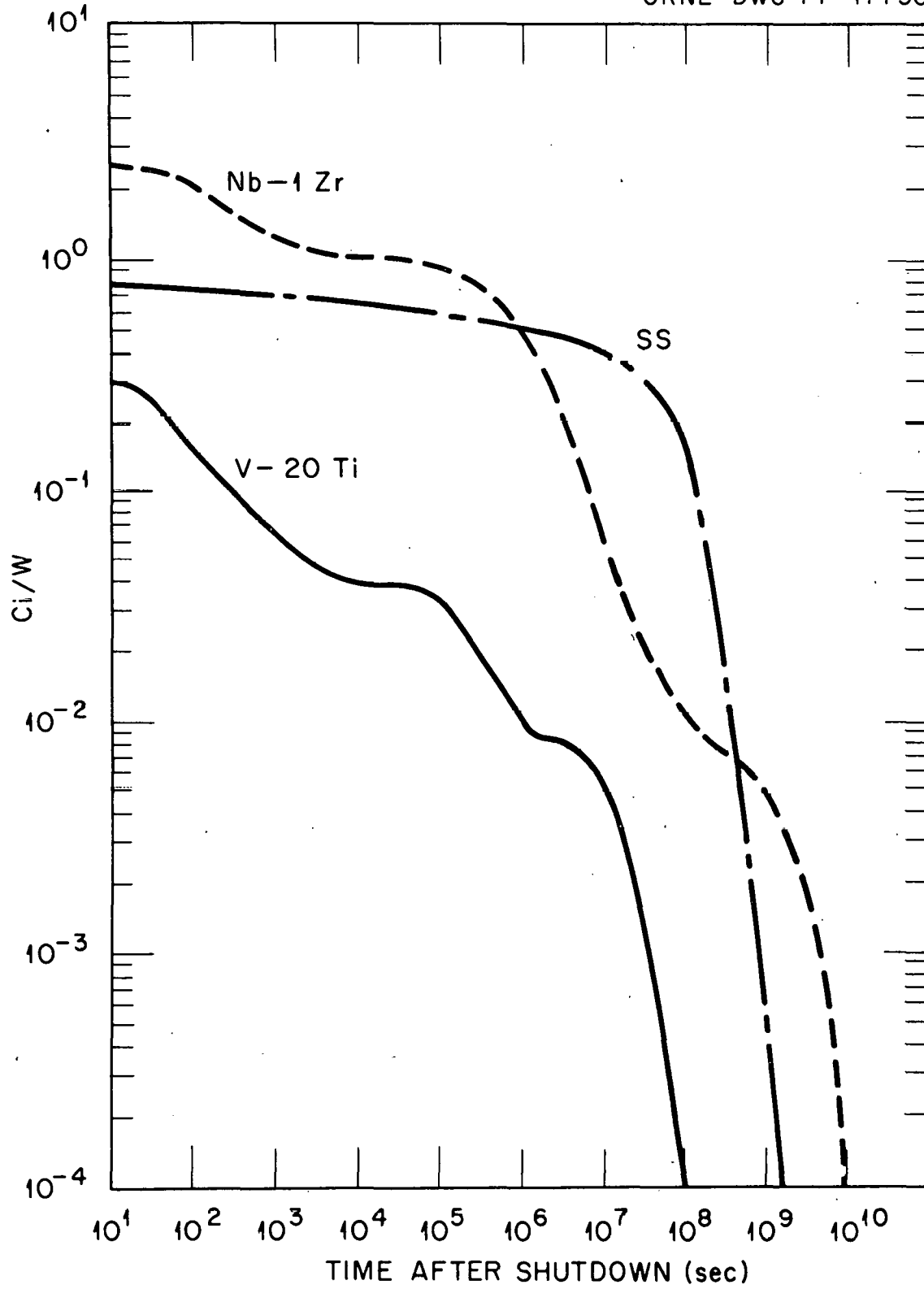
Fig. 2. Decay of radioactivity following shutdown of UWMAK-I
for a 10-year operating time.

ROUTINE RELEASES

The only radioactive substance that will be released in measurable quantities during routine operation of a fusion reactor power plant is tritium. Activation products could be released in extremely small quantities as corrosion products in the liquid effluents, although these are not expected to be significant. Being an isotope of hydrogen, tritium has a high permeability and is extremely difficult to contain at high temperatures. The two major release points of tritium are: (1) outward through all reactor materials into the reactor hall and (2) from the coolant system into the steam system.[16] In some designs, the main source of tritium contamination in the working environment will be the leakage of tritium from the coolant through turbine shaft seals and casings and from the piping for the coolant. Successful containment of tritium will likely require that most of these components be enclosed within a secondary containment, which would be maintained under an inert atmosphere or at a negative pressure (the vacuum building concept is an extension of this) to minimize any tritium escape. Almost all of the tritium diffusing into the steam system will be oxidized, and small amounts of the tritiated water will be lost to the environment. Since release rates through the steam system boundaries are dependent on tritium partial pressures, the objective in minimizing the tritium release via this flow path is to limit the concentration or partial pressure of tritium in the primary coolant to a sufficiently low value. Excessive permeation of tritium could be prevented by using metallic or ceramic diffusion barriers.[13]

Draley and Greenberg[9] have made an extensive study of tritium release during the normal operation of a 470-MW$_e$ tokamak fusion reactor. In the calculations, tungsten was used on the steam generator as a tritium barrier. This barrier was calculated to reduce the release of tritium into the steam generator by a factor of 100. The final analysis showed 2.6 Ci/day being discharged into the environment, with 2.0 Ci/day being released to the air and 0.6 Ci/day being released to the cooling water. This level is compared to releases of 11 Ci/day for current pressurized-water reactors (PWRs) and 0.13 Ci/day for boiling-water reactors (BWRs), both normalized for a 470-MW$_e$ output. In addition, both PWRs and BWRs could be augmented by some 34 Ci/day at fuel reprocessing plants per net 470 MW$_e$ of fission plant output.[9]

Kabele et al.[17] have provided the tritium releases for the 5000-MW$_t$ UWMAK-I; this data is presented in Table 3. Several assumptions were made in this analysis: (1) no effort was made to "improve" the design; (2) tritium permeation rates in metals were assumed to be proportional to the square root of the tritium partial pressure; and (3) all tritium released to the containment building atmosphere was assumed to be oxidized to water, with 90% of the tritium being absorbed on molecular sieves and the remaining 10% being released to the environment. In Table 3, the largest tritium loss is through the steam cycle; however, this tritium is released to the atmosphere, as the UWMAK-I design contains wet-cooling towers. Table 3 does not include a value for tritium leakage from a fuel storage system; even though the actual quantity of tritium stored will be quite large ($\sim 10^7$ Ci per 1000 MW$_e$),[10] the releases should be small. It should be noted here that leakage rates of the reference designs will

Table 3. Tritium releases for UWMAK-I

Source	Release to Containment (Ci/day)	Release to Environment (Ci/day)
Blanket and Shield	1.1	0.1
Shield Cooling System	1.1	0.1
Piping Systems (Li + Na)	7.7	0.8
Heat Exchanges	0.05	0.005
Power Systems (Steam)	2.0	10.3
^{3}H Extraction System	17.5	1.8
Diverter System	1.6	0.2
Fuel Handling Systems	3.0	0.3
TOTAL	34	14

vary with time, as the tritium barriers and other parts of the design change; one such oxide barrier was calculated to reduce the loss rate through the steam generator by a factor of 100.[17]

At present, there are no generally applicable standards regulating the release of tritium to the environment other than Federal and Agreement State regulations governing occupational and public radiation exposures. Present regulations for light-water reactors limit the radioactivity released in gaseous effluents so that the dose equivalent to an individual in an unrestricted area does not exceed 5 mrem/yr to the whole body or more than 15 mrem/yr to the skin. For liquid effluents, the upper limit is 3 mrem/yr for the whole body and no more than 10 mrem/yr to any organ.[18] Adequate tritium instrumentation is available to monitor at these levels.[19]

Dose Assessment

Local Analysis - Using Oak Ridge National Laboratory (ORNL) as a hypothetical site for a fusion reactor, radiological dose commitments for unit releases of tritium have been calculated. A computer program[20] was used to calculate ground-level air concentrations of ^3H. The diffusion equations used for these calculations are based on the Gaussian Plume model developed by Pasquill[21] and modified by Gifford.[22] The program corrects for plume depletion via radioactive decay and deposition. In a comparison of calculated versus measured values of ^{131}I at ORNL using this program, agreement was found, within a factor of 2, for distances between 3,000 and 13,000 m from the source.[23]

The meterological data for the routine releases were separated into two Pasquill stability classes, C and E, and were reduced to 8 discrete

wind speeds and the 16 cardinal directions (E, ENE, NE, ...); this data was the synthesis of over 62,000 individual measurements made at ORNL during the period 1960-68.[24] The population data was also divided into the 16 directions and along various distances measured radially; the population for the 80-km radius surrounding Oak Ridge National Laboratory is approximately 7.0×10^5 persons. The maximum vertical dispersion coefficient was 37.6 m, and the deposition rate for HTO was taken to be 1 cm/sec. The effective stack height was assumed to be 100 m. While calculations are made for distances of up to 80 km with the Gaussian plume model, it should be recalled that verification has only been performed for a small fraction of this distance and that numbers obtained at great distances using this model should be taken only "loosely."

For a continuous tritium oxide release of 1 Ci/day, the whole-body radiological dose commitment via the inhalation route for the 80-km radius was computed to be 2.4×10^{-2} man-rem. The closest resident to ORNL would receive a maximum dose via inhalation; this value was 7.0×10^{-4} mrem/yr. For the ingestion pathway, in which all foodstuffs consumed are assumed to be in equilibrium with the air concentrations of tritium, the 80-km radius cumulative and closest resident doses were estimated to be 1.7×10^{-3} man-rem and 2×10^{-3} mrem/yr, respectively. Further, if 1 Ci of ^3H is lost to the condenser coolant per day, with a daily flow of 2.5×10^6 m^3 once-through condenser (typical of a 1000-MW$_e$ facility), the water concentration would be 4×10^{-7} Ci/cm^3; this is five orders of magnitude less than the MPC$_W$ for uncontrolled areas. The dose from ingesting 2.2 liters of this condenser water per day is 2×10^{-2} mrem/yr. Estimates of downstream water flow rates and user populations

to the Gulf of Mexico give a cumulative dose commitment in the order of
0.1 man-rem. Thus, it appears that even if a fusion plant can not meet
the above low emission rates of a few curies per day, it could increase
discharges by factors of 10 to 100 without a significant potential dose
commitment to the local population.

Uncertainties in Tritium Dose Calculations - In all the above calculations,
tritium is assumed to be of the oxide form; in reality, some of the atmos-
pheric discharges may be as HT, resulting in much lower dose commitments.
If tritium is released in the gaseous form, however, the calculation of
dose becomes more complex. The problem of calculating dose resulting from
a release of HT arises from two areas of uncertainty: (1) the dose-con-
version factor to be used for an exposure to gaseous tritium and (2) the
rate of conversion of HT to HTO in the environment. The absorbed dose
resulting from an exposure to HTO appears to be well defined; however, the
effect from an exposure to HT is subject to debate. Legal concentration
guides, which are based on recommendations of the International Commission
on Radiological Protection,[25] show tritium as HTO a factor of 200 times
more hazardous than tritium as HT (an HTO/HT risk factor of 200). This
is obtained by comparing the HTO critical organ (total body) with the HT
critical organ (skin), though the tritium beta (18.6 keV maximum energy)
has a questionable effect on the dead skin layer. Pinson and Langham[26]
report that, after inhalation of HT and HTO, equivalent amounts of tritium
appear in the body fluids of man when the specific activity of HT in the
air inhaled was about 15,000 times greater than that of HTO.

Since tritium is substantially more hazardous as HTO than as
HT, the conversion rate of HT to HTO must be taken into account in

calculations for a release of HT. This rate has been experimentally measured in laboratory experiments for mixtures of high concentrations of tritium gas in: oxygen;[27] rare gases and oxygen;[28-29] dry air;[28] and water vapor with rare gases, oxygen or air.[28,29,30] Preliminary results at ORNL are available for dry and moist air with low T_2 concentrations, and theoretical studies to explain these laboratory results are also underway.[31] Actual empirical conversion rates in the environment, however, are yet lacking, although there is some indication that conversion by plants and soil organisms is rapid, perhaps in the order of days.[32]

A lower limit to the conversion rate for the atmosphere is difficult to determine, but an isotopic exchange rate of between three weeks and one year has been suggested by Doury,[33] who based his calculations on the data of Yang and Gevantman.[30] This data, however, was taken at high tritium concentrations where self-radiation is a predominant factor. Kummler[34] has estimated the atmospheric lifetime of molecular hydrogen to be near two years. An upper limit conversion rate of 10%/day was used by Otaduy et al.,[35] who attempted to determine the effect of the unknown conversion rate on the calculation of dose downwind from a gaseous tritium release. For an HTO/HT risk factor of 200, the relative risks due to releases of pure HT are compared for several conversion rates at typical meteorological conditions for the United States (see Fig. 3). The relative risk for the 0%/day conversion of HT in Fig. 3 is numerically equal to X/Q (the stack dilution factor) in sec/m^3, and numerical values of relative risk are related to dose by a constant factor. It is seen that for distances up to 5 km from the point of release, the difference
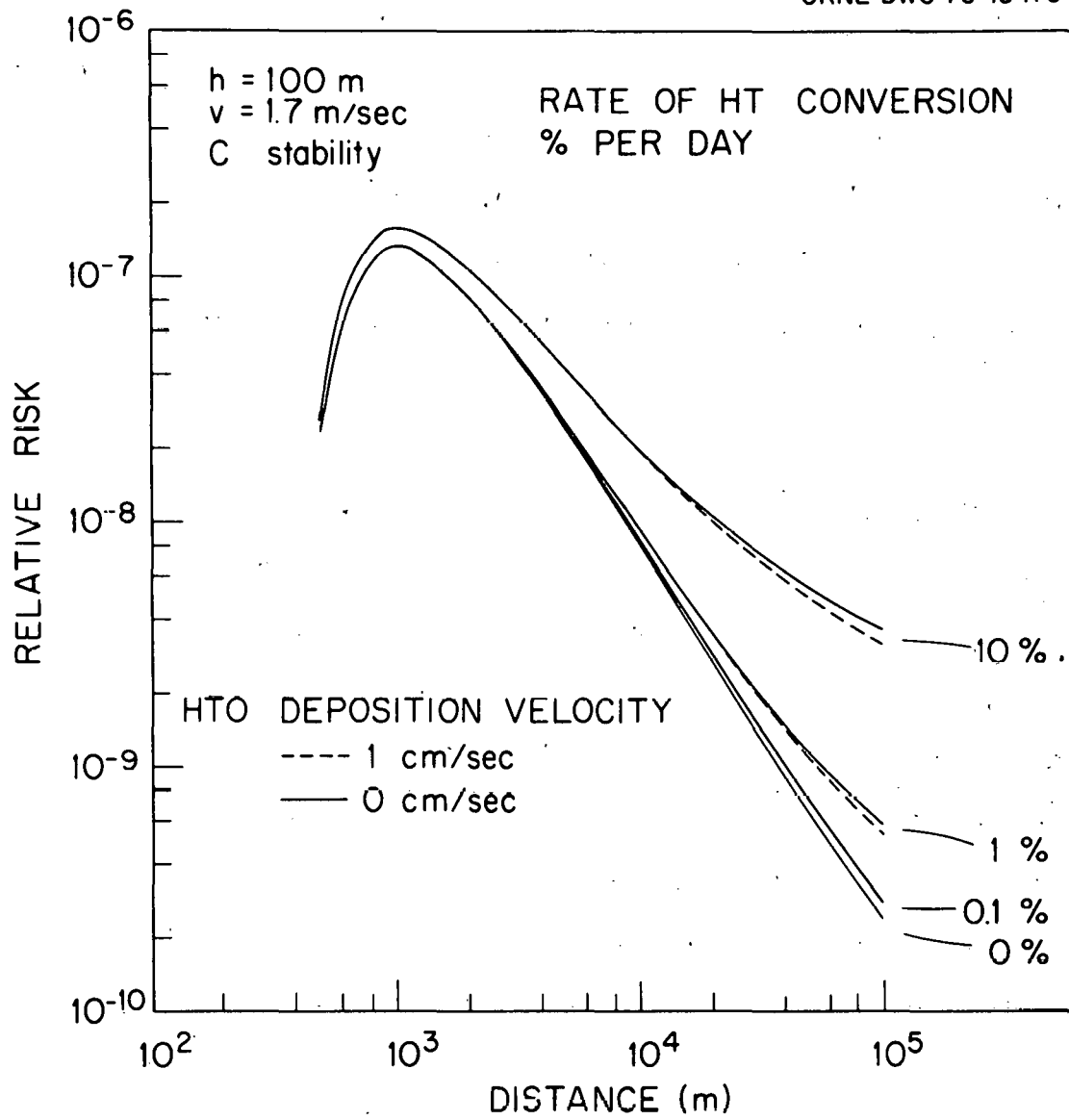
Fig. 3. Relative risk from HT and HTO versus distance for various conversion rates.

in risk between 10%/day conversion and no conversion is less than a fac-
tor of 14. For a low HTO/HT risk factor (200) and knowing the relative
HT and HTO release fractions, order-of-magnitude dose estimates can be
made even with inadequate knowledge of the conversion rate.

Global Analysis - Beyond the regional impact of a fusion reactor plant,
in a total fusion economy, the global implications must also be con-
sidered. Previously, Jacobs[36] reviewed the movement of tritium in the
various compartments of the environment and concluded that, in general,
the movement of tritiated water would follow rather closely the movement
of ordinary water. Thus, a generalized global hydrologic model developed
by Easterly and Jacobs[37] is used as a scheme to determine the fate of
tritium introduced into the environment.

A block diagram of the generalized global hydrologic cycle is
seen in Fig. 4. The model is described by a set of linear, homogeneous,
differential equations. The water compartments are assumed to mix uni-
formly and instantaneously and to interact with each other in a linear
manner. The volumes of the compartments and the transfer coefficients
for the model were taken from values in the literature.

It was shown that if tritium is continuously injected into sur-
face waters, at equilibrium, 2.0×10^{-5}% ends up in man's compartment.
Of all the compartments to which HTO could be released, discharge to the
surface waters resulted in the highest percentage of HTO in man's com-
partment. For a continuous discharge of tritium into the atmosphere, 1.7
$\times 10^{-6}$% of the tritium ends up in man. Injection of tritium into the deep
ocean proved to be the least hazardous. A continuous release of 1 Ci/day
to the atmosphere, gives an average dose of 1×10^{-10} mrem/yr; using a
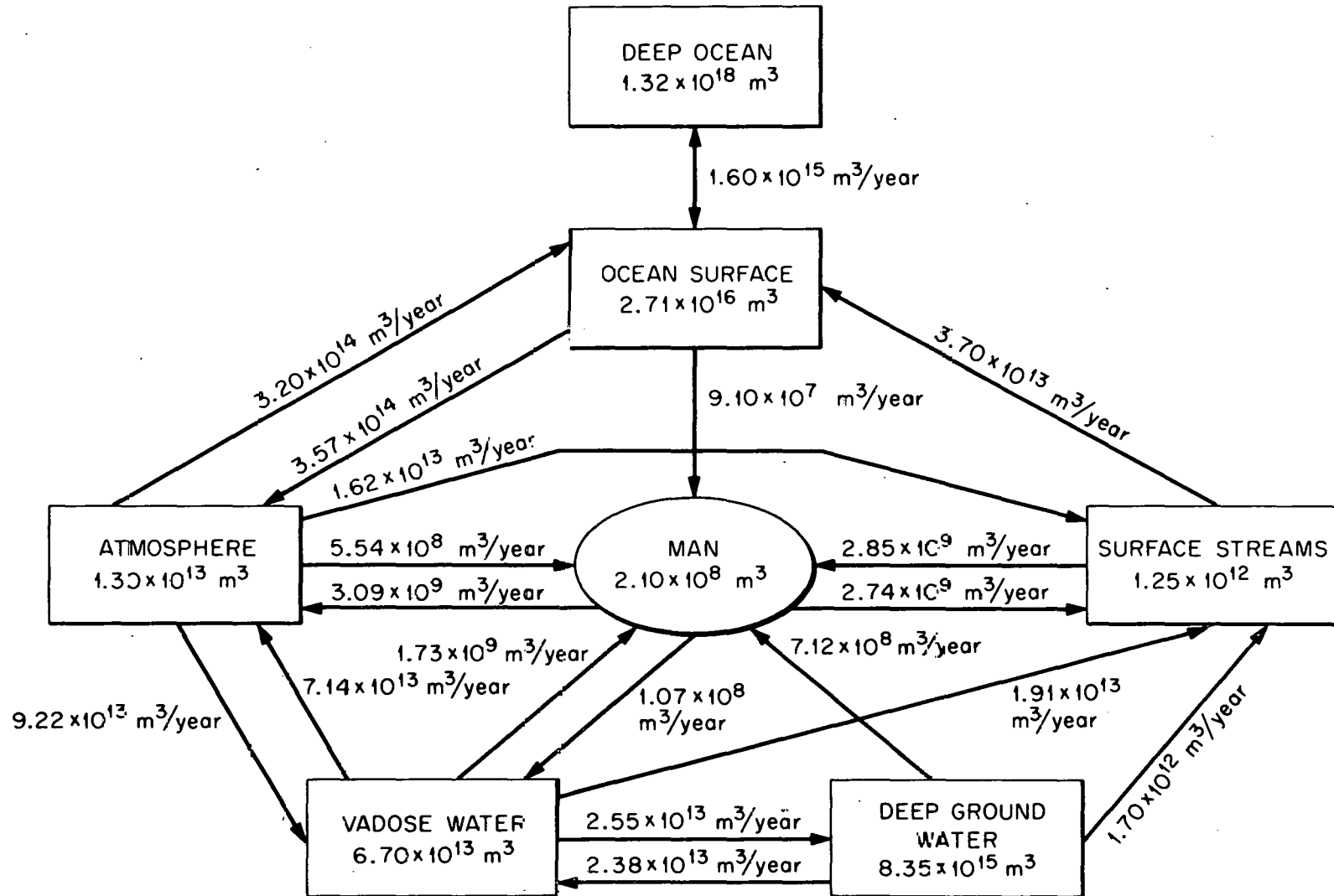
Fig. 4. Global hydrologic cycle with respect to man.

global population of 5 x 10^9, this computes to be 0.5 x 10^{-3} man-rem/yr. Similarly, for a continuous 1 Ci/day release into the surface waters, the resulting average whole-body and global doses are 1 x 10^{-9} mrem/yr and 0.5 x 10^{-2} man-rem/yr, respectively. It is likely that this latter value is lower than would be experienced since the actual situation which will prevail is one of a more concentrated use of the surface waters near the sources of tritium than the uniform model assumes. In the case where explicit information is available, as in the present study, direct estimates are preferable.

ACCIDENTAL RELEASES

In addition to the evaluation of routine consequences of low-level reactor emissions, accidents and events of low probability but potentially serious consequences must be evaluated. These accidents have the potential to release large quantities of tritium and some activation products. Such events include accidents owing to human error and/or mechanical malfunction internal to the nuclear operation, externally generated accidents (aircraft impact, natural disasters, etc.), and aggressive acts such as sabotage or acts of war. No large-scale uncontrolled nuclear reaction is possible because of the low fuel density present in the plasma, yet several causes of accidents are envisioned. The magnetic field could increase due to faulty control; this in turn would compress the plasma and increase the reaction rate, resulting in an increase in the blanket temperature. The extent of such damage can be estimated by observing that the adiabatic addition of half of the energy contained in 2 g of fuel would increase the total blanket temperature by a calculated 100°C.[9] Further, if a loss of refrigerant caused the superconducting magnets to go "normal," a large quantity of stored electrical energy (approximately 3×10^8 joules) might be released suddenly. Hall et al.,[38] state that if all the magnetically stored energy is released suddenly, an explosion would occur which would almost certainly destroy the containment structure. This then could possibly lead to a release of the tritium and fragment or vaporize some of the activated structural materials. However, proper design should limit the energy release to an acceptable value.[9]

An accident scenario important for fusion reactors, as well as fission reactors, is the loss-of-coolant accident. Calculations for the

UWMAK-I design indicate that complete loss of coolant flow during thermonuclear burn would cause the first wall to reach a temperature of 600°C in about 10 sec. Embrittlement from formation of helium bubbles in the metal, which occurs around 630°C could then lead to failure of the wall, release of lithium into the vacuum chamber, and consequent quenching of the fusion reaction.[39,40] It is noted here that these accident pathways would be disruptive and expensive but not catastrophic unless secondary effects produce a major fire and/or breach of containment.

Finally, the rupture of a potassium-to-water steam heat exchanger poses a special hazard. Massive injection of water or steam into molten or gaseous potassium would result in rapid reactions producing hydrogen and solid corrosive materials. Serious hydrogen explosions and/or fires might cause a loss of system security and the subsequent release of the tritium contained in the blanket structure. Hot molten lithium would ignite spontaneously on contact with air resulting in a serious fire.[38] Lithium reacts vigorously and exothermally both with air and with water; it also reacts with concrete (actually, with the water liberated from concrete by endothermic dehydration). Although the kinetics of these reactions are not well established, the calculated maximum flame temperatures for both the lithium-air and lithium-concrete reactions are in the range of 2,400 to 2,500°K. These temperatures are below the melting points of refractory metals, such as niobium, but above the melting points of other potential structural materials, such as the stainless steels.[39] The actual temperature to which a structural member could be raised depends on the geometry and heat transfer parameters, and design conditions are not expected to allow these structural members to approach flame temperatures.

Some metals are known to ignite in air at temperatures below
which they would otherwise maintain structural safety; others do not ig-
nite in air; and still others melt prior to ignition. Conditions of
forced air flow are required for the ignition of 1-cm niobium cyclinders
at 1500°C.[40] Other metals tested with a higher surface to volume ratio
were ignited in air (e.g., mild steel and some titanium alloys) or in
oxygen (e.g., SS 430 and molybdenum) at temperatures below their melting
points.[41] The oxides of some metals, notably vanadium and molybdenum,
have melting points considerably reduced (to around 650-800°C) from that
of the pure metals, and volatilization is more easily obtained.[41]

Lithium Analysis

Lithium fires create the greatest hazard from a MFR. As stated
previously, when lithium comes into contact with oxygen, the molten lith-
ium will burn, and the reaction may be violent or explosive. The combus-
tion of the lithium contaminated with corrosion products will result in
the release of aerosols containing these products. In the UWMAK-I design,
it is highly unlikely that greater than 1% of the Li could be completely
consumed in a fire because of the inert gases in the Li floor of the
reactor building and the vacuum around the reactor.[12] The formation of cor-
rosion product oxides is expected to have low probability in a lithium
fire since Li_2O is even more stable than Na_2O. In order to evaluate the
consequences of such emergency situations, it is necessary to know the
behavior of lithium, which is quite variable depending on many parameters.
Nonuniform temperatures varying with depth of the liquid causes a varia-
tion in release rates of radioactive materials within the liquid, and
high temperatures during burning may lead to high levels of crud bursts

of suspended solids. Mechanisms by which activated corrosion products escape during a fire are complex and not well understood.

One important factor in determining the release of radioactive materials during a lithium fire in MFRs is determining the distribution of these materials throughout the burning coolant. Lithium slowly cor- rodes the metallic surfaces of the structural materials of an MFR which are radioactive as a result of neutron activation. Radioactive corrosion products may remain in solution in the lithium or may be deposited on surfaces far removed from the site of activation. Many uncertainties exist with the thermodynamic conditions of the lithium systems which cause release and dissolution of these materials. These uncertainties make the predictability of the types of corrosion-product activity and the distribution of these radioisotopes throughout the coolant difficult.

The general technology of lithium is not as well known as that of sodium.[42] Some of the earlier observations of the corrosion rates of metals in lithium are of doubtful value, because the chemistry of the systems was poorly controlled, and more accurate solubility data of var- ious impurities in a lithium system are needed. For example, corrosion rates of vanadium are 110 times greater at $600^{\circ}C$ in sodium which contained 5 to 15 ppm oxygen than those in sodium which contained < 1 ppm oxygen. Since much of the experimental data on corrosion of refractory metals by lithium were not adequately characterized with respect to nonmetallic impurities present, the information is only useful as qualitative guide- lines. Nevertheless, the information gained from experience with sodium and potassium can be used to estimate possible effects with Li.[43]

Some solubility data for several metals in lithium are available and are shown in Fig. 5.[44] For these metals, the equilibrium solubility differs by almost three orders of magnitude between nickel and molybdenum. This variation may increase the uncertainty in the predictions of associated corrosion products in the coolant. The presence of nitrogen in lithium nitride and the higher stability and solubility of other nonmetallic impurities in lithium cause incompatibilities with structural metals due to the reactions of these elements.[42] The effects of various amounts of lithium nitride on the corrosion resistance to 316 stainless steel is shown in Table 4.[45] Pure iron generally has been found to be quite resistant to attack by lithium, however, iron containing small amounts of carbon often is penetrated intergranularly by lithium in certain temperature ranges. The variation of corrosion rates of metals in lithium can be seen in Tables 5 and 6.[45] Although the temperatures of the data presented in Tables 4 to 6 are above the practical limits for many structural components, the data is instructive for observing the variations in solubility with several key parameters.

Corrosion products generated at the hot locations in a flowing system usually are transported and deposited at the colder locations.[12] These considerations may have a significant bearing on materials selection and plant design. In determining the corrosion-product inventory, the four essential factors are: corrosion-product generation, transport, deposition and removal. The relatively large first wall area in contrast with the primary coolant of a MFR amplifies the corrosion-product generation. As noted earlier, assessment of corrosion phenomena in a lithium-cooled MFR is compromised by the relatively small amount of pertinent
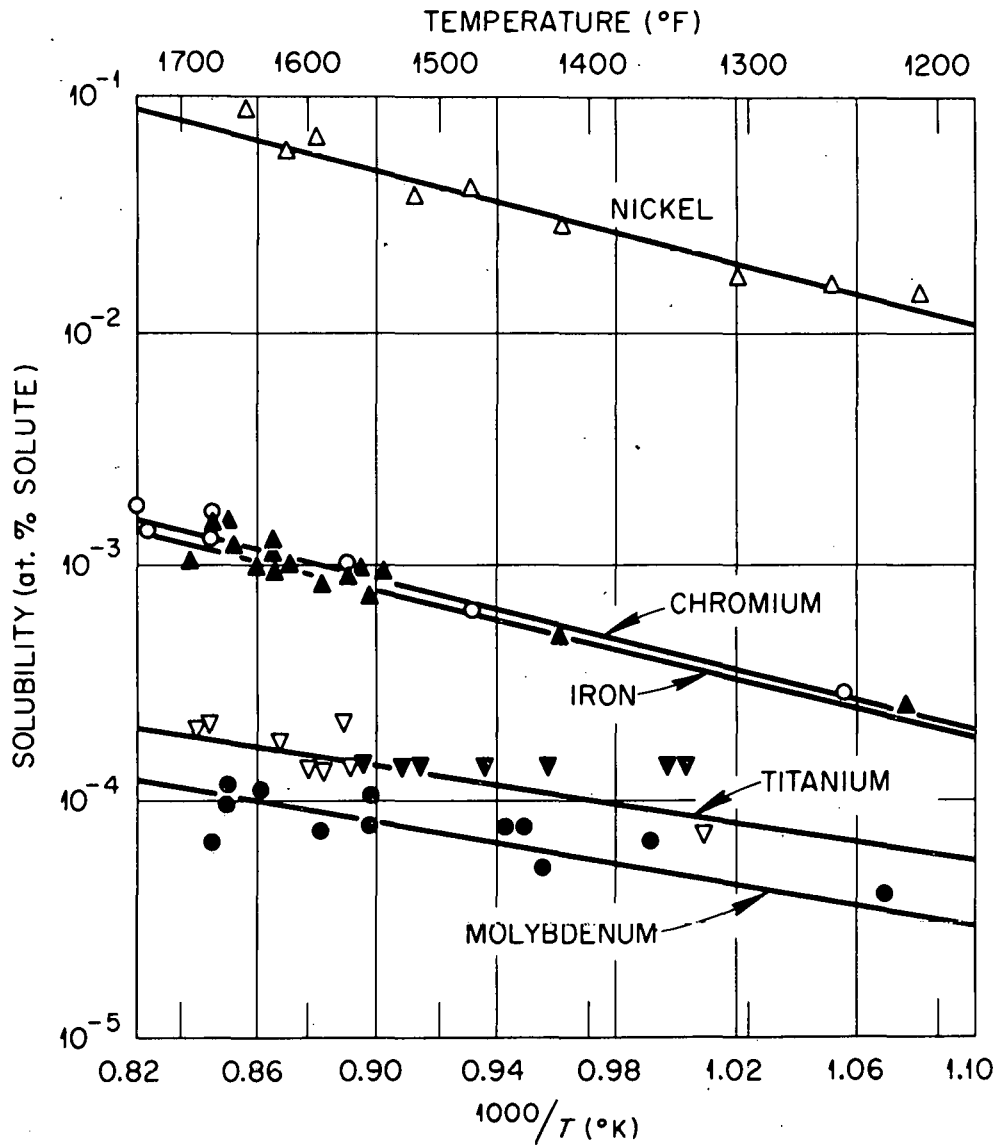
Fig. 5. Variation of the solubility of some metals in lithium with the reciprocal of the absolute temperature.

Table 4. Effect of additions of lithium nitride[a] to lithium[b]
on the corrosion resistance of Type 316 stainless steel[c]

Temperature °C	Lithium Nitride Addition to Lithium (Wt %)	Weight Change[d] (mg/cm^2)
816	0.5	+.37
816	1.0	-1.91
816	2.0	-0.16
871	0.1	0
871	0.25	-0.62
871	1.0	-2.3

[a]Lithium nitride analysis: 58.4% Li, 37.1% N_2, 1.5% CO_3 (% wt).

[b]Nitrogen content of lithium prior to addition of lithium nitride: 0.05% wt.

[c]Test conditions: 100 hr; surface area/volume = 13 cm.^{2}/cm.^{3}.

[d]The addition or subtraction of corrosion products to the stainless steel structure.

Table 5. Lithium corrosion on metals in two-component static
test systems

Metal	Temperature (°C)	Time (hr)	Surface Area Vol. of Li (cm^2/cm^3)	Weight Change (mg/cm^2)
Beryllium	816	100	10	+0.26
Beryllium	1000	100	10	+0.25
Iron[a]	816	100	13	0
Iron	816	400	7	-0.31
Iron	1000	400	7	-0.29
Molybdenum	816	100	13	0
Tantalum	816	100	13	+1.2
Titanium[a]	816	100	13	+0.38
Vanadium	816	100	13	+1.3
Zirconium	816	100	13	0

[a]Duplicate tests.

Table 6. Results of 100-hr lithium corrosion tests on iron-base alloys in two-component static test systems[a]

Alloy[b]	Nominal Composition (Wt %)	Weight Change (mg/cm^2)	
		816°C (100 hr)	1000°C (400 hr)
1035 Steel[c]	98 Fe - 0.35C	- 0.16	NA[d]
304 SS[c]	59 Fe - 19 Cr - 10 Ni - 0.08 C max.	0	- 1.9
309 SS[c]	62 Fe - 23 Cr - 13 Ni - 0.02 C max.	- 0.16	- 0.31
310 SS[c]	53 Fe - 25 Cr - 20 Ni - 0.25 C max.	- 0.16	NA
316 SS[c]	67 Fe - 17 Cr - 12 Ni - 2 Mo - 0.10 C max.	0	- 0.47
317 SS[c]	64 Fe - 19 Cr - 12 Ni - 3 Mo - 0.10 C max.	- 0.16	NA
330 SS	48 Fe - 15 Cr - 35 Ni - 0.25 C max.	- 0.78	NA
347 SS[c]	69 Fe - 18 Cr - 10 Ni - 1 Nb - 0.08 C max.	0	NA
446	73 Fe - 25 Cr - 0.35 C max.	+ 0.47	NA
Incaloy[c]	42 Fe - 34 Ni - 21 Cr - 0.1 C max.	0	NA
Multimet[c]	32 Fe - 21 Cr - 20 Ni - 20 Co - 0.12 C max.	0	NA

[a]Surface area/volume of lithium = 13 cm^2/cm^3.

[b]SS = Stainless Steel.

[c]Duplicate tests.

[d]Not analyzed.

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data.[12] Much of the experimental data on corrosion of refractory metals by lithium are useful only as guides, since the materials were not adequately characterized with respect to nonmetallic and metallic impurities present.

Using the data of Gill et al.[46] for corrosion rates extended to an annual basis approximately 2500 kg/yr of corrosion products would be released to the primary circuit.[12] An additional 250 kg/yr would be contributed from radiation sputtering on the inside surface of the first wall.[12] Assuming ASTM (American Society for Testing and Materials) stainless steel, the radioactivity of this 2500 kg of corrosion products is estimated to be 4×10^6 Ci and the absorbed dose rate would be 1000-10,000 rad/hr in the coolant system.[12]

Dose Assessment

Tritium Impact - In an accident, large quantities of tritium could be released into the environment. In the accident dose assessment used in this paper, it is assumed that the following factors apply: The release is restricted to the NE sector (the most dominant wind direction at ORNL); "F" stability; and a deposition velocity of 1 cm/sec. For the activation products, an aerodynamic mean diameter of 1 μm is assumed and a ground-level elevation release is postulated.

At this time, it is still difficult to write accident scenarios for the release of tritium and HTO/HT fractions from blanket-cooling systems. Hence, the worst case is taken, and it is assumed that the tritium in the UWMAK-I reactor blanket and coolant, 3×10^7 Ci, is released in the oxide form into the environment. The radiological dose commitment to the nearest resident in the NE direction (8 km) is 5 rem to the whole body via inhalation for a ground release. Assuming a release height of

50 m (which may be more representative due to the strong upward convective action of the heat from the burning metals in an accident), the whole body dose equivalent is reduced to 1 rem. The 80-km man-rem dose, by either release height is $\sim 1 \times 10^4$ man-rem. Thus, even for this upper limit release, the dose commitments are calculated to be small relative to the natural background 80-km dose commitment of approximately 7.4×10^4 man-rem/yr. Further, it is difficult to imagine an accident such that all the tritium in the blanket and coolant would breach all the containment systems and be released into the environment.

Impact of Activation Products - Activation products which are not released during normal operations, except in extremely small quantities as corrosion products, have the potential to be released in an accident situation. As shown in Fig. 5, the different activation products have different solubilities in lithium, indicating different amounts of the various radionuclides will be in the coolant when it burns; other source term uncertainties were also discussed in the previous section of this paper. Therefore, for a "first-order" crude accident analysis, it is assumed that the major radioisotopes in the 316 SS, Nb-1% Zr, and V-20% Ti structures are all released during a fire at a rate of 1 Ci/sec for a period of 1 day. The whole-body dose at 8 km is calculated for each of these nuclides. The radioisotopes ^{57}Ni and ^{45}Ti are not included, as appropriate dose-conversion factors could not be found in the literature. Summaries of the analysis are presented in Table 7. In the first column are listed the major isotopes for each structure. The next column lists the relative quantity of these isotopes in the structure, where the numbers are multiples of the lowest quantity. The third column lists the

Table 7. Hazard assessment for the
316 SS, Nb-1% Zr, and V-20% Ti structures

Nuclide	Relative Quantity	Relative Whole-Body Dose	Hazard Index
		316 SS	
^{60}Co	7	2340	16,380
^{54}Mn	36	197	7,091
^{55}Fe	209	27	5,643
^{58}Co	43	120	5,160
^{57}Co	15	65	975
^{51}Cr	39	3	117
^{56}Mn	63	.1 (0.197 mrem)[a]	63
^{49}V	1 (0.67 Ci/kW$_t$)[a]	5	5
		Nb-1% Zr	
^{89}Sr	37	13	481
^{92m}Nb	152	2	304
^{95}Nb	41	5	205
^{95m}Nb	50	1 (1.99 mrem)[a]	50
^{93m}Nb	3	10	30
^{90}Y	3	2	6
^{89}Zr	1 (1.01 Ci/kW$_t$)[a]	2	2
		V-20% Ti	
^{45}Ca	2	39	78
^{48}Sc	8	5	40
^{46}Sc	1	36	36
^{47}Sc	1 (1.58 Ci/kW$_t$)[a]	1 (1.16 mrem)[a]	1

[a]Actual value.

whole-body inhalation dose, where again the numbers are multiples of the lowest quantity. The second and third columns are then multiplied together to arrive at a hazard index. This relative hazard index has the dimensions of $mrem/kW_t$, and the highest numbers signify greatest health hazard. The hazard index, however, assumes that there is no discrimination in the isotopes escaping into the environment. It is known that this is not completely correct as 316 SS will melt in a liquid-metal fire accident, whereas, the niobium and vanadium structures could possibly burn, giving oxides with distinct dispersion characteristics. Further study is needed to determine the effect of various containment schemes and plant countermeasures, the physical and chemical forms and fraction of the releases, the appropriate particle sizes, and the environmental transport to arrive at a more quantitative radiological assessment of an accidental release. A recent study[47] completed by the authors critically analyzes the metabolic and environmental aspect of niobium; similar in-depth studies need to be conducted for other elements, such as Ni and Ti. Since the behavior of most of the structural materials in accident conditions is largely a matter of speculation, considerable research in this area is required before anything quantitative can be said about source terms.

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