NUCLEAR ENGINEERING

MASSACHUSETTS INSTITUTE
OF TECHNOLOGY

APPLICATION OF PROBABILISTIC CONSEQUENCE ANALYSIS TO THE ASSESSMENT OF POTENTIAL RADIOLOGICAL HAZARDS OF FUSION REACTORS

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ABSTRACT

A methodology has been developed to provide system reliability criteria based on an assessment of the potential radiological hazards associated with a fusion reactor design and on hazard constraints which prevent fusion reactors from being more hazardous than light water reactors. The probabilistic consequence analyses, to determine the results of radioactivity releases, employed the consequence model developed to assess the risks associated with light water reactors for the Reactor Safety Study.

The calculational model was modified to handle the isotopes induced in the structural materials of two conceptual Tokamak reactor designs, UWMAK-I and UWMAK-III. Volatile oxidation of the first wall during a lithium fire appears to be a primary means of disrupting induced activity, and the molybdenum alloy, TZM (UWMAK-III), tends to be more susceptible than 316 stainless steel (UWMAK-I) to mobilization by this mechanism. It was determined that the radiological hazards associated with induced activity in these reactor designs imply reliability requirements comparable to those estimated for light water reactors. The consequences of estimated maximum possible releases of induced activity, however, are substantially less than the maximum light water reactor accident consequences.

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1. R. Sawdye, J. S. Sefcik and M. S. Kazimi, "Reliability Requirements for Admissible Radiological Hazards from Fusion Reactors," Trans. Am. Nucl. Soc. 27, 65-66, November 1977.

2. M. S. Kazimi et al., "Aspects of Environmental and Safety Analysis of Fusion Reactors," MITNE-212, Dept. Nucl. Eng., M.I.T., October 1977.

3. D. Dube, M. S. Kazimi and L. M. Lidsky, "Thermal Response of Fusion Reactor Containment to Lithium Fire," Paper presented at Third Top. Meeting on Fusion Reactor Technology, Santa Fe, N.M., May 1978.

4. R. W. Sawdye and M. S. Kazimi, "Application of Probabilistic Consequence Analysis to the Assessment of Potential Radiological Hazards of Fusion Reactors," MITNE-220, Dept. Nucl. Eng., M.I.T., July 1978.

5. D. A. Dube and M. S. Kazimi, "Analysis of Design Strategies for Mitigating the Consequences of Lithium Fire within Containment of Controlled Thermonuclear Reactors," MITNE-219, Dept. Nucl. Eng., M.I.T., July 1978.

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TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

I. INTRODUCTION

Decisions concerning the future methods of electricity production must include considerations of the potential environmental impacts and public safety hazards associated with each option. A comparison of the relative environmental and safety problems may influence the eventual choice of an energy resource, and recognition and assessment of these hazards may influence the development of resource technologies. Therefore, it is important to examine the environmental and safety aspects of fusion power reactors even though they are presently in an early and conceptual stage of development. There have been several recent publications concerned with this topic,[1-6] and interest in fusion reactor impacts will inevitably increase along with the viability of this technology.

Since a fusion reactor is a nuclear energy device, one of the main safety concerns involves radiological hazards. The radioactivity contained in a typical Tokamak fusion reactor facility will be in three basic forms:

1) tritium
2) induced activity in structural materials
3) activated corrosion products

The radiological hazard assessment must examine the operational and accidental impacts on the welfare of plant personnel and the public, associated with each form of radioactivity.

The routine operational hazards of radioactivity tend to be dealt with at an earlier stage in the design process, with accident analyses requiring more detailed total system designs. If a methodology could be developed to evaluate possible accidental hazards of conceptual fusion reactor designs, it might serve to influence the development and evolution of those system designs.

An actual risk assessment of fusion reactors would require detailed systems designs from which the possible accident sequences and their probabilities could be determined. This approach would employ event-tree and fault-tree analyses similar to those used in the Reactor Safety Study (WASH-1400)[7] to assess the public risks associated with commercial light water reactors in the United States. Since the available designs for large commercial fusion power reactors are conceptual in nature, determination of a comprehensive set of reactor accident sequences and probabilities may not be feasible. A reasonable alternative to the problem of hazard evaluation for fusion reactors may be to attempt to establish some overall system reliability requirements which will assure that the safety of fusion reactor designs will be at least as good as other forms of power generation, such as light water reactors. Thus, some type of maximum tolerable accident probabilities would be used to determine reliability criteria that may be utilized in the evolution of fusion reactor designs.

In this work, a methodology will be presented to determine system reliability criteria which should prevent the

potential radiological hazards of fusion reactors from exceeding those of commercial light water reactors. This methodology is explained in Chapter II and is demonstrated by an example in the following chapters. The methodology utilizes the consequence model developed for the Reactor Safety Study, the "Calculations of Reactor Accident Consequences" computer code (CRAC),[8] to calculate the probabilities of various consequence magnitudes following a given set of conditions for releases of radioactivity from a reactor containment. The example analysis examines the potential hazards of the induced radioactivity in structural materials of Tokamak fusion reactors. A similar approach may also be used to examine the potential hazards of tritium, the other major form of activity. The conclusions and recommendations of this study may be useful as guidelines in the continuing development of Tokamak reactor designs.

II. PROPOSED METHODOLOGY

The major steps in this methodology consist of the identification of the potential radioactive releases from a Tokamak reactor, the calculation of various consequences of these releases, determination of maximum tolerable accident probabilities which limit the potential hazards, and the establishment of minimum reliability requirements for system designs. Each of these steps is explained in the following sections of this chapter.

2.1 Identification of Release Accidents

To determine the possible releases of activity, the inventories of radioisotopes must be determined. For a Tokamak reactor, the two major inventories of radioactivity are the tritium and the induced activity in reactor structures. The tritium is bred in the reactor blanket and used to fuel the plasma. The induced activity results from the neutron activation of reactor structural materials. The induced activity appears to present a potential public hazard only during accident conditions, assuming the normal disposal of activated structures after their useful life in the reactor poses no significant problems. The tritium, however, presents a potential hazard following an accidental release and during the normal operation of the reactor due to its rapid diffusion through materials, particularly at high temperatures.[6] Both

forms of radioactivity may be important in an assessment of the public and occupational hazards of a Tokamak reactor facility, however, occupational and tritium hazards will not be specifically examined in this study.

For any radioactivity to be a threat to the health and welfare of the public, it must be released from the reactor and the containment structures. The activity must be in forms which can be dispersed from the reactor site into the environment. Accident sequences must be identified which are capable of producing these releases, and each of these event sequences can then be analyzed to determine the magnitude of the activity release and the conditions describing the physical release process. This part of the hazard analysis, however, is very dependent on the actual system design. Because of the early stage of fusion reactor conceptual design, the methodology followed in this work assumes that various fractions of the radioactive inventory can be released up to a limit determined by an analysis of an extreme accident. Hence, the identification of potential release accidents involves the determination of possible ranges for parameters used to calculate the radioactive dispersion and radiological consequences.

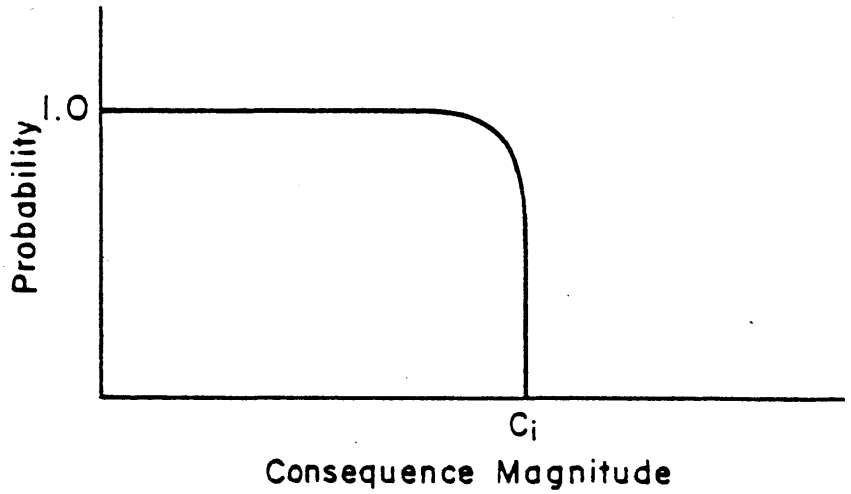
2.2 Generation of Consequence-Probability Functions

The next major step in this methodology is to utilize the consequence model to calculate the best estimates for the various effects or results of the identified release accidents. The CRAC code requires the input of radioactive inventories and other parameters describing the release accident, data

describing population distribution about the reactor site, weather conditions during the course of the accident, and dose conversion factors for the health effects of the exposure to various radioisotopes.[8] The code proceeds to calculate the probabilities of the magnitudes of various health effects and economic costs resulting from reactor accidents. These consequences are presented as a set of complementary consequence distribution functions, which give the probability that the magnitude of a specific consequence will exceed a particular value. Thus, the complementary function is essentially a probability density function integrated from an arbitrary value for the independent variable (consequence magnitude) to infinity.

The original purpose of this code was the generation of complementary cumulative distribution functions describing the public risk associated with light water reactors in the United States. These cumulative functions are essentially summations of individual reactor accident consequence-probability functions weighted by the probabilities associated with the specific accident conditions. For a specific set of site and weather conditions, a single given set of reactor accident parameters will result in a normalized or conditional complementary distribution function. An example of each type of function is shown in Figure 2.1. Therefore, a cumulative function is calculated by weighting each of the conditional functions by their associated site, weather and accident probabilities, then summing over the complete group of accidents.

A. CONDITIONAL COMPLEMENTARY PROBABILITY DISTRIBUTION FUNCTION



B. CUMULATIVE COMPLEMENTARY PROBABILITY DISTRIBUTION FUNCTION

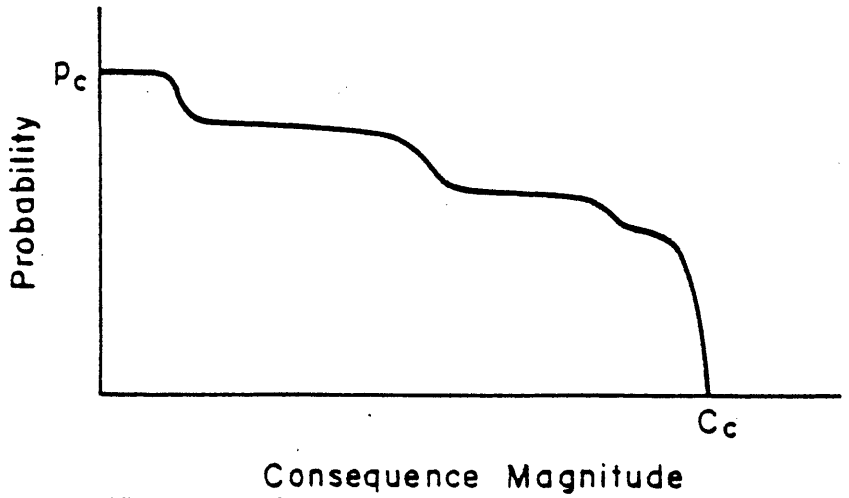


FIGURE 2.1 PROBABILITY DISTRIBUTION FUNCTIONS USED IN METHODOLOGY

For the present work, a set of cumulative functions are generated for light water reactors (a pressurized water reactor and a boiling water reactor) with specific site and weather conditions, and pertaining to the complete set of accidents derived in the Reactor Safety Study. These accidents represent the spectrum of light water reactor accidents with significant impact on public safety. The CRAC code is then used to generate conditional functions for a set of accident conditions assumed to represent a range of possible releases from a Tokamak reactor. The same site and weather conditions are used to avoid their variable effects on the results. These conditional functions are normalized to the assumed accident probability, which, for this methodology, is unity. These two types of functions are then used together to determine maximum tolerable probabilities for particular types of Tokamak reactor accidents under investigation.

2.3 Determination of Maximum Tolerable Accident Probabilities

The crucial step in this methodology is the determination of accident probability limits for Tokamak reactors. The basic technique consists of placing a light water reactor cumulative function and the corresponding (same consequence) Tokamak reactor conditional distribution function on the same graph. The cumulative curve is plotted according to its functional values, however, the conditional curve is placed below the cumulative curve and is shifted upwards along the probability scale (ordinate) until contact is made. Therefore, the conditional curve remains totally beneath the cumulative curve,

and the two curves representing the same health effect or economic cost for the two reactor types, with the same site and weather conditions, will have an appearance similar to Figure 2.2.

The ordinate intercept, P_i, of the curve representing the conditional distribution function, is the maximum tolerable probability for accidents with consequences exceeding those represented by the particular conditional function. For example, if a series of conditional functions are derived for a range of release accidents with varying fractions of activity inventory released, and the release fraction is the only variable parameter, a complementary probability distribution function can be generated to provide the maximum tolerable probability or frequency for accidents with releases exceeding various fractions of the inventory. Figure 2.3.a shows the graphical technique to obtain the intercept values for a series of conditional consequence distribution functions representing various release fractions, f_i. Figure 2.3.b shows the resulting complementary probability distribution function for the maximum tolerable accident probabilities. This methodology, therefore, provides maximum tolerable accident frequencies, or system failure rates, for release fractions which exceed various values. These limitations on system failure rates can be used to establish minimum system reliability requirements which, in effect, limit potential radiological hazards of Tokamak reactors to those determined for light water reactors.

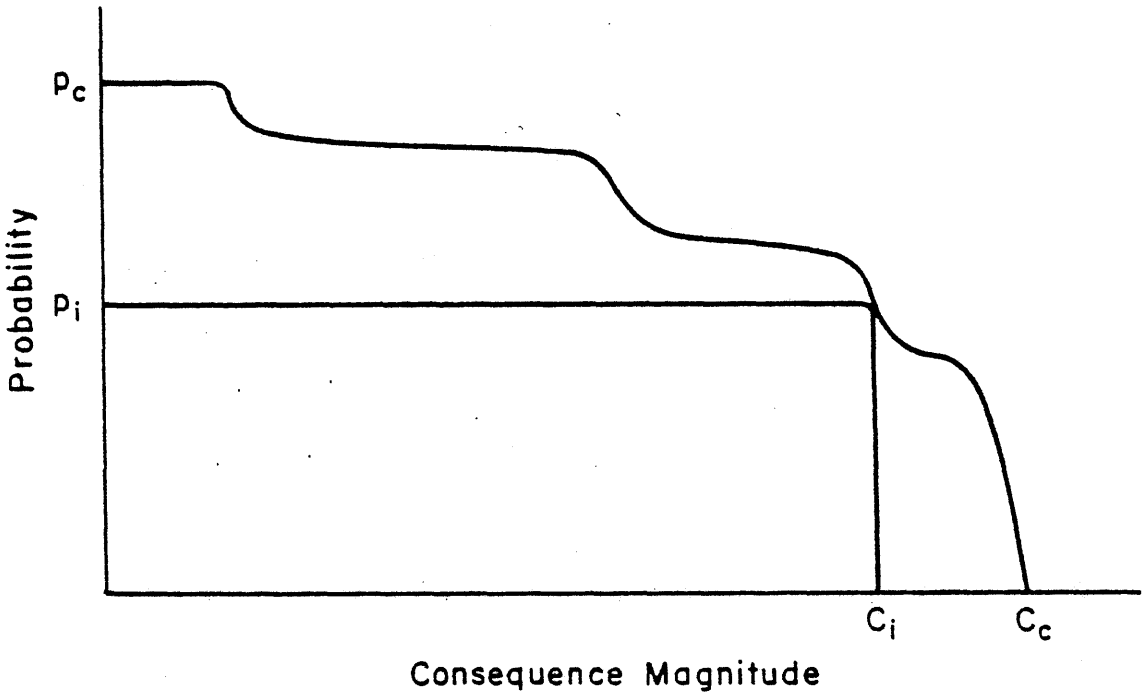
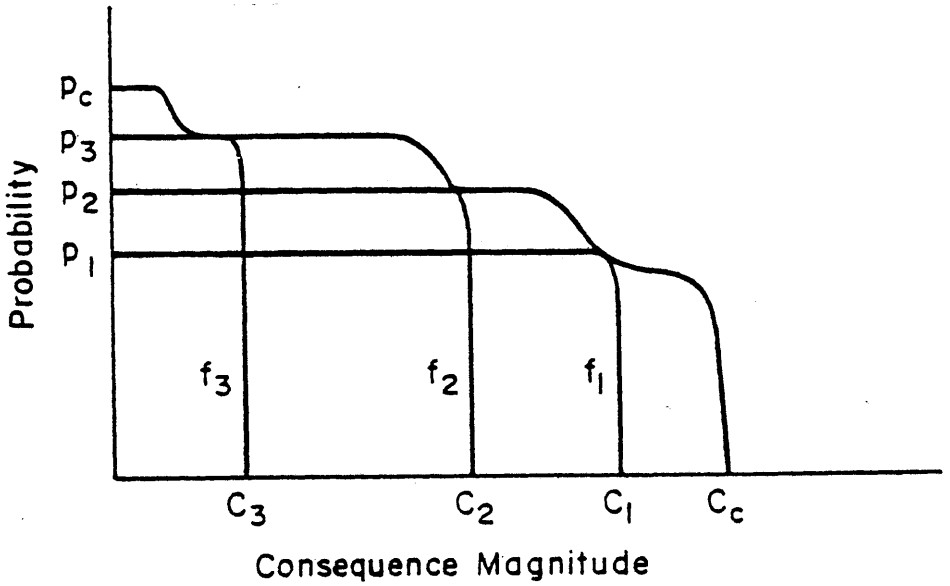


FIGURE 2.2 BASIC TECHNIQUE OF THE METHODOLOGY

A. GRAPHICAL TECHNIQUE WITH VARIABLE ACCIDENT PARAMETER - RELEASE FRACTION



B. RESULTS - MAXIMUM TOLERABLE PROBABILITY FOR RELEASES GREATER THAN OR EQUAL TO VARIOUS VALUES

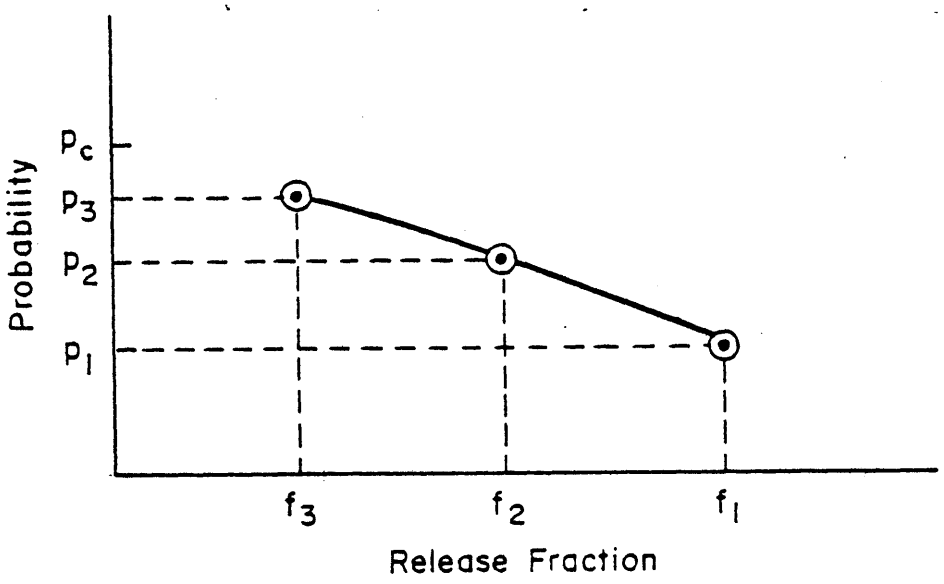


FIGURE 2.3 DETERMINATION OF PROBABILITY LIMITS

2.4 Establishment of Minimum System Reliability Requirements

If detailed system designs were available for Tokamak reactors, and event and fault-tree analyses were possible, an allowable accident probability or frequency of occurrence would have a direct correspondence to a product of estimated failure rates for sets of subsystems. Hence, reliability requirements for subsystems or even individual components might be directly inferred from these limitations on accident frequencies, since a minimum reliability requirement is basically an inverse of a maximum allowable failure rate or frequency. Since Tokamak reactor designs have not yet reached the required stage of development, the significance of the accident probabilities is in their use as overall system design guidelines.

The evolution of Tokamak reactor designs will involve decisions concerning required reliabilities for components and the necessity of various safety-related systems. The maximum tolerable radioactivity release probabilities can establish targets for system designs, reveal the relative importance of public and occupational hazards, and may even reveal the absence of a serious safety problem. This particular example study will look at the potential radiological hazards associated with induced activity in two types of reactor structural material; 316 stainless steel and a molybdenum alloy, TZM. Not only may the release frequency limitations be of value in the development of basic Tokamak system designs, but they may also influence the choice of structural materials.

III. POSSIBLE RELEASES OF INDUCED ACTIVITY FROM TOKAMAK FUSION REACTORS

The examination of possible releases of induced activity must begin with information concerning the amount and distribution of radioactive isotopes in the reactor structural materials. Consideration of basic requirements for releases of activity from the containment and the mechanisms that may be capable of achieving these releases should lead to analyses of particular accident sequences which pose the greatest potential impact on the safety of the public. The most that can be expected from these analyses at the present stage of design development are possible ranges of release conditions which can be used in the described calculational methodology to determine system reliability requirements.

3.1 Induced Activity Inventories

During the operation of a fusion reactor, there will inevitably develop an inventory of induced activity, or activation products, in the first wall and blanket structures. The amount and assortment of radioisotopes will depend on the nature of the particle fluxes, the type of structural materials and, to some extent, the duration of exposure to these particles. The designers of fusion reactor systems may attempt to minimize the induced radioactivity by "isotopic tailoring,"[9] however, the hazards associated with the disruption of the lower radioactivity materials still need to be evaluated.

The first concern in assessing the potential radiological hazard of the induced activity is to determine the inventory and distribution of activity in the reactor structures. Two conceptual systems UWMAK-I[10] and UWMAK-III,[11] will be used here to exemplify the Tokamak-type fusion power reactor activation problem. Studies have been made to determine the activation products which will be induced during the operation of these reactors. The two systems employ different structural materials (316 stainless steel in UWMAK-I and TZM in UWMAK-III), and neither material was assumed to be tailored to minimize radioactivity.

A calculational scheme has been developed at the University of Wisconsin to determine the induced activity inventories.[12] The model computes the activity per kilowatt for each radioisotope, which can then be divided by their maximum permissible concentration in air, (MPC), as given in 10CFR20,[13] to obtain a quantity known as the "biological hazard potential" (BHP). The MPC values are determined by the best available data on the biological effects of each radioisotope. When this data is lacking, a conservative, or low, MPC value is used, resulting in a high BHP value for the particular isotope.

Using the activity distributions obtained from reference (14), a list of the significant isotopes was compiled for each structural material. In compiling the list, any isotope with a half life less than 30 minutes was excluded. Also, any isotope contributing less than approximately, 0.01% of the total BHP was excluded. The half life criterion was used in

the Reactor Safety Study to simplify the consequence calculation by greatly reducing the number of radioisotopes that had to be considered. This was based onthe assumption that these isotopes would contribute little hazard in an accident due to the delay between shutdown and the time when the radioactivity reached the nearby population. The change in activity between shutdown and 30 minutes is relatively small for the inventories studied here. The significant isotopes are listed in Tables 3.1 and 3.2 along with their activities and BHP values in the reactor first wall and the total activated structural material.

3.2 Requirements for Activity Releases

The Reactor Safety Study found that the single significant form of activity release, for the public to be seriously affected by a reactor accident, was that of a plume containing volatile substances and airborne particulates. The plume is formed by the release of materials through a ruptured containment and the transport and dispersal of these effluents by the atmosphere. It seems reasonable to presume that the public radiological hazards associated with the induced activity in a Tokamak reactor would involve the emission of airborne activation products in the form of a plume. Thus, the basic requirements for a hazardous accidental release of the induced activity are assumed to be:

1) disruption of radioactive structures
2) breach of containment structures
3) transport of volatile or particulate forms of

TABLE 3.1

UWMAK-I Induced Activity Inventory in 316 Stainless Steel First Wall and Blanket Structural Material

(after 2 yrs. operation at 1.25 MW/m^2 wall loading)

Isotopes	First Wall		Total Activated Material	
	Activity (Ci X 10^{-4})	BHP (km^3 air/KWth)	Activity (Ci X 10^{-4})	BHP (km^3 air/KWth)
Mn54	7,730	15.46	26,680	53.36
Co58	11,690	11.69	44,930	44.93
Co57	3,302	6.604	9,689	19.38
Co60	817.0	5.447	2,371	15.81
Mn56	24,030	2.403	169,400	169.4
Fe55	34,790	2.319	98,600	6.573
Ni57	687.5	1.375	1,764	3.527
Zr89	26.86	0.5372	85.04	1.701
Cr51	15,480	0.3870	49,820	1.245
Mo99	1,051	0.3003	14,820	4.236
Nb96	14.57	0.2914	44.56	0.8912
Ni63	5.735	0.1147	42.25	0.8451
Fe59	22.67	0.02267	142.7	0.1427
Nb95	23.68	0.01579	70.99	0.04734
Zr95	7.280	0.01456	22.82	0.04563
Zr88	0.5375	0.01075	1.511	0.03023
Cr49	84.05	0.005603	209.7	0.01398
Tc99m	1,051	0.004204	14,820	0.05928
Total	100,814	47.00	433,513	322.2

Note: Table contains no isotopes with half lives less than 30 minutes and no isotopes with a first wall BHP value less than approximately 0.01% of total first wall BHP

Data from reference (14).

TABLE 3.2

UWMAK-III Induced Activity Inventory in TZM
First Wall and Blanket Structural Material
(after 2 yrs. operation at 2.50 MW/m^2 wall loading)

Isotopes	First Wall		Total Activated Material	
	Activity ($Ci \times 10^{-4}$)	BHP (km^3 air/KWth)	Activity ($Ci \times 10^{-4}$)	BHP (km^3 air/KWth)
Zr89	833.4	16.67	833.4	16.67
Mo99	29,800	8.510	29,800	8.510
Nb96	417.8	8.360	417.8	8.360
Nb91m	224.0	4.480	224.0	4.480
Nb95	1,033	0.6890	1,033	0.6890
Zr95	219.3	0.4390	219.3	0.4390
Tc99m	29,910	0.1200	29,910	0.1200
Y88	5.792	0.1160	5.792	0.1160
Sc46	39.80	0.0995	39.80	0.0995
Ca45	25.15	0.0502	25.15	0.0502
Zr97	43.92	0.0293	43.92	0.0293
Y90m	0.4670	0.00940	0.4670	0.00940
Nb92m	919.6	0.00497	919.6	0.00497
Sc47	44.72	0.00438	44.72	0.00438
Nb95m	655.8	0.00437	655.8	0.00437
Nb93m	10.18	0.00407	10.18	0.00407
Nb97	308.1	0.00308	308.2	0.00308
Sr89	0.4511	0.00301	0.4511	0.00301
Y91	0.7517	0.00150	0.7517	0.00150
Y90	2.240	0.00149	2.240	0.00149
Total	64,494	39.60	64,494	39.60

Note: (1) Table contains no isotopes with half lives less than 30 minutes and no isotopes with a first wall BHP value less than approximately 0.01% of total first wall BHP.

(2) Essentially all of the activated TZM is associated with the first wall

Data from reference (14).

radioactive materials out of the ruptured containments. Each event or sequence of events which results in the satisfaction of these requirements consists of processes by which stored energy is released in an uncontrolled and destructive manner. Thus, of crucial interest to any accident analysis are the mechanisms capable of utilizing stored energy to bring about a release of radioactivity.

3.3 Mechanisms Capable of Generating Releases

The Tokamak reactor system consists basically of a toroidal vacuum chamber surrounded by large magnet coils, piping, various auxiliary equipment, and structural members. The assembly is encased in shielding and is enclosed by a containment structure. A schematic of a reactor system is shown in Figure 3.1. A release of induced activity can only occur if part of the first wall or blanket material is disrupted and allowed to escape from the vacuum chamber and through any containment structures. The possible energy sources for the disruption of these structures appear to be the plasma, the magnet system along with its liquid helium cooling system, the after heat or decay heat in the first wall and blanket after shutdown, and the liquid metal (lithium) coolant. A comparison of the approximate magnitudes of some of these energy sources and the energy requirement for vaporization of a molybdenum first wall is shown in Figure 3.2.

The plasma, with a total energy on the order of 10^9J.,[11] could melt or vaporize part of the first wall if a quench or dump was sufficiently localized. A study of melting rates

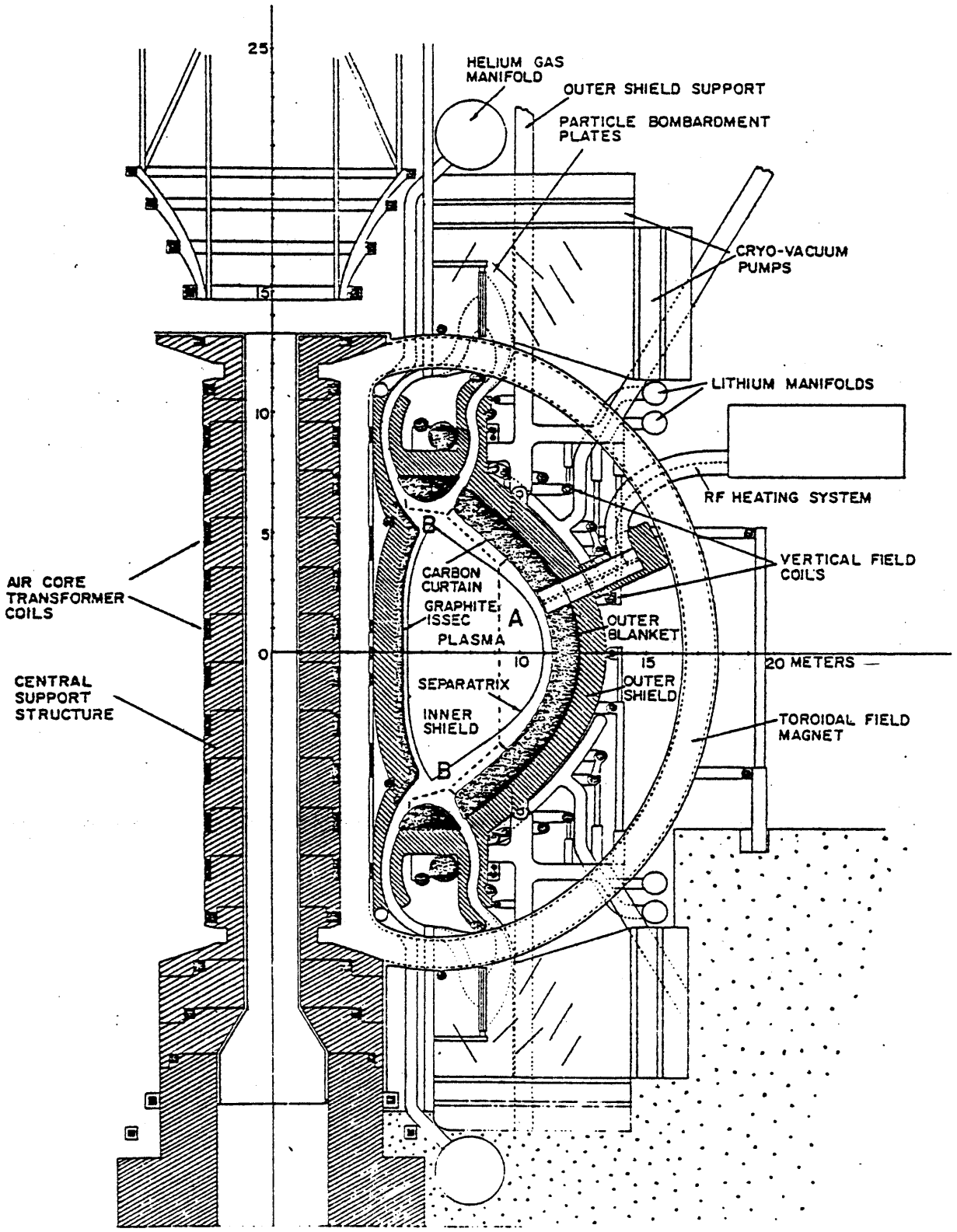


FIGURE 3.1 UWMAK-III REACTOR CROSS-SECTION (FROM REFERENCE 11)

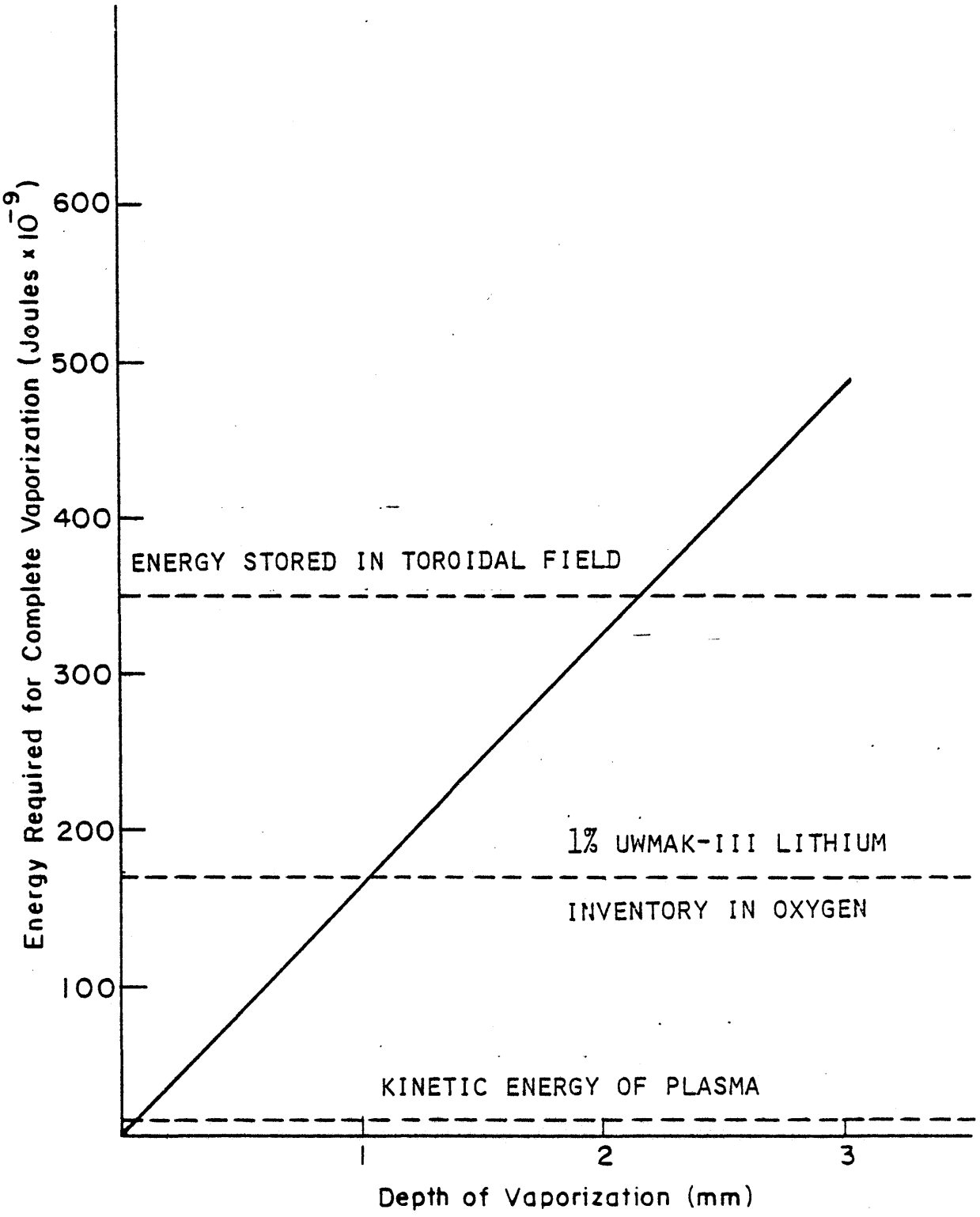


FIGURE 3.2 COMPARISON OF ENERGY SOURCES TO ENERGY REQUIRED TO VAPORIZE AN EFFECTIVE SMOOTH WALL THICKNESS OF UWMAK-III FIRST WALL

for sudden energy dumps[15] showed that the stainless steel would melt slower than molybdenum if no significant material ablation takes place during the energy deposition. If material ablation is important, however, the steel may have a higher melting rate. This energy source itself would not be likely to result in a breach of the reactor vessel and containment.

The much larger stored energy associated with the toroidal magnetic field, which is on the order of 10^{11} J,[11] could result in melting or vaporization of significant fractions of activated structures if a localized energy dump could occur. Recent investigations[16] of magnet failures have shown that the major effect would likely be the development of large forces on some coils, which will most likely be directly supported by a structural frame. Also, there does not appear to be a strong inductive coupling between the coils and the first wall or blanket to produce significant Joule heating. It appears that the magnets may present their greatest threat to the integrity of other structures such as lithium piping, auxiliary equipment and possibly the containment structure.

The helium coolant of the superconducting magnet system could potentially cause structural damage due to thermal interactions and overpressurization of the containment following its vaporization.[3] By itself, however, it does not appear to be capable of disrupting activated material.

The decay heat in the activated structures following shutdown is less of a problem than that associated with decay heat in fission reactors. The typical values of decay power at

shutdown are about 2% or less of the thermal operating power.[12] Fission reactors typically have decay powers greater than 7% of operating power, and the decay power densities can be over 10 times greater than those of Tokamak reactors.[17] Thus, the decay heat in Tokamak reactors may be capable of causing some disruption, particularly in cases of large losses of lithium coolant, however, it does not appear to be the major mechanism leading to disruption of activity as the decay heat is in fission reactors.

The lithium coolant inventory is capable of generating large amounts of heat by interacting with air and concrete, possibly resulting in great structural damage and mobilization of induced activity. The heat generation is on the order of 10^{11} - 10^{12}J. per percent of lithium inventory spilled.[18] Peak flame temperatures of approximately 2500 °K are theoretically possible, which could melt stainless steel, but not molybdenum.[3,18] The reactions with both air and concrete may produce chemically reactive species of oxygen, hydrogen, and nitrogen which are capable of attacking and possibly disrupting activated structures.[3] To summarize, in a recent study of Tokamak power reactor safety,[3] it was concluded that "lithium-air and lithium-concrete reactions have the potential for generating large amounts of heat, significant pressures, and active chemical species capable of volatilizing structural materials such as a radioactive first wall."

Therefore, it appears that very serious possible accident sequences involving large-scale disruption and release of

induced activity may be initiated by a lithium spill or other events which lead to a lithium spill. If the vacuum chamber is ruptured and the first wall is exposed to a lithium fire, first wall ablation could lead to the formation of a radioactive plume. Convective currents within the containment could transport volatilized radioactivity to a rupture in the containment building, which might also serve to allow an influx of air to support the lithium fire. It is conceivable, under these circumstances, that a measurable portion of the induced activity might be released if the volatilization of the first wall is extensive. It is therefore important to examine the volatilization process more closely to determine the possible extent of first wall disruption.

3.4 Possible Formation of Volatile Oxides Following a Lithium Spill and Fire

A possible mechanism for the induced activity to be extensively disrupted and for the activity to be released from the reactor site, is the formation of volatile oxide species. The required high-temperature oxidation process could be initiated by the generation of high-temperature gases during a lithium fire in air, and the exposure of the reactor first wall to these gases following a rupture of the vacuum vessel. The specific accident sequences capable of both rupturing the reactor vacuum chamber and causing a lithium spill may involve magnet system failures which seriously damage the reactor structure. It may even be postulated that the accident leads to generation of missiles causing a breach of the containment, and,

thus, to its failure. Event-tree and fault-tree accident analyses are required to determine the potential for such accident sequences.

The important requirements for first wall disruption by this oxidation process are:

1) generation of a high-temperature oxidizing atmosphere
2) exposure of first wall to the reactive gases
3) poor oxidation resistance of first wall alloy material
4) rapid formation of volatile oxide species

In this study, it is assumed that the activity in the blanket structures and the shielding are not readily mobilized by this postulated accident, and that the activated corrosion products will not pose as serious a problem being spread throughout the entire lithium coolant and subject to possible coolant clean-up systems. The two alloys which will be investigated are 316 stainless steel and TZM, whose nominal compositions are given in Table 3.3.

3.4.1 Generation of High-Temperature Oxidizing Atmosphere

A recent investigation of lithium fires in UWMAK-III at MIT[18] indicates the possibility of creating high-temperature atmospheres in the containment. This study employed a model to determine temperature and pressure histories for lithium spills into air and various other atmospheres. The MIT study also used a chemical equilibrium model to determine theoretical maximum flame temperatures for lithium reactions in air and nitrogen. Some of the pertinent results are discussed in this section, since the information will be used to assess the

TABLE 3.3

Nominal Compositions of First Wall Alloys

Alloy (reactor design)	Constituent	Mass Percentage
316 Stainless Steel (UWMAK-I)	Fe	62
	Cr	18
	Ni	14
	Mo	3
	Mn	2
	Si	1
TZM (UWMAK-III)	Mo	99.4
	Ti	0.5
	Zr	0.1

Data used from references (10) and (11)

TABLE 3.4

Peak Flame Temperatures for Lithium Fires with Melting and Boiling Temperatures for First Wall Materials

Atmosphere	Peak Flame Temperature (°C)	Material	Melting Temp. (°C)	Boiling Temp. (°C)
air	2230	316SS	1450	2730
		Mo	2600	4600
nitrogen	1040	Ti	1660	3290
		Zr	1850	4380

Data used from references (18) and (19)

potential for formation of volatile oxide species from both 316 SS and TZM first walls.

The significance of the peak flame temperature is that it represents the theoretical maximum temperature that can be generated locally in a lithium fire. These temperatures can be used to assess the possibility of direct vaporization or melting of the first wall materials by a lithium flame. Comparisons of the calculated peak flame temperatures with melting and boiling points of the alloys, presented together in Table 3.4, show that the melting of 316 stainless steel appears to be the only direct result of the exposure of the two materials to the flame of a lithium-atmosphere reaction (assuming the thermal properties of TZM are very close to those of molybdenum). Thus, if a fire took place within a stainless steel vacuum vessel, some melting of the first wall may be possible. Melting of the first wall would lead to further release of lithium coolant into the reactor vessel, which may result in the depletion of oxygen in the chamber atmosphere and could limit the reaction rate to one controlled by the diffusion or convection of oxygen into the vessel through any possible ruptures.

The melted stainless steel could mix with the liquid lithium at temperatures approaching 1700 °C in a pool fire.[18] These temperatures are sufficient to cause volatilization of CrO_3 and MoO_3 (Section 3.4.3), however further investigation of the possible interactions between liquid lithium and molten stainless steel is required.

The temperature and pressure histories for lithium fire conditions analyzed in reference (18) reveal the potential for creating maximum containment gas temperatures of about 1050 °C in one hour or less after the onset of the spill and fire.[18] A sample of the gas temperature results is presented in Figure 3.3, which represents a "best estimate" case for a containment with no specific fire mitigation features. The lithium spill consisted of the total inventory of one of the eighteen blanket sections of UWMAK-III, or 5.6% of the total plant inventory. The atmosphere inside the containment was ambient air. It was assumed, therefore, that there was no inert atmosphere to mitigate the reaction. The lithium-concrete reaction was not considered, assuming that the steel liners on the concrete surfaces successfully prevented physical contact.

The important conclusion from these results is the creation of high-temperature reactive gases. The chemical equilibrium analyses showed that some free oxygen atoms would be present in this atmosphere along with diatomic oxygen molecules, and at a temperature of 1050 °C and low pressures (1 torr), the oxidation probability for oxygen atoms at a metal surface may be almost three orders of magnitude greater than the probability for oxygen molecules.[20] The effect of atomic oxygen on the oxidation rate, however, would be less with significant pressures of other species and the equilibrium gas mole fraction of atomic oxygen appears to be only 0.002 for a lithium-air reaction.[18] The impact of these results on the formation of volatile oxides will be discussed later in this chapter.

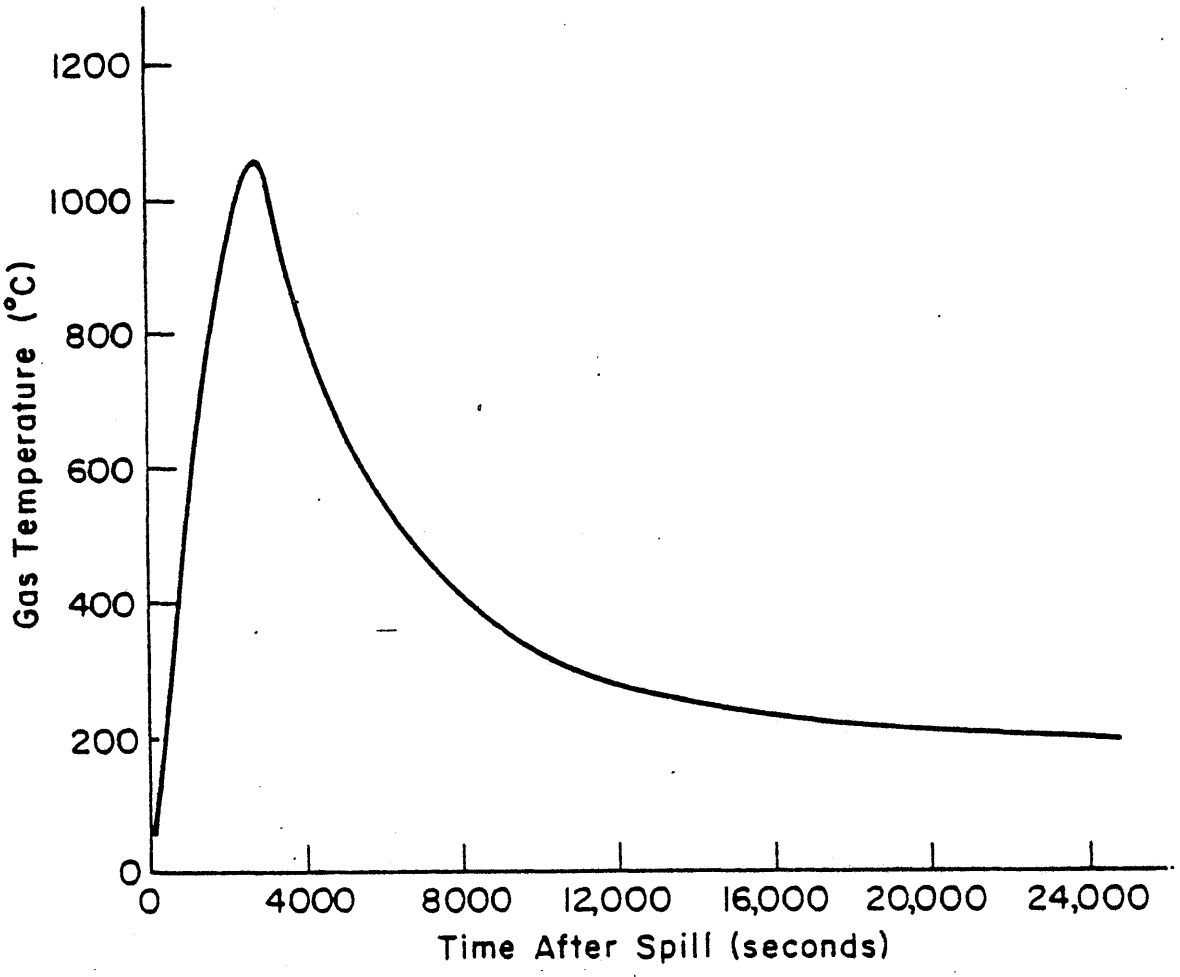


FIGURE 3.3 UWMAK-III CONTAINMENT GAS TEMPERATURE HISTORY FOR A LITHIUM FIRE WITH NO MITIGATING FEATURES (FROM REFERENCE 18)

3.4.2 Exposure of First Wall to Oxidizing Atmosphere

To bring the first wall in contact with the high-temperature gases produced in a lithium fire in the containment, the vacuum vessel must be ruptured to allow the gases to penetrate into the toroidal chamber. If a hole was created in the vessel with its interior at an operating pressure of 10^{-5} torr,[11] the gas pressure at the exterior would cause a sonic flow to be generated at the break cross section until the pressures were substantially equalized. In Appendix A.1 of this report, the possible time for this process to take place is estimated. It is found that for a hole greater than 100 cm^2 in area, the chamber may take less than 10 minutes to reach the pressure level of the gas in the containment, and for a hole greater then 1000 cm^2, it may take less than 1 minute. The important point is that if activity volatilized by the gases entering the reactor chamber is to be transported out of the reactor, rather large holes will be required, and thus the vacuum chamber will be filled quite rapidly for the conditions of interest in this study.

In the case of a large hole being created at the start of the lithium fire, it is seen from Figure 3.3 that it may be possible that the gases rushing into the chamber will not be particularly hot, and thus would not present a serious problem of disruption of material if convection currents within the containment cannot bring hot gases into the reactor vessel. If, however, the hole is very large or the vessel is ruptured in a number of places which serve to circulate the hot

containment gases through the vessel, both volatilization and transport of activity may be possible.

3.4.3 Oxidation Resistance of First Wall Alloys

The best conditions for resistance of an alloy to oxidation are the formation of a compact (non-porous) protective oxide layer on the exposed surface, with further oxidation of the base metal being controlled by solid state diffusion of oxygen or metal ions through the scale. Alloys for high temperature use in oxidizing environments should provide a strong oxide scale with a high melting point, a relatively high negative free energy of formation to hold on to the oxide ions, and low diffusion coefficients for metal ions and oxygen.[22] To assess the possibilities of first wall disruption the oxidation resistance of each of the alloys should be examined.

i) 316 SS

The oxidation resistance of stainless steels is generally good at high temperatures.[22] The presence of molybdenum as an alloying element in 316 SS, however, may lead to a phenomenon called "catastrophic oxidation," wherein alloying elements form volatile oxides or oxides with low melting temperatures.[20,21,23] The melting point of MoO_3 is 795 °C, and its presence in an oxide layer can lead to the disintegration of the oxide at high temperatures.[20] The liquid oxide may dissolve other oxides, and thereby accelerate the metal oxidation.[20] In addition, protective oxide scales consisting of Cr_2O_3 can be oxidized to CrO_3, which is volatile at temperatures above 1000 °C.[23,24,25]

Molybdenum has been found to cause catastrophic oxidation when added in certain concentration ranges to both iron-nickel and iron-chromium alloys. In Figure 3.4, the shaded regions indicate the metal weight loss (in mg/cm^2) after 2 hours exposure to air at 1000° C. It is seen on the iron-chromium-molybdenum graph that an alloy of Fe-18Cr-3Mo should experience a metal weight loss of about 112 mg/cm^2, which means that after 2 hours, 112 mg of the alloy have been oxidized per square centimeter.

If the nickel content (14%) of 316 SS is assumed to have no effect on this oxidation rate, then the stainless steel first wall may be subject to rapid oxidation and volatilization. The volatilization of MoO_3 would be expected at temperatures above 700° C,[21] and CrO_3 volatilization above 1000 °C. Also, temperatures above the gas temperature can be reached in the oxide and metal due to the rapid release of heat in the oxide formation during catastrophic oxidation.[20,23]

If the weight loss taken from Figure 3.4 could be assumed to represent a uniform oxidation rate over the 2 hour test period, the time for total oxidation of the first wall exposed to air at 1000° C may be estimated. In Appendix A.2, this calculation shows that if catastrophic oxidation is to consume the entire first wall at this constant rate, the wall would be totally oxidized in 36 hours. This shows that the process is rather slow for significant oxidation during an accident, since the temperature of the gases in the containment will not remain at 1000° C for more than several hundred seconds

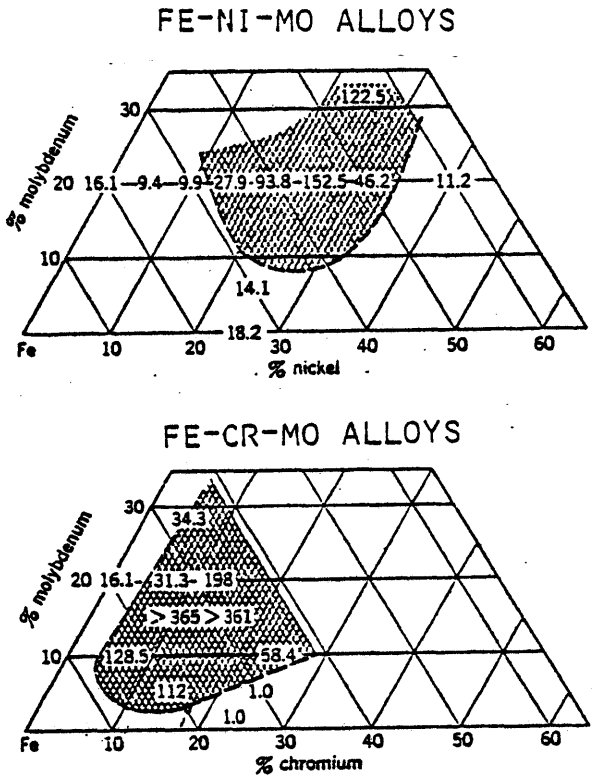


FIGURE 3.4 CATASTROPHIC OXIDATION IN IRON-CHROMIUM-MOLYBDENUM AND IRON-NICKEL-MOLYBDENUM ALLOYS:

SHADED AREAS FOR CATASTROPHIC OR RAPID OXIDATION; NUMBERS GIVE APPROXIMATE METAL WEIGHT LOSS IN MG/CM^2 FOR 2 HOURS EXPOSURE TO 1000 °C AIR
(FROM REFERENCE 20)

(Figure 3.3). In addition, the volatilization would involve only some of the molybdenum and chromium in the alloy.

ii) TZM

The high-temperature oxidation of TZM is assumed to be very similar to that of molybdenum. The small percentages of titanium (0.5%) and zirconium(0.1%) are assumed to have little effect on the oxidation properties of molybdenum, though they appear to be capable of adversely affecting the oxidation resistance of metals.[22] The formation of liquid and volatile oxides of molybdenum can be rapid at temperatures above 700 °C, thus significant amounts of TZM first wall may be disrupted by this mechanism.

At high temperatures, molybdenum can be oxidized to a volatile compound, MoO_3, which evaporates from the surface at a rate determined by the temperature of the metal and gas, the gas pressure, and the gas flow rate along the surface.[20] Figure 3.5 was constructed from data presented in reference (20), and it is seen that increases in oxygen pressure, gas flow rate, and temperature generally increase the surface recession rate of molybdenum. The flow of gases along the surface serve to remove evaporated molecules which can form a boundary layer limiting the arrival rate of oxygen at the surface.[20] Thus, the reaction rate limiting effects of the boundary layer are reduced with increasing flow rates. The data presented in Figure 3.5 for flowing air represent the maximum oxidation rates observed, thus, the surface recession rate was limited only by the reaction rate at the surface.[20] The partial

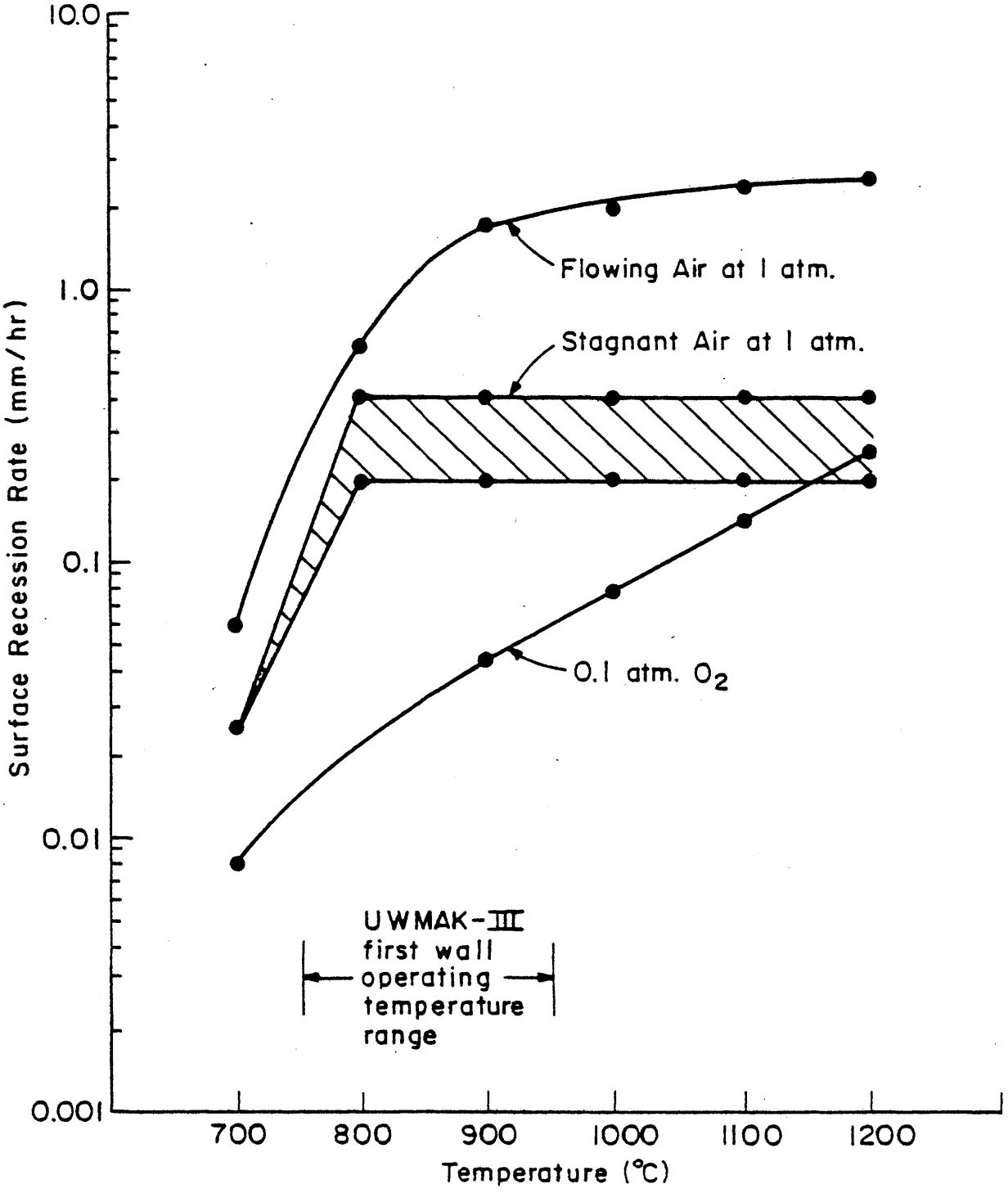


FIGURE 3.5 VOLATILE OXIDATION OF MOLYBDENUM

pressure of oxygen affects the surface reaction rate, which is greater in air (pO_2=0.21 atm) than in oxygen at 0.1 atm pressure. The results presented in reference (18) reveal that following the spill of lithium from a single coolant loop in UWMAK-III, the oxygen gas mole fraction is reduced during the lithium fire, but remains above 0.15. This study will assume that the high-temperature gases in the containment remain essentially air. The temperature also has a strong effect, with significant reaction rates in air above 700 °C. The rate of volatilization in air increases with temperature to a rate determined by the surface reaction (flowing air) above 1200 °C, or to a rate determined by the transport of oxygen through the volatile oxide boundary layer above 800 °C. There does not appear to be a reaction rate "saturation" effect for the 0.1 atm O_2 data due, probably, to the lower back-reflection rate, or higher molecular escape fraction, for the decreased total gas pressure.[20]

The significance of these results for the study of first wall disruption is that for air temperatures of 1000 °C, which may be generated during a lithium-air fire, it is possible to have a disruption rate of from 0.2 to 2.0 mm/hr, depending on the flow of gases at the first wall. Since the outer or minimum first wall thickness for UWMAK-III is 1.5 mm,[11] this allows for the total disruption or volatilization of the first wall in air at 1000 °C in 45 to 450 minutes. Figures 3.6 shows the time for total TZM first wall volatilization in 1 atm air for various temperatures, given either stagnant or flowing air.

3.4.4 Maximum Estimated First Wall Releases

The important parameter for accidental releases of induced

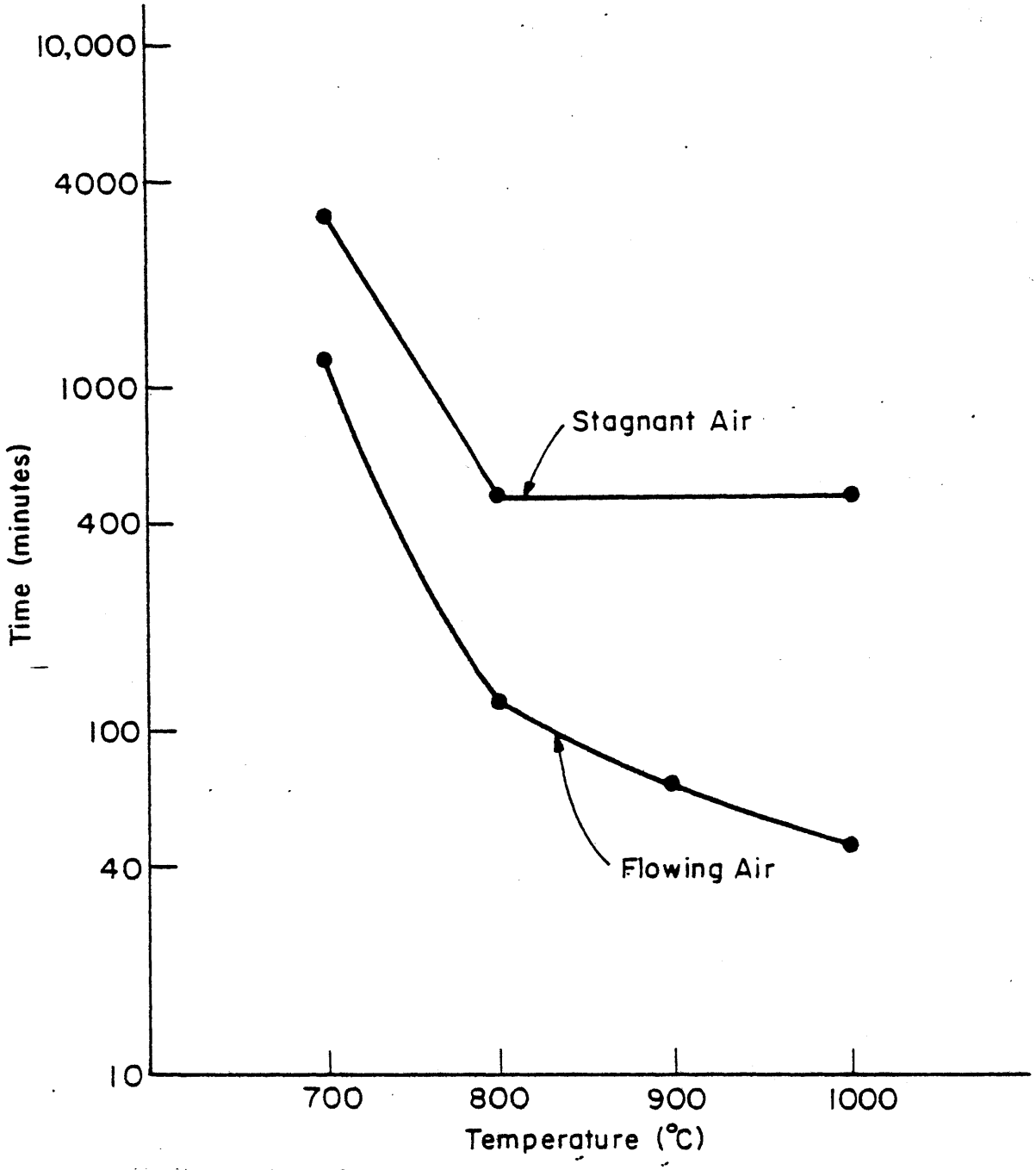


FIGURE 3.5 TIME TO VOLATILIZE UWMAK-III FIRST WALL (THICKNESS = 1.5 MM)

activity is the amount of the first wall which is mobilized and transported out of the reactor containment. If it is assumed that the material that is volatilized from the first wall is eventually released from the plant, then the results of the examination of high-temperature oxidation are directly applicable. Maximum estimates for these release magnitudes are desired to provide a range of possible release fractions for the consequence calculations, thus, some arbitrary assumptions will be made.

The release magnitude of 316 stainless steel activity will be based on the fraction of the first wall oxidized by the catastrophic oxidation process described in the previous section. It should be recalled that this is actually not the amount of material that is volatilized, but it should provide an upper limit for the first wall mobilization. Total catastrophic oxidation of the UWMAK-I first wall was estimated to take 36 hours at 1000° C. The containment gas temperature is not expected to be at or above this temperature for more than 1000 seconds as seen in Figure 3.3. Therefore, if a uniform disruption rate is assumed, no more than:

$$\frac{1000}{36(3600)} = 0.0077$$

or 0.8% of the first wall may be disrupted during this time period. If convection currents, vibrations, stresses, or other forms of disrupting action are assumed to be capable of accelerating the oxidation by dislodging some of the solid oxides to form airborne particulates, it may be conceivable to have

additional material being released. Many complex processes are involved in the eventual release of these materials from the containment, but it seems reasonable to limit the possible release magnitudes to below 10% of the total UWMAK-I first wall.

The release magnitude of TZM will be based on the surface recession rates described in the previous section. Since their rates vary significantly over the temperature range of interest (700-1000° C), an estimate which accounts for the temperature change during the lithium fire will be required. A simplified temperature history will be used for the calculation, and it is basically a linearized form of the function in Figure 3.3. This modified function is shown in Figure 3.7. The curves for the recession rates in air, shown in Figure 3.5, will be simplified to a line between the two maximum values at 700 °C and 1000 °C on the semilog plot. This modified function is shown in Figure 3.8, along with the unmodified functions.

In Appendix A.3, a calculation of the estimated fraction of the UWMAK-III first wall using these approximations reveals that about 31% of the wall can be volatilized. Again, as with the UWMAK-I first wall, if convection currents, vibration, stresses or other disrupting actions are assumed to be capable of dislodging some of the solid oxides which will form when the gases in the containment cool to temperatures below 700 °C, it is conceivable that additional material may be released in the form of airborne particulates. A reasonable absolute maximum release magnitude may be 50% of the total first wall.

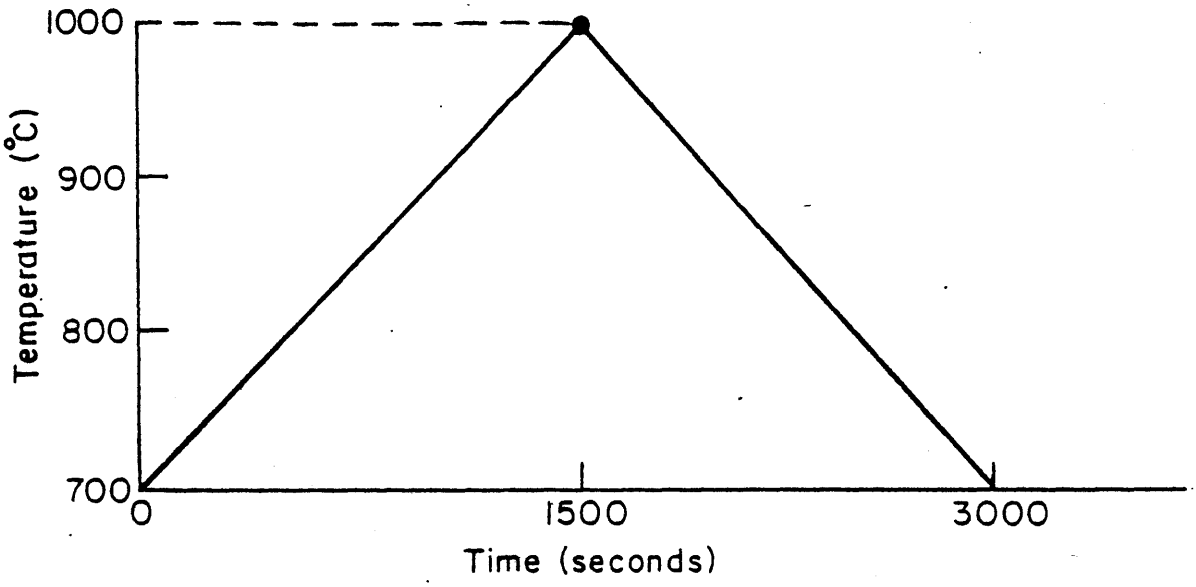


FIGURE 3.7 SIMPLIFIED TEMPERATURE HISTORY FOR LITHIUM FIRE

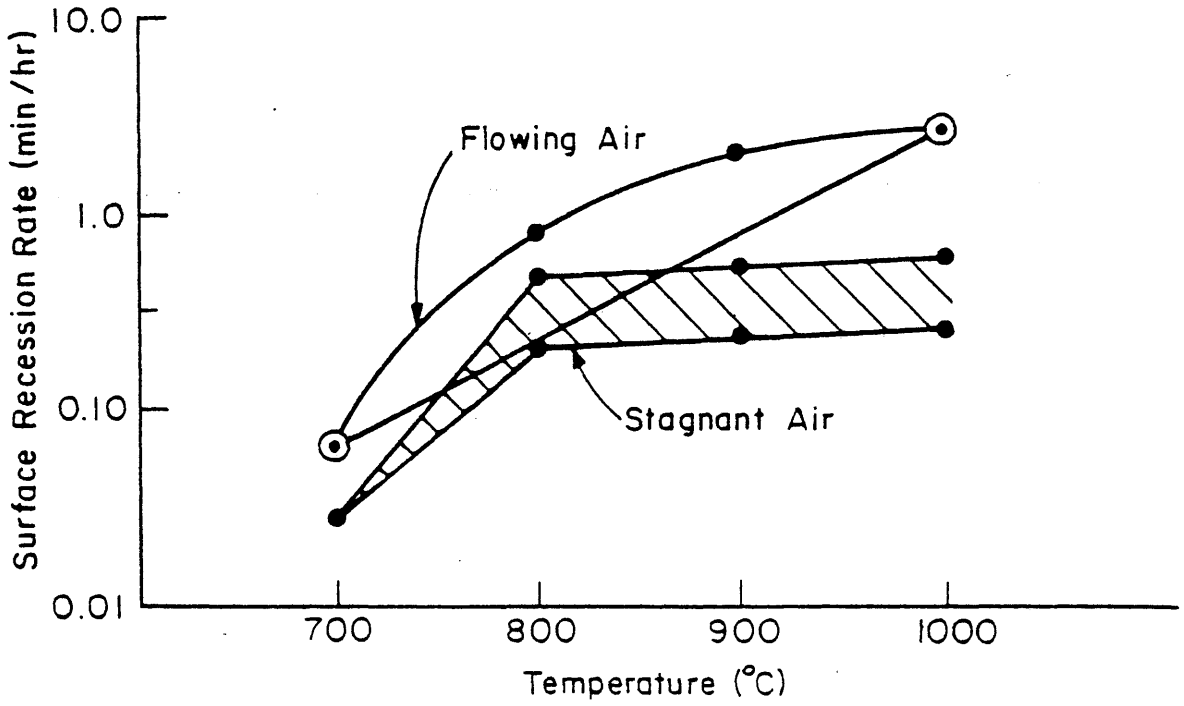


FIGURE 3.8 SIMPLIFIED FUNCTION FOR VOLATILE OXIDATION OF MOLYBDENUM

The maximum possible release magnitudes for the first wall induced activities, shown in Table 3.5, will be used with the consequence model to estimate minimum system reliability requirements. The release limits establish possible ranges for the release magnitudes and, at this point, it appears that the TZM first wall in UWMAK-III may pose the greatest radiological hazard to the public. It is important to remember that these release magnitudes were estimated with the assumption that a lithium fire produces a gas temperature history similar to Figure 3.3. It has been shown in the MIT investigation of lithium fires,[18] that with proper design of reactor systems, the maximum gas temperatures may be limited to substantially below 750° C, which is the temperature where volatilization begins to become a significant problem. Therefore, it may be possible to design a fusion reactor system for which induced activity can not be volatilized and released from the reactor containment. The reaction of lithium with concrete was not specifically examined in this study, however, and further work should assess its potential for disrupting induced activity.

TABLE 3.5

Estimated Maximum Possible Release Magnitudes for Volatile Oxidation of Tokamak Reactor First Wall Structures

Reactor Design	First Wall Material	Maximum Percentage of First Wall
UWMAK-I	315SS	10
UWMAK-III	TZM	50

IV. CONSEQUENCE-PROBABILITY CALCULATIONS

The CRAC computer code developed for the Reactor Safety Study (WASH-1400)[7] is used to generate complementary probability distribution functions for the consequences of releases of radioactivity. A brief description of the CRAC code and the modifications required for this study are discussed in this chapter. The accident conditions assumed for the calculations, and the calculated results are presented for each type of first wall material (316 SS and TZM) and for both types of light water reactors (PWR and BWR). These calculated consequence-probability distribution functions serve as the basis for determining the maximum tolerable accident probabilities, as described in Chapter II.

4.1 Description of the Consequence Model

A detailed description of the consequence model can be found in Appendix VI of the Reactor Safety Study, and the operation of the code is described in its user's manual.[8] A basic description is given in the rest of this section, followed by a discussion of the modifications which were required to enable the code to handle the radioisotopes not already incorporated in its data sets.

A schematic outline of the consequence model is shown in Figure 4.1. The starting point for the calculation is seen to be the accident release conditions. The various parameters involved in these conditions are:

i) the magnitude of the released radioactivity

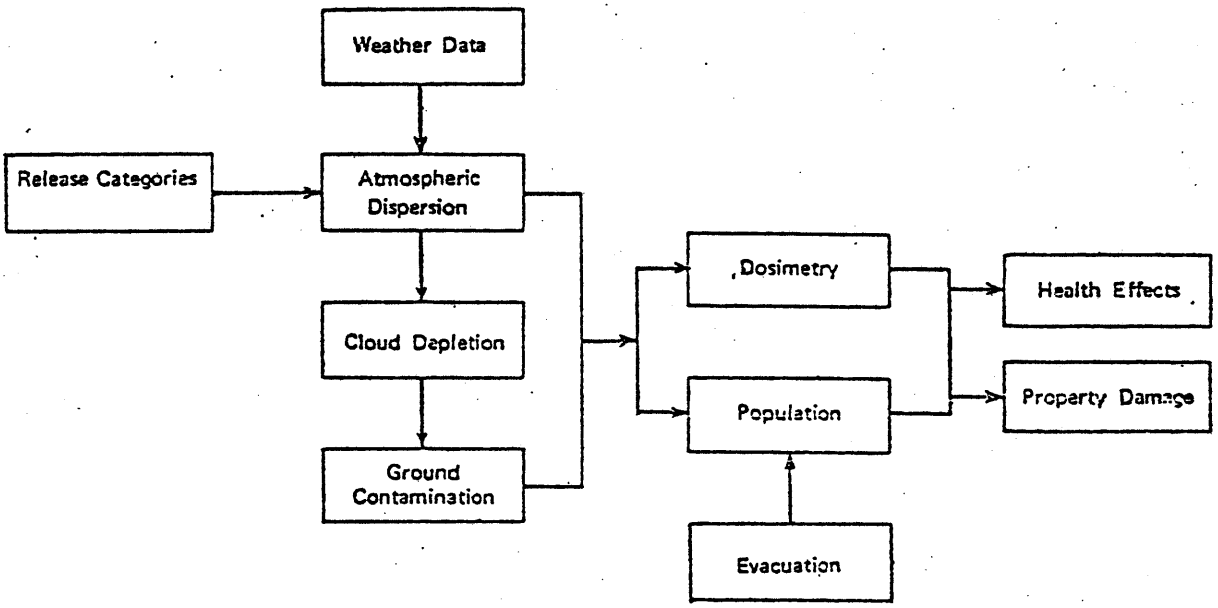


FIGURE 4.1 SCHEMATIC OUTLINE OF CONSEQUENCE MODEL (FROM REFERENCE 7)

ii) the duration of the release event;

iii) the heat content of the plume;

iv) the release height above the ground;

v) the delay time and warning time (for evacuation) associated with the accident sequence;

vi) and a probability for the occurrence of each category of releases.

For commercial light water reactor plants, the spectrum of releases were discretized into nine PWR and five BWR categories.

A meteorological model then computes the dispersion and deposition of radioactive materials as a function of time after the accident and distance from the reactor site. The model incorporates factors accounting for the decay of the radioactivity and includes the effects of thermal stability, wind speed and precipitation in a Gaussian dispersion model. The temporal variations of these weather conditions are obtained by using samples from a year's weather data from various reactor sites. Variations of the mixing layer are also included. The effects of the plume lifting off the ground due to its sensible heat content is included, and the plume is not allowed to penetrate the mixing layer. The code has options allowing for the specification of weather conditions independent of the site data files.

Once the concentration of the radioactivity in the air and on the ground is determined, the model calculates the possible doses from various modes of exposure. These include external irradiation from the cloud and radioisotopes deposited

on the ground, and internal irradiation from inhaled radioisotopes and from ingestion of contaminated crops, water and milk. The distribution of people about the reactor site is used along with an evacuation model to obtain a set of doses for the affected population. These doses are transformed to actual health effects (see Appendix B.2 of this report), such as early (within a year) fatalities, early illnesses (serious radiation sickness), cancer deaths and genetic effects. The code also computes estimates for property damage or economic costs for the accidents. These final results are used as a measure of the accident consequences.

The final results are presented in the form of complementary probability distribution functions for the spectrum of release conditions at various reactor sites using the spectrum of weather conditions. The risk assessment for a large number of individual reactor accidents is expressed as a set of complementary cumulative distribution functions for each of the potential consequences. For this study, as described in Chapter II, cumulative functions will be generated for light water reactors with their spectrum of release categories, and conditional functions will be generated for Tokamak reactor releases with the probabilitites for their occurrence assumed to be unity.

4.2 Modification of Consequence Model

The modifications required to utilize the CRAC code with first wall activity releases were basically the incorporation of dose conversion factors for radioisotopes not already included in the fission isotope data file. These dose conversion

factors are used to determine specific dose values from radioisotope activity concentrations. The factors required for the code are of two basic types; those associated with internal exposure through inhalation of activity and those associated with external exposure from activity in the air and on the ground. Each isotope has a specific set of dose conversion factors due to the nature of its radiation (α,β,γ or x-rays) and the energy distributions associated with each particle or photon. The external dose conversion factors were obtained from Oak Ridge National Laboratory,[26] and were calculated with the EXREM-III computer program. The inhalation dose conversion factors were also obtained from Oak Ridge,[27] and were calculated using the ICRP Task Group model. A detailed description of these dosimetric models is given in Appendix VI of the Reactor Safety Study.

At the time of this study, the dose conversion factors for some of the isotopes listed in Tables 3.1 and 3.2 were not available. However, from Appendix B.1 of this report it is seen that isotopes representing approximately 98% of the total BHP of the requested 316 stainless steel inventory and 79% of the total BHP of the requested TZM inventory, were received and incorporated into the CRAC code. A new data file containing the dose coversion factors for these isotopes was generated, and was accessed by the program during the consequence calculations for releases of induced first wall activity.

The consequences calculated by this model fall into two basic categories; those that result from the initial exposure

of the population to the released activity during the short times associated with the activity dispersal, and those that result from long term chronic exposures. The long term exposures require data for the behavior of the isotopes in the environment and the human body over long time periods. Since this type of data was not readily available for some of the new isotopes, the consequences based on chronic doses were not used in this study.

The consequences used in this study were the health effects of the initial or early exposure of the population to the radioactive cloud and the ground contamination during the evacuation and relocation time periods. The specific results which depend solely on these exposures are early fatalities, early illnesses, and initial latent effects (cancers) affecting various organs. These are the only results which will be used to determin the maximum tolerable accident probabilities for the Tokamak releases. The doses required by the model for these health effects are presented in Appendix B.2 of this report.

4.3 Accident Conditions

The model requires input to describe the conditions associated with the release accident. The important input parameters can be grouped into five categories:

1) population distribution around site
2) weather conditions during accident
3) magnitude of radioactivity release
4) description of plume formation
5) description of evacuation measures

Each of these types of parameters has an important effect on the eventual consequences, and the same population, weather and evacuation conditions will be assumed for LWR and Tokamak releases. The conditions directly involved in the releases of radioactivity from the containment will be the only differences between the two reactor types, thus providing a valid basis for comparing the consequences of releasing the different radioactive materials.

4.3.1 Population Distributions

The assumed population distribution about the reactor site was a uniform population density of 200 people per square mile. The average population density of the United States (including Alaska and Hawaii) is about 56 people per square mile based on the 1970 census, and the most densely populated state, New Jersey, has over 900 people per square mile. Thus, the assumed population density is a reasonable value for many areas in the country.

4.3.2 Weather Conditions

The atmospheric conditions during a release accident strongly influence the dispersal of radioactivity. Appendix VI of the Reactor Safety Study contains a detailed description of the atmospheric dispersion model, and the important parameters used to estimate the dilution and transport capability of the atmosphere are:[7]

1) wind speed
2) occurrence of rain
3) atmospheric stability (related to the temperature

variation with altitude)

4) mixing height (for stable and unstable conditions)

The assumed conditions for the calculations were all chosen to have a relatively high probability of occurring, according to the data presented in reference (7). The values chosen for the model are given in Table 4.1.

4.3.3 Evacuation

The evacuation model assumes that a small radius around the plant, about 5 miles, is evacuated when warning is received (warning time is discussed later along with plume formation parameters). In addition, a 45° angle sector centered about the wind direction is evacuated out to a distance of 25 miles. The evacuation velocity is assumed to be 1.2 mph, which was used in the Reactor Safety Study to represent a reasonable speed. The evacuation area is shown in Figure 4.2.

4.3.4 Release Description

This category of input parameters is the only one which will vary among the reactor types. These parameters describe the amount of radioactivity in the plume and the type of plume that is formed. The fission reactor parameters are described in detail in Appendix VI of the Reactor Safety Study and the isotope inventory and release conditions used in this study for the light water reactors are given in Appendix B.3 of this report.

The most important parameter for the Tokamak reactor calculations is the fraction of the induced activity inventory released from the containment. The examination of the possible formation of volatile oxides during a

TABLE 4.1

Weather Conditions Used in Consequence Calculations

Parameter	Value or Condition
wind speed (m/s)	2.0
rain	none
atmospheric stability	neutral
mixing height (km)	
stable	0.6
unstable	1.2

TABLE 4.2

Tokamak Reactor First Wall Release Fractions Used in Consequence Calculations - Percentage of First Wall

2	8	30	60	90
4	10	40	70	100
6	20	50	80	

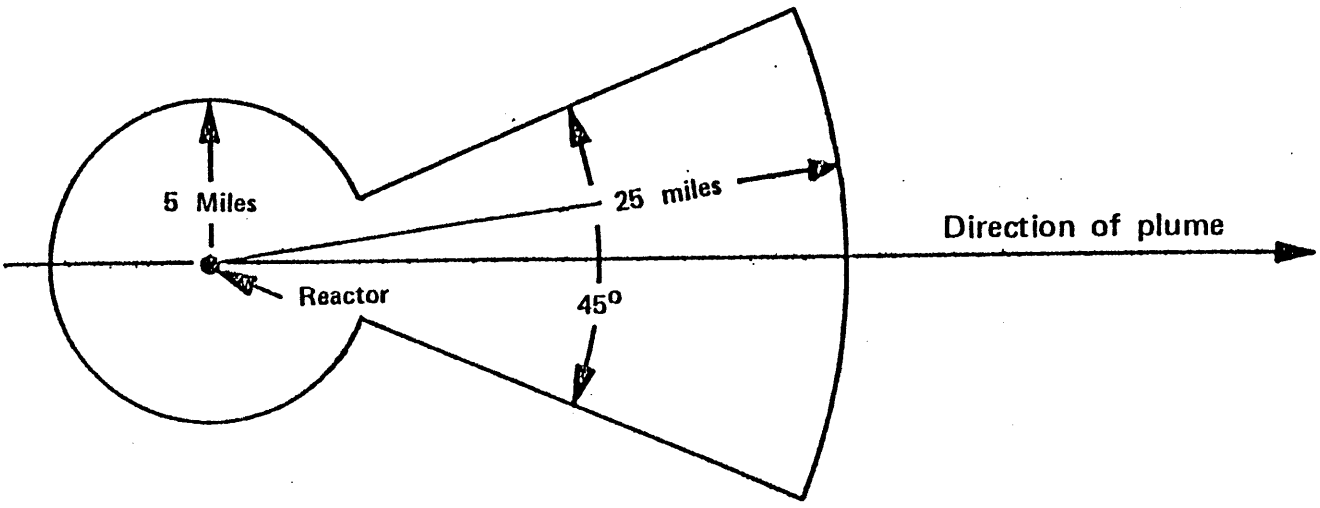


FIGURE 4.2 AREA OF EVACUATION FROM SITE OF RELEASE ACCIDENT

lithium-air reaction in Section 3.4 suggested maximum values for this parameter. These values were used to establish ranges for possible releases, however, calculations were made for release magnitudes up to the entire first wall inventory, since these results would be of interest in an assessment of the hazards associated with induced activity. The values assumed for the release fractions are given in Table 4.2.

The remaining parameters required to describe the plume formation and population distribution following the accident are:

1) time between reactor shutdown and activity release from containment
2) warning time for evacuation before activity release from containment
3) duration of release
4) release height
5) thermal energy release rate

It was determined in the Reactor Safety Study that a cold release at ground level would maximize the acute or early consequences due to the intense exposure at distances close to the reactor where the radioactivity concentrations are still high. On the other hand, hot releases from the top of the containment would result in a rising plume which would reduce the exposure of the nearby population to the high concentrations of activity. The gas temperatures in the containment will vary during the course of the accident, however, to achieve the highest possible initial exposures, a cold release at ground level is assumed for the Tokamak reactor consequence calculations.

The duration of the release determines the length of the plume and thus the initial dilution of the activity in the direction of the wind. Also, for slow releases, temporal variations in wind direction will greatly reduce the concentration of activity in the plume. Again, to achieve the worst possible health effects, a reasonably fast release with a total duration of 1 hour was assumed.

The time between reactor shutdown and release determines the initial decay of activity. The warning time gives the evacuation a start before the release begins. Assuming that the warning is given when the reactor shuts down and the release begins only one half hour after the shutdown, both times will be equal to one half hour. Examination of Figure 3.3 reveals that this time corresponds approximately to a gas temperature of 850° C if the lithium spill is assumed to begin at reactor shutdown. At this temperature, significant volatilization of molybdenum may occur, but this assumption implies that the radioactivity is being released from the containment almost immediately after it is volatilized from the first wall.

The input parameters for the Tokamak reactor plume formation are given in Table 4.3. These will be used with the various release fractions for both first wall radioactive inventories. These conditions are expected to provide consequences which represent an extreme for the initial exposure of the population.

TABLE 4.3

Consequence Model Input Parameters for Tokamak Reactor Release Characteristics

Parameter	Values
time between reactor shutdown and release from containment (min.)	30
warning time for evacuation (min.)	30
duration of release (hr.)	1
release height (m.)	0
thermal energy release rate (cal/s)	0

4.4 Results of Consequence Calculations

The consequence probability distribution functions calculated by the CRAC model for the light water reactor and Tokamak reactor accidental radioactivity releases are presented in this section. The light water reactor results are in the form of complementary cumulative probability distribution functions which represent estimated radiological hazards associated with the operation of these reactors on a site with the given population and weather conditions. If the analyzed conditions for the population and weather exist throughout the year, then the probabilities are on a per year basis. The results calculated for induced activity releases following a Tokamak reactor accident are in the form of conditional complementary probability distribution functions which represent the hazards associated with a given particular accident, with a probability of occurrence equal to unity. These are very different types of functions, with the conditional functions essentially providing single values for consequence magnitudes, while the cumulative functions provide probability distributions. The consequences for the TZM first wall releases will be slightly under-predicted due to the isotopes not included in the model, as described in Appendix B.1.

The health effects used in this study to measure the accident consequences are:

1) number of latent cancer fatalities due to initial exposures
2) number of early fatalities (within a year)

3) number of early illnesses

Appendix VI of the Reactor Safety Study describes the basis for determining the doses required for these effects, and these doses are presented in Appendix B.2 of this report. The final calculated results are presented in Figures 4.3 through 4.5.

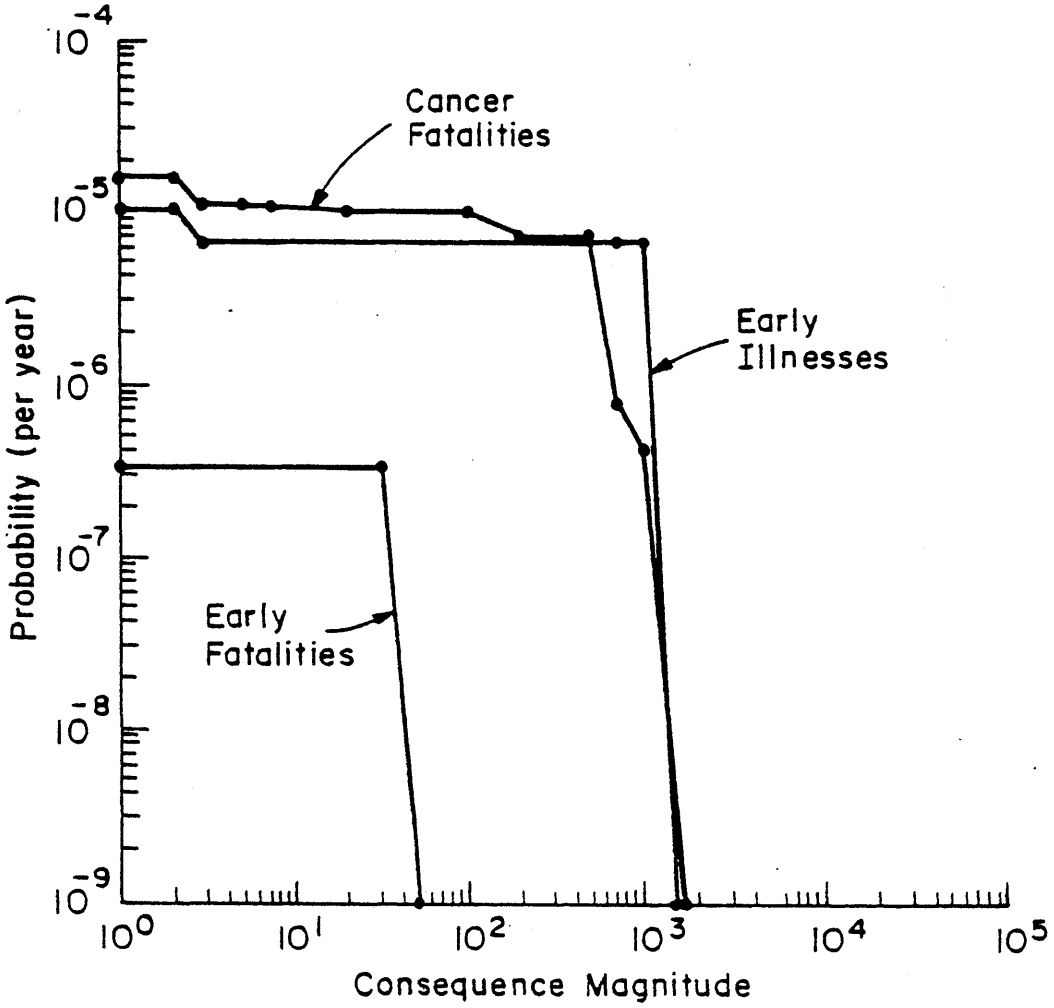


FIGURE 4.3A COMPLEMENTARY CUMULATIVE PROBABILITY DISTRIBUTION FUNCTIONS FOR PWR RELEASE CONSEQUENCES

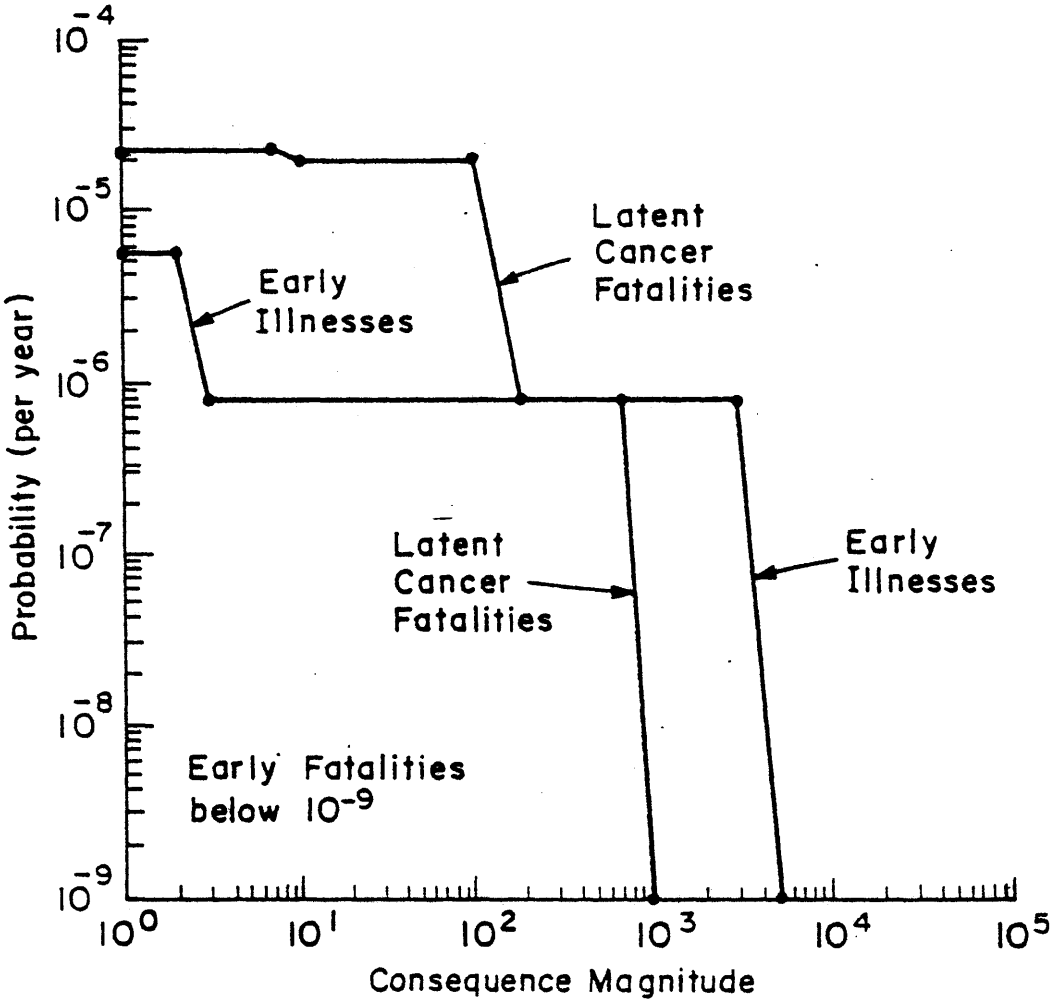
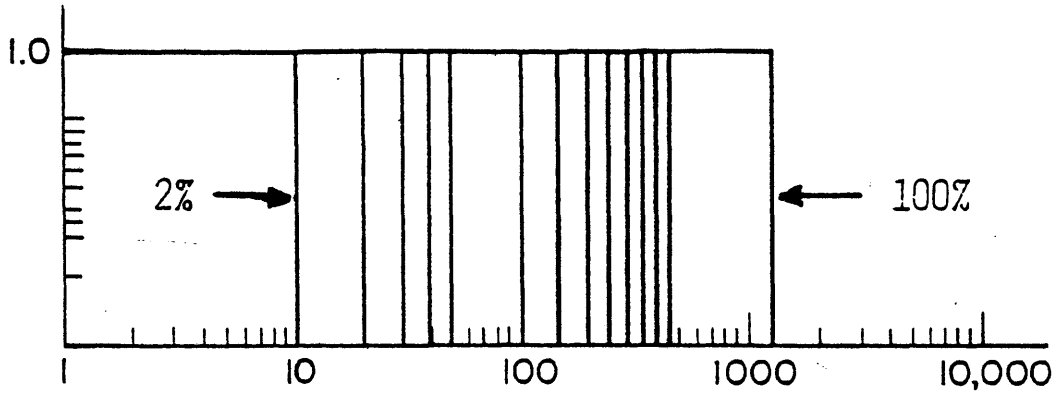
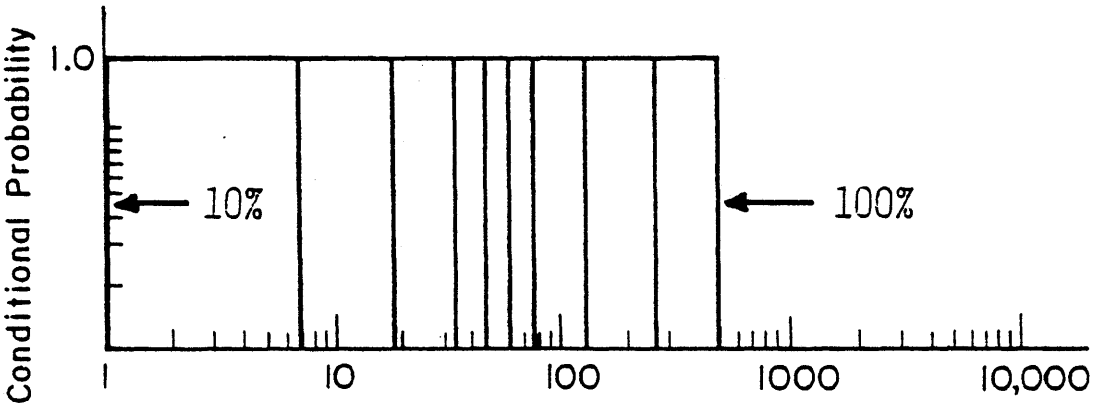


FIGURE 4.3B COMPLEMENTARY CUMULATIVE PROBABILITY DISTRIBUTION FUNCTIONS FOR BWR RELEASE CONSEQUENCES

A. LATENT CANCER FATALITIES



B. EARLY ILLNESSES



C. EARLY FATALITIES

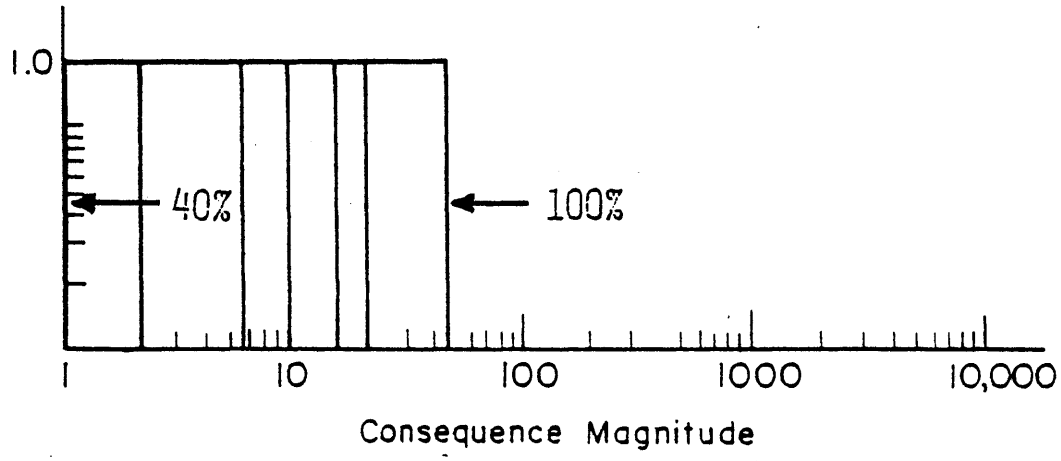
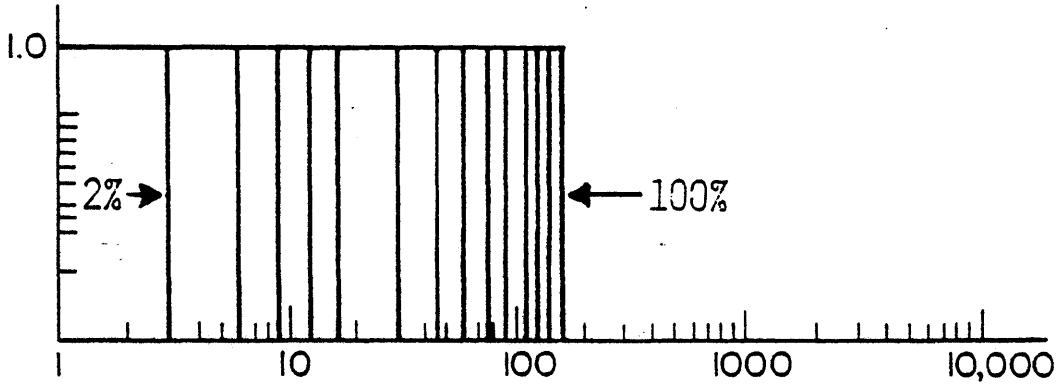


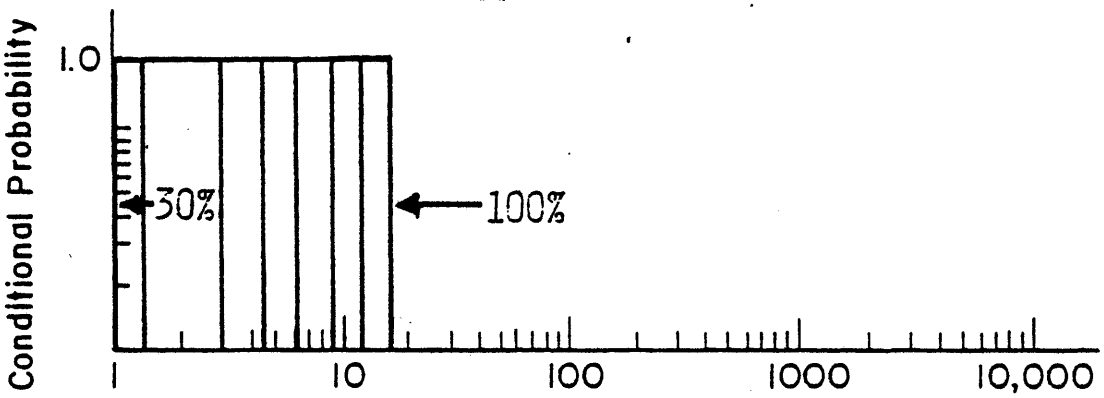
FIGURE 4.4 COMPLEMENTARY CONDITIONAL PROBABILITY DISTRIBUTION FUNCTIONS FOR UWMAK-I FIRST WALL ACTIVITY RELEASES

(THE VARIOUS PERCENTAGES OF THE FIRST WALL ARE GIVEN IN TABLE 4.2)

A. LATENT CANCER FATALITIES



B. EARLY ILLNESSES



C. EARLY FATALITIES

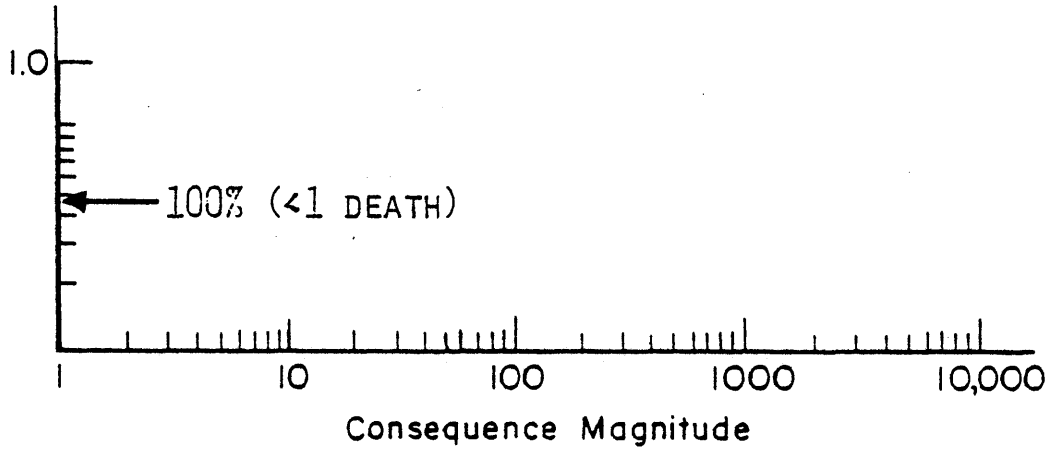


FIGURE 4.5 COMPLEMENTARY CONDITIONAL PROBABILITY DISTRIBUTION FUNCTIONS FOR UWMAK-III FIRST WALL ACTIVITY RELEASES

(THE VARIOUS PERCENTAGES OF THE FIRST WALL ARE GIVEN IN TABLE 4.2)

V. MAXIMUM TOLERABLE RELEASE PROBABILITIES

The basic methodology used to derive a set of maximum tolerable release probabilities for a range of release fractions for the induced activity inventories was described in Chapter II. This technique employs the results presented in Figures 4.3 through 4.5, which were calculated using the CRAC code along with the accident conditions described in Chapter IV. The possible ranges of the release fractions for the 316 stainless steel first wall in UWMAK-I and the TZM first wall in UWMAK-III, which were based on the potential of volatile oxidation to disrupt these structures (Chapter III), can be interpreted as defining the range of credible or conceivable accidents involving the release of induced radioactivity. The sets of maximum tolerable release probabilities, however, will include values corresponding to various release fractions of the first wall. The implications that the release probabilities have for system reliability requirements will be discussed at the end of this Chapter.

5.1 Determination of Release Probability Limitations

The consequence magnitudes represented by the complementary conditional curves in Figure 4.4 and 4.5 are given in Table 5.1. The values obtained for the maximum tolerable accident probabilities corresponding to each one of these consequence magnitudes is given in Tables 5.2 and 5.3. Each

TABLE 5.1

Consequence Magnitudes for Tokamak Reactor Induced Activity Releases

Percentage of First Wall	UWMAK-I (316SS)			UWMAK-III (TZM)		
	latent cancer fatalities	early illnesses	early fatalities	latent cancer fatalities	early illnesses	early fatalities
2	10	0	0	2.97	0	0
4	20	0	0	5.94	0	0
6	30	0	0	8.91	0	0
8	40	0.697	0	11.9	0	0
10	50	0.724	0	14.9	0	0
20	100	6.73	0	29.7	0	0
30	150	17.8	0	44.6	0.382	0
40	200	33.2	0.134	59.4	1.30	0
50	250	46.9	2.14	74.3	2.92	0
60	300	60.0	5.60	89.2	4.53	0
70	350	76.5	9.31	104	6.32	0
80	399	129	14.7	119	8.90	0
90	448	268	20.3	134	11.7	0
100	1260	496	46.7	149	15.2	0.006

TABLE 5.2

Probability Limits for UWMAK-I Releases Established by Light Water Reactor Accident Consequences

Percentage of First Wall	PWR Probability Limits (per year)			BWR Probability Limits (per year)		
	latent cancer fatalities	early illnesses	early fatalities	latent cancer fatalities	early illnesses	early fatalities
2	10^{-5}	1.0	1.0	2×10^{-5}	1.0	1.0
4	10^{-5}	1.0	1.0	2×10^{-5}	1.0	1.0
6	10^{-5}	1.0	1.0	2×10^{-5}	1.0	1.0
8	10^{-5}	10^{-5}	1.0	2×10^{-5}	5×10^{-6}	1.0
10	10^{-5}	10^{-5}	1.0	2×10^{-5}	5×10^{-6}	1.0
20	10^{-5}	6×10^{-6}	1.0	2×10^{-5}	7×10^{-7}	1.0
30	9×10^{-6}	6×10^{-6}	1.0	2×10^{-6}	7×10^{-7}	1.0
40	7×10^{-6}	6×10^{-6}	3×10^{-7}	7×10^{-7}	7×10^{-7}	$<10^{-9}$
50	7×10^{-6}	6×10^{-6}	3×10^{-7}	7×10^{-7}	7×10^{-7}	$<10^{-9}$
60	7×10^{-6}	6×10^{-6}	3×10^{-7}	7×10^{-7}	7×10^{-7}	$<10^{-9}$
70	7×10^{-6}	6×10^{-6}	3×10^{-7}	7×10^{-7}	7×10^{-7}	$<10^{-9}$
80	7×10^{-6}	6×10^{-6}	3×10^{-7}	7×10^{-7}	7×10^{-7}	$<10^{-9}$
90	7×10^{-6}	6×10^{-6}	3×10^{-7}	7×10^{-7}	7×10^{-7}	$<10^{-9}$
100	10^{-7}	6×10^{-6}	10^{-9}	$<10^{-9}$	7×10^{-7}	$<10^{-9}$

TABLE 5.3

Probability Limits for UWMAK-III Releases Established by Light Water Reactor Accident Consequences

Percentage of First Wall	PWR Probability Limits (per year)			BWR Probability Limits (per year)		
	latent cancer fatalities	early illnesses	early fatalities	latent cancer fatalities	early illnesses	early fatalities
2	10^{-5}	1.0	1.0	2×10^{-5}	1.0	1.0
4	10^{-5}	1.0	1.0	2×10^{-5}	1.0	1.0
6	10^{-5}	1.0	1.0	2×10^{-5}	1.0	1.0
8	10^{-5}	1.0	1.0	2×10^{-5}	1.0	1.0
10	10^{-5}	1.0	1.0	2×10^{-5}	1.0	1.0
20	10^{-5}	1.0	1.0	2×10^{-5}	5×10^{-6}	1.0
30	10^{-5}	10^{-5}	1.0	2×10^{-5}	5×10^{-6}	1.0
40	10^{-5}	10^{-5}	1.0	2×10^{-5}	7×10^{-7}	1.0
50	10^{-5}	6×10^{-6}	1.0	2×10^{-5}	7×10^{-7}	1.0
60	10^{-5}	6×10^{-6}	1.0	2×10^{-5}	7×10^{-7}	1.0
70	9×10^{-6}	6×10^{-6}	1.0	10^{-5}	7×10^{-7}	1.0
80	9×10^{-6}	6×10^{-6}	1.0	7×10^{-6}	7×10^{-7}	1.0
90	8×10^{-6}	6×10^{-6}	1.0	3×10^{-6}	7×10^{-7}	1.0
100	8×10^{-6}	6×10^{-6}	3×10^{-7}	2×10^{-6}	7×10^{-7}	$<10^{-9}$

probability value is associated with one of the two first wall materials, one of the two light water reactors, a particular health effect, and with a specific release fraction. To integrate these results into a single complementary probability distribution function for each first wall material which will provide the maximum tolerable probability for accidents with release fractions equal to or greater than a particular value, a minimum probabilitiy associated with each release fraction is required. This probability is simply the minimum value found in a particular row for each release fraction in Tables 5.2 and 5.3. These minimum probability values assure that releases of induced radioactivity will not imply a greater radiological hazard than is inherent in either light water reactor type. The minimum probability values associated with the release fractions of both first wall activity inventories are presented in Table 5.4.

The probabilities in Table 5.4 represent the upper bounds of tolerable probabilities for accidents involving releases which exceed certain magnitudes. The complementary probability distribution functions developed from these tabulated values are presented in Figures 5.1 and 5.2. The regions beneath these curves can be interpreted as "safe" or "allowable" combinations of accident probability and release magnitude. Thus, tolerable risks are associated with any system design with failure rates and accident conditions which can be characterized by a curve or set of points in these regions.

TABLE 5.4

Minimum Probability Limits for Tokamak Reactor Releases Established by Light Water Reactor Accident Consequences

Percentage of First Wall	Maximum Tolerable Probability for Release (per year)	
	UWMAK-I (316SS)	UWMAK-III (TZM)
2	10^{-5}	10^{-5}
4	10^{-5}	10^{-5}
6	10^{-5}	10^{-5}
8	5×10^{-6}	10^{-5}
10	5×10^{-6}	10^{-5}
20	7×10^{-7}	10^{-5}
30	7×10^{-7}	5×10^{-6}
40	$<10^{-9}$	5×10^{-6}
50	$<10^{-9}$	7×10^{-7}
60	$<10^{-9}$	7×10^{-7}
70	$<10^{-9}$	7×10^{-7}
80	$<10^{-9}$	7×10^{-7}
90	$<10^{-9}$	7×10^{-7}
100	$<10^{-9}$	$<10^{-9}$

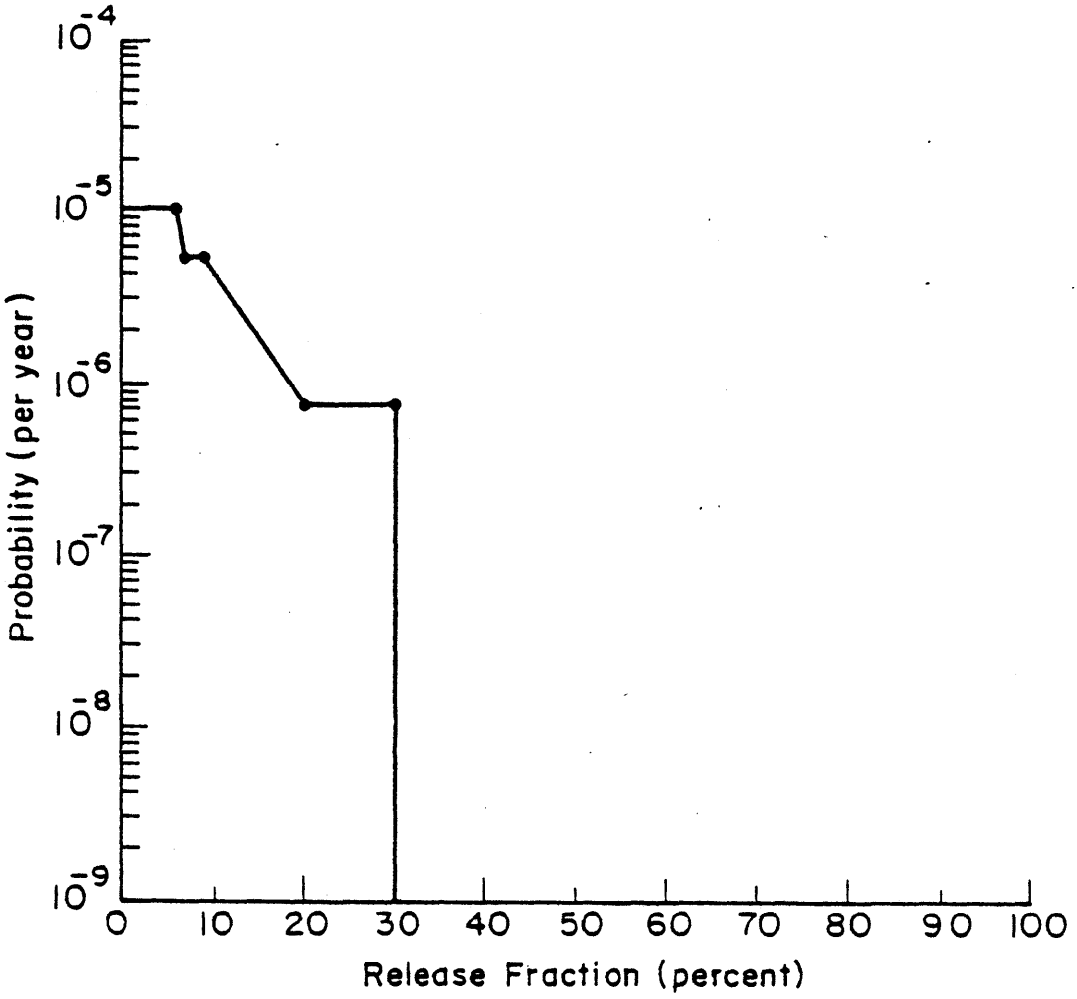


FIGURE 5.1 COMPLEMENTARY MAXIMUM TOLERABLE PROBABILITY DISTRIBUTION FUNCTION FOR UWMAK-I FIRST WALL RELEASES

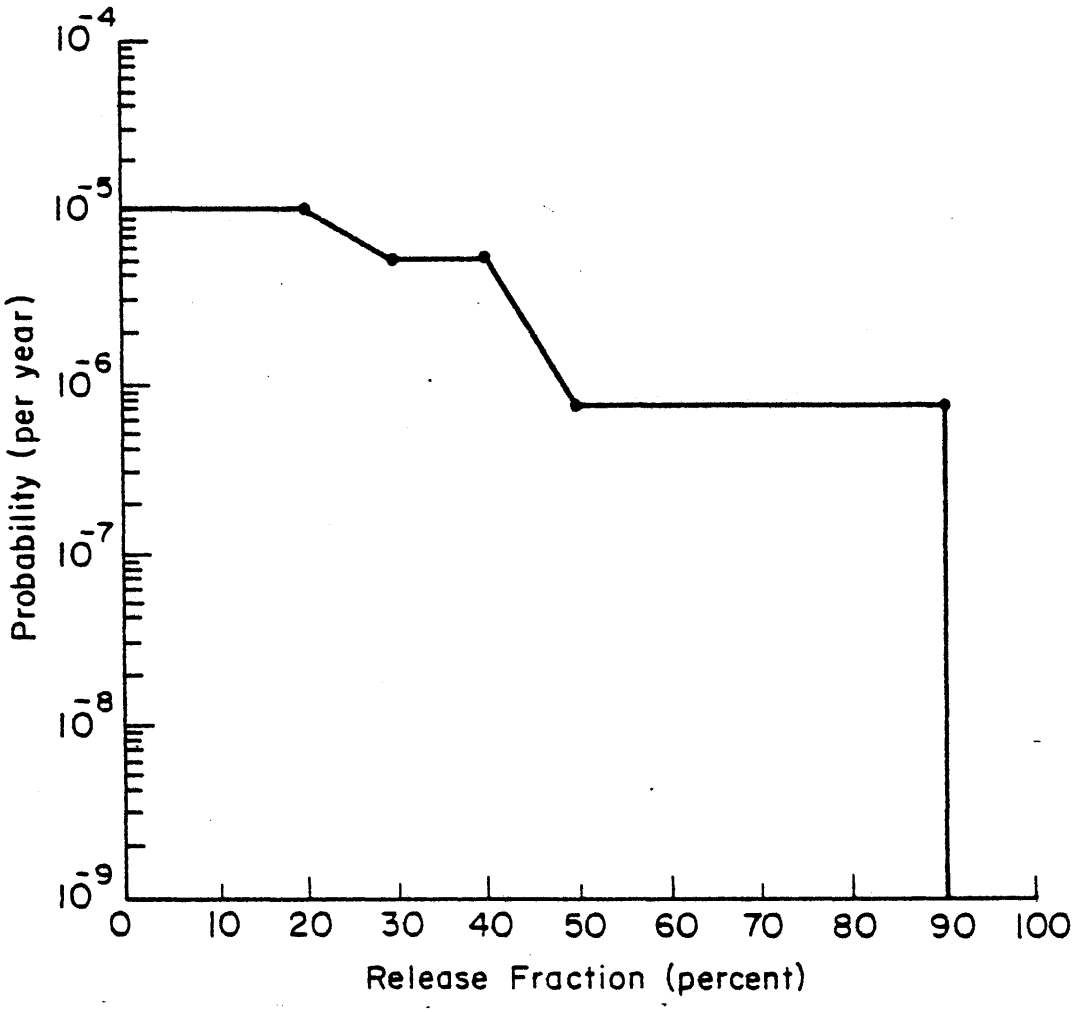


FIGURE 5.2 COMPLEMENTARY MAXIMUM TOLERABLE PROBABILITY DISTRIBUTION FUNCTION FOR UWMAK-III FIRST WALL RELEASES

An examination of the two curves reveals a greater hazard is associated with the steel first wall in UWMAK-I. Essentially no releases larger than 30% of the first wall activity are tolerable, while releases of up to 90% of the molybdenum first wall in UWMAK-III can be tolerated. However, from the discussion of Chapter III, the release of stainless steel activation products appear to be more difficult to achieve. Both of these maximum release magnitudes are deemed tolerable if the probability of their occurrence is limited to below 10^{-6} per year.

The basic characteristics of the curves are similar; they both show that all release accidents must be limited to a frequency of 10^{-5} per year, and the maximum tolerable probabilities vary between 7×10^{-7} and 10^{-5} per year. These probability values are similar to those associated with light water reactor accidents involving radioactivity releases. (See Appendix B.3) An important observation concerning the release fractions is that the maximum credible releases determined for the first wall designs analyzed in Chapter III are within the tolerable regimes.

5.2 Implications for System Reliability Requirements

The maximum tolerable accident frequencies essentially establish minimum reliability requirements for the system designs. The reliabilities of the various components must be such that the release magnitudes cannot occur with a greater frequency than is established by Figures 5.1 and 5.2. The

various designs which will be developed for Tokamak reactor systems may be analyzed to determine if they will satisfy these minimum requirements, and the need for redundancy and engineered safety features can be assessed. An important observation is that the apparent reliability that will be required in Tokamak reactor systems seems to be quite similar to the reliability inherent in present light-water reactor designs, based on considerations of the radiological hazards of induced activity. More stringent reliability requirements may be established by examination of the tritium hazard or by economic and operational considerations.

A comparison of the hazards associated with the two first wall materials, using only the results presented in Figures 5.1 and 5.2, might lead to the conclusion that the stainless steel material is inherently more hazardous. It should be noted that the volume of the first wall material is 2.22[10,11] times larger in the UWMAK-I design as compared to the first wall in UWMAK-III, therefore, the inventory of steel activation products is greater. In addition, it should be recalled that data for some of the TZM isotopes were not available, and the inventory used in this analysis represented only 78.5% of the first wall BHP as contrasted to 97.8% of the BHP for the steel wall.
If a comparison of hazards associated with equal volumes of each material is desired, an estimation for the maximum tolerable release fractions may be obtained by applying appropriate factors to the release fractions in Figures 5.1 and 5.2.

As an example of this type of comparison, the release fractions of the UWMAK-I curve can be multiplied by 2.22 to essentially reduce the material volume in the first wall. Additionally, the release fractions of the UWMAK-III curve can be multiplied by 0.785 to approximate the inclusion of the total isotope inventory BHP. The two curves are presented in Figure 5.3, and it is seen that the hazards associated with equal volumes of the two materials employed in their respective reactors are approximately similar. It is important to note at this point that the UWMAK-III design incorporates layers of carbon on the plasma side of the TZM first walls which reduce the activation rates,[11] and therefore it is expected that the hazards associated with an unprotected TZM first wall could be greater than the hazards predicted for TZM in this study.

Thus, it appears that the reliability requirements associated with the potential radiological hazards of the first wall will depend on various aspects of the first wall design, with the more obvious aspects being the volume and type of material. An attempt to reduce the first wall volume will be limited by plasma physics, neutron loading (radiation damage), corrosion or structural considerations. However, first wall volume-reduction represents an important means of minimizing the radiological hazards.

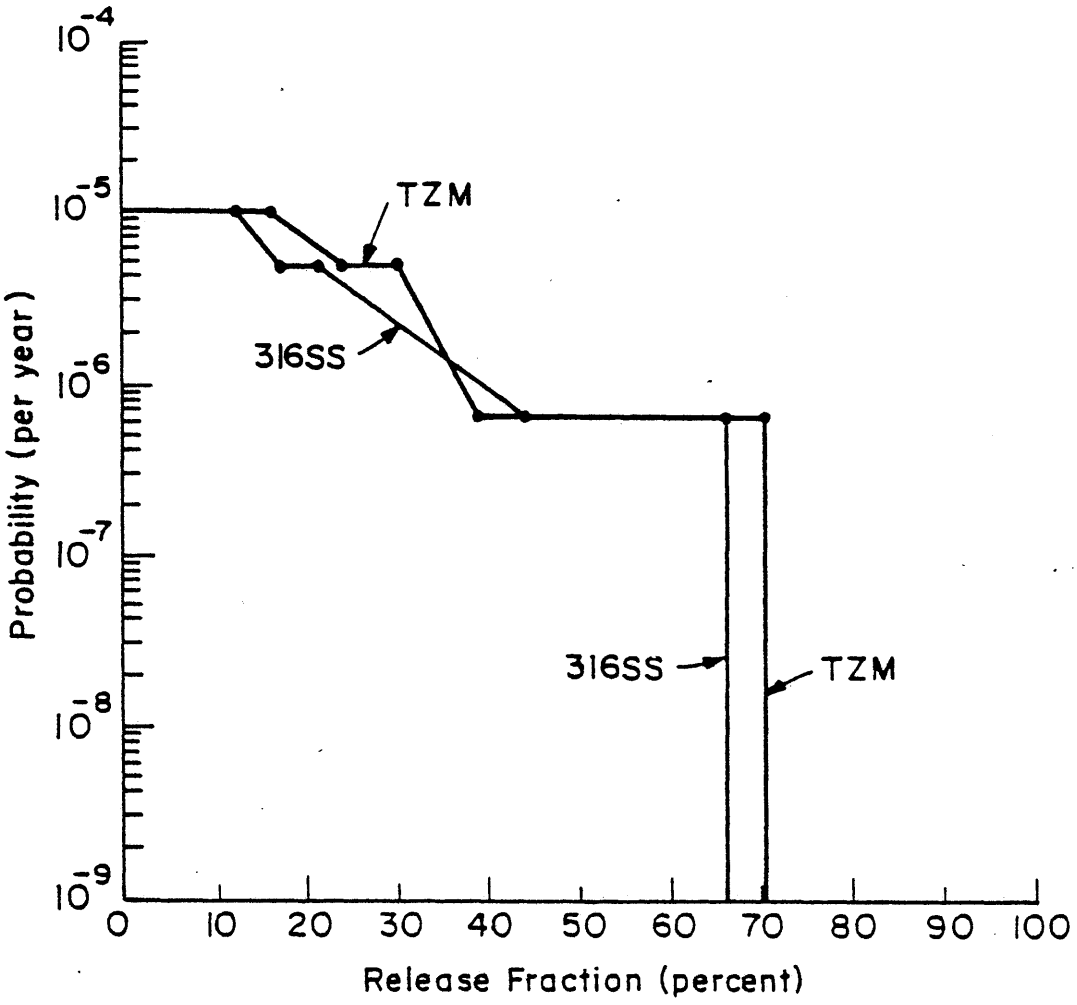


FIGURE 5.3 EQUAL VOLUME FIRST WALL HAZARD COMPARISON

(ACTIVATION OF TZM REDUCED BY CARBON LAYERS IN UWMAK-III DESIGN)

VI. CONCLUSIONS AND RECOMMENDATIONS

6.1 Conclusions

The methodology employed in this study provides a useful approach to the determination of Tokamak reactor system reliability requirements based on radiological hazard limitations. This technique may be used to investigate induced activity in reactor structural materials, activated corrosion products, or possibly even tritium. The results of an analysis are in the form of complementary probability distribution functions giving the maximum tolerable probabilities or frequencies for various accident conditions. These maximum tolerable probabilities, which are essentially the reliability requirements, will depend on the system design, the inventory and forms of radioactivity, and the radiological release consequences used to limit the hazards to an acceptable level.

The methodology is also capable of providing comparisons of potential hazards associated with various designs, or with options within a system design, such as the materials of construction. Different forms of radioactivity within a system design may represent various hazard levels, and a methodology such as this one can provide the analytical basis upon which different activity inventories are compared. Thus, the technique can be used to determine required system reliabilities and to make comparative assessments of the potential hazards of radioactivity in Tokamak fusion reactor systems.

The examination of possible releases of induced activity from a Tokamak reactor revealed that volatile oxidation appears to be the most likely mechanism to disrupt substantial amounts of radioactivity. The extent of first wall disruption is strongly dependent on the first wall material, the oxidizing gas temperature, and the flow of gas along the first wall. Lithium-air reactions seem to be capable of creating the high-temperature gases required for rapid oxidation and volatilization processes. Stainless steel appears to be less susceptible to volatile oxidation than TZM, and reductions in the gas temperatures during a lithium reaction can prevent any first wall mobilization. The extensive reactor vessel damage which seems to be required for significant gas velocities on the first wall surface may not be very likely due to the lack of a possible mechanism for creating multiple vessel ruptures for gas circulation. Altogether, it does not appear very probable that significant releases of induced activity will occur.

An examination of the potential hazards associated with the first wall induced activity in the UWMAK-I and UWMAK-III designs, revealed that the estimated possible release accidents result in consequences which are less than those associated with light water reactor accidents. In addition, the system reliability requirements established for the induced activity hazards appear to be within the capabilities of present technology, and are comparable to the reliabilities estimated for light water reactor systems.

The induced radioactivity in Tokamak reactor designs may actually present little hazard to the public since the following series of conservative assumptions were made in this study to determine maximum possible releases:

1) a mechanism exists which may lead to releases of induced activity from the containment — high-temperature volatile oxidation during a lithium-air reaction

2) the lithium inventory of an entire coolant loop (UWMAK-III design) may be spilled into an air atmosphere

3) no measures are taken to mitigate the consequences of the lithium fire

4) the vacuum chamber is ruptured with the first wall immediately exposed to the high gas temperatures of the lithium fire

5) The high gas temperatures cause catastrophic and volatile oxidation of the first wall

6) the containment is breached

7) the oxidized material may be transported out of the containment to form a plume

8) the plume forms 30 minutes after the reactor shuts down

9) the plume is in the form of a cold, ground level release, maximizing the acute effects of initial exposures

10) the hazards associated with the maximum induced activity releases must be less than the hazards associated with average light water reactor release accidents

11) isotopic tailoring was ignored.

Using the conservative possible releases estimated with the above assumptions, maximum tolerable probabilities for the worst accidents involving induced activity in UWMAK-I and UWMAK-III were between 7×10^{-6} and 10^{-5} per year. These values are an order of magnitude above the estimated probabilities for the worst light water reactor accidents, which are between 4×10^{-7} and 10^{-6} per year. In other words, it appears that the maximum credible accident for a Tokamak reactor may be allowed to occur ten times more frequently than the maximum credible light water reactor accident.

6.2 Recommendations for Further Work

Further work in this area of fusion reactor safety should involve the various aspects of induced activity and its disruption. These specific areas are:

1) Tokamak reactor designs which minimize or eliminate the liquid lithium coolant, and thereby reduce the potential and severity of lithium reactions

2) design features which could mitigate the consequences of lithium fires to eliminate catastrophic and volatile oxidation

3) lithium-concrete interactions and design features to mitigate the potential consequences

4) oxidation kinetics and oxidation resistance associated with various alloying elements

5) the hazards associated with structural materials other than stainless steel and TZM

6) possible mechanisms for achieving radioactivity releases other than volatile oxidation

Further efforts should also continue to develop a consequence model which can handle isotopes associated with induced activity. In addition, long term or chronic effects of induced activity releases should be incorporated into the dose calculation models. Dose models for activation products might be combined with a tritium dose model to integrate all potential hazards of radioactivity for Tokamak reactors into one complete radiological consequence model. Once a comprehensive model is created, hazard assessments and system reliability requirements for a great variety of release conditions is possible.

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APPENDIX A

CALCULATIONS FOR POSSIBLE RELEASES OF INDUCED ACTIVITY FROM TOKAMAK FUSION REACTORS

A.1 Time to Pressurize the Vacuum Vessel following a Rupture

To estimate the length of time to bring the pressure within a ruptured vacuum vessel to approximately the same pressure existing in the containment, the assumption of ideal isentropic compressible flow was used. The governing equation for the mass flow rate at a throat at sonic conditions is:

$$\dot{m}_{sonic} = \frac{A_{th}\, P_o}{\sqrt{T_o}} \sqrt{\frac{k}{R} \left(\frac{2}{k+1}\right)^{\left(\frac{k+1}{k+1}\right)}}$$

where;

a) $\dot{m}_{sonic}$ is the mass flow rate at sonic conditions in the throat (kg/sec)

b) A_{th} is the area of the throat (m^2)

c) P_o is the pressure in a reservoir (N/m^2)

d) T_o is the absolute temperature (°K)

e) R is the ideal gas constant $\frac{N - m}{kg-°K}$

f) k is the specific heat ratio, C_p/C_v

To determine the possible pressurization time for a vacuum chamber rupture in a lithium spill accident, a pressure of 1 atm and a temperature of 1000 °C will be assumed for the above equation and the properties of air will be substituted;

thus,

P_o = 15 psia = 103,420 N/m^2

T_o = 1000 °C = 1273 °K

k = 1.32

R = 53.3 ft-lb_f/lbm-°R = 287 N-m/kg-°K

$$\dot{m}_{sonic} = 114.8\ A_{th}\ kg/sec.$$

The criteria for complete pressure equalization will be assumed to be the time at which the mass of air contained within the chamber at 15 psia and 1000 °C, is transported by the sonic mass flow rate at the rupture, or:

$$t_{pressurization} = \frac{m_{air\ in\ chamber\ at\ 15\ psia\ and\ 1000\ °C}}{\dot{m}_{sonic}}$$

where, $m_{air\ in\ chamber} = P_o\ V_{chamber}/R_{air}\ T_o$

and,

$V_{chamber}$ = approximate volume of the UWMAK-III[11] toroidal vacuum chamber $\approx$ 3000 m^3

Thus, the length of time is estimated to be:

$$t_{press.} = \frac{849}{114.8\ A_{th}} = \frac{7.40}{A_{th}}\ \text{sec.}\ (A_{th}\ \text{in}\ m^2)$$

$$= 70{,}000/A_{th}\ \text{sec.}\ (A_{th}\ \text{is}\ cm^2)$$

This estimated function for pressurization time is presented in Figure A.1.

A.2 Estimated Time for Total Catastrophic Oxidation of 316 SS First Wall in Air at 1000 °C

The time for the entire first wall to be oxidized, assuming the uniform oxidation rate from Figure 3.4, is:

$$t_{oxid.} = \rho_{316SS}\ \delta_{first\ wall}/\ (\frac{\Delta m}{A})\ /\Delta t$$

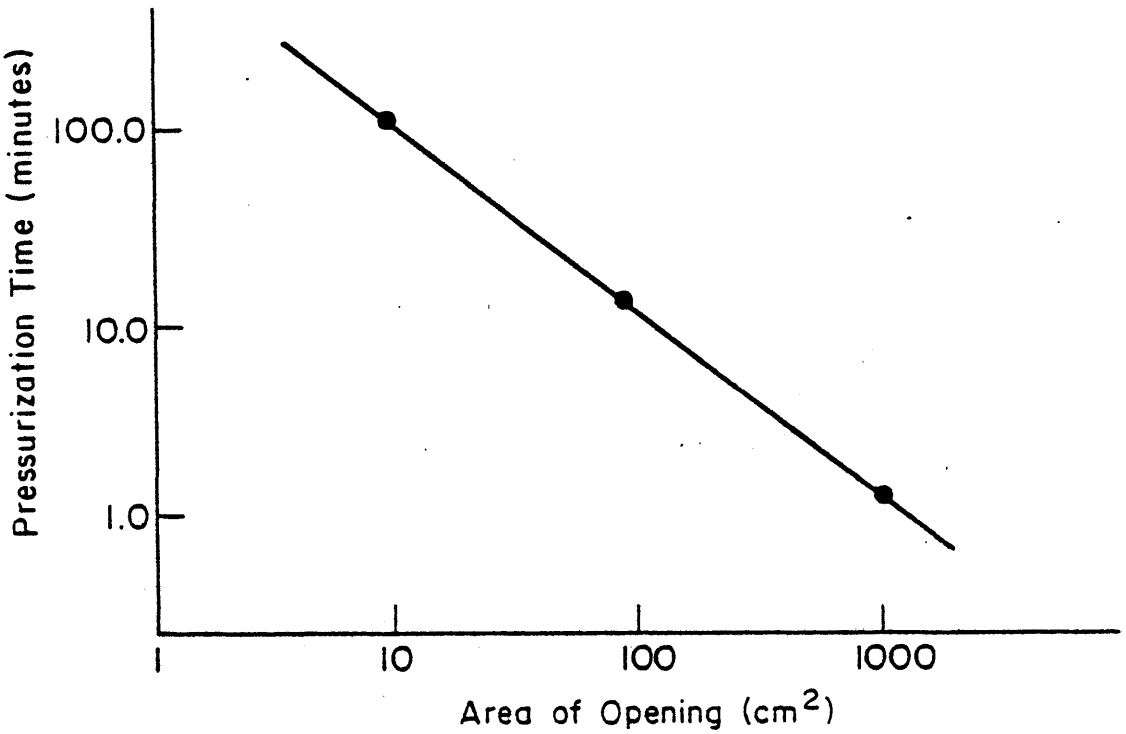

FIGURE A.1 ESTIMATED PRESSURIZATION TIME FOR RUPTURE OF VACUUM CHAMBER

where, ρ_{316SS} is the density of 316SS = 7.98 gm/cm^3

$\delta_{first\ wall}$ is the first wall thickness in UWMAK-I = 0.25 cm[10]

$(\Delta m/A)$ is the weight loss from Figure 3.4 = 112 mg/cm^2 = 0.112 gm/cm^2

Δt is the measuring time = 2 hours

thus, $$t_{oxid.} = \frac{7.98(0.25)}{0.112/2} = 36 \text{ hours}$$

A.3 Estimated Fraction of UWMAK-III First Wall Volatilization

From Figure 3.8, the simplified recession rate function is of the form:

$$\ln(\dot{x}) = a(T) + b$$

or, $$\dot{x} = \exp(a(T) + b)$$

where, $\dot{x}$ = surface recession rate (mm/m)

T = temperature (°C)

a,b = constants

Solving for the constants by substituting into the equation two points from the curve (end points);

$$a = 0.0117$$

$$b = -11.00$$

thus,

$$\dot{x} = \exp(0.0117T - 11.00)$$

The extent of first wall disruption can be expressed as

$$\int_0^{t_{total}} \dot{x}\, dt$$

or,

$$2 \int_0^{1/2 t_{total}} \exp(0.0117\, T(t) - 11.00)\, dt$$

where, t_{total} = 3000 seconds = 0.833 hours from Figure 3.7

and $T(t) = 700 + 720t$ °C, t in hours

thus, the extent of the disruption is;

$$x = 2 \int_0^{0.833/2} \exp(0.0117(700 + 720t) - 11.00)\, dt$$

$$= 2 \int_0^{0.4167} \exp(8.424t - 2.810)\, dt$$

$$= \frac{2}{8.424} \left[\exp(8.424(0.4167) - 2.810) - \exp(-2.810)\right]$$

$$= 0.2374\ [2.014 - 0.0602]$$

$$= 0.464 \text{ mm}$$

The UWMAK-III outerfirst wall thickness is 1.5 mm,[11] thus the estimated fraction of the first wall volatilized by oxidation is

$$\frac{0.464}{1.50} = 0.31 \text{ or } 31\%$$

APPENDIX B

CONSEQUENCE MODEL

INPUT PARAMETERS

B.1 Portion of First Wall Activity Inventories Incorporated into Consequence Model

Dose conversion factors for some of the isotopes given in Tables 3.1 and 3.2 were not available for this study. The BHP values associated with the missing isotopes are used to assess the amount of the total hazard not represented in the consequence model. Tables B.1 and B.2 give the isotopes that were incorporated into the consequence model for each of the reactor first wall inventories. The first wall activities and BHP values are given for each isotope, and the fraction of the total first wall BHP represented by each set of isotopes is also given.

B.2 Calculation of Initial Dose Effects

The acute effects of early exposures are the fatalities and illnesses which occur within one year of the exposure. The calculation of acute effects is based on individual organs, and for each organ considered, the consequence model accumulates the dose from each radioisotope and each exposure path; cloud shine, inhalation, and ground exposure. The acute dose effects are based on a three segment linear interpolation of dose response curves. The model obtains a probability of an acute effect based on doses to particular organs, but the process allows a person to be a fatality only once. Synergistic effects between organ effects are not included. The acute dose effect criteria are input by specifying coordinates for a dose response curve. The response curve is shown in Figure B.1,

TABLE B.1

UWMAK-I First Wall Isotope Inventory
Used in Consequence Model

Isotope	Activity ($Ci \times 10^{-4}$)	BHP (km^3 air/KWth)
Mn54	7,730	15.46
Co58	11,690	11.69
Co57	3,302	6.604
Co60	817.0	5.447
Mn56	24,030	2.403
Fe55	34,790	2.319
Ni57	687.5	1.375
Zr89	26.86	0.5372
Mo99	1,051	0.3003
Ni63	5.735	0.1147
Fe59	22.67	0.02267
Nb95	23.68	0.01579
Zr95	7.280	0.01456
Tc99m	1,051	0.004204
TOTAL	85,235	46.31

from Table 3.1:

1) represented fraction of first wall activity is:

$$85,235/100,814 = 0.85$$

2) represented fraction of first wall BHP is:

$$46.31/47.00 = 0.98$$

TABLE B.2

UWMAK-III First Wall Isotope Inventory
Used in Consequence Model

Isotope	Activity ($Ci \times 10^{-4}$)	BHP (km^3 air/KWth)
Zr89	833.4	16.67
Mo99	29,800	8.510
Nb91m	224.0	4.480
Nb95	1,033	0.6890
Zr95	219.3	0.4390
Tc99m	29,910	0.1200
Y88	5.792	0.1160
Zr97	43.92	0.0293
Nb92m	919.6	0.00497
Nb95m	655.8	0.00437
Nb97	308.1	0.00308
Sr89	0.4511	0.00301
Y91	0.7517	0.00150
Y90	2.240	0.00149
Total	63,956	3.107

From Table 3.2:

1) represented fraction of first wall activity is;

$$63,956/64,494 = 0.99$$

2) represented fraction of first wall BHP is;

$$31.07/39.60 = 0.785$$

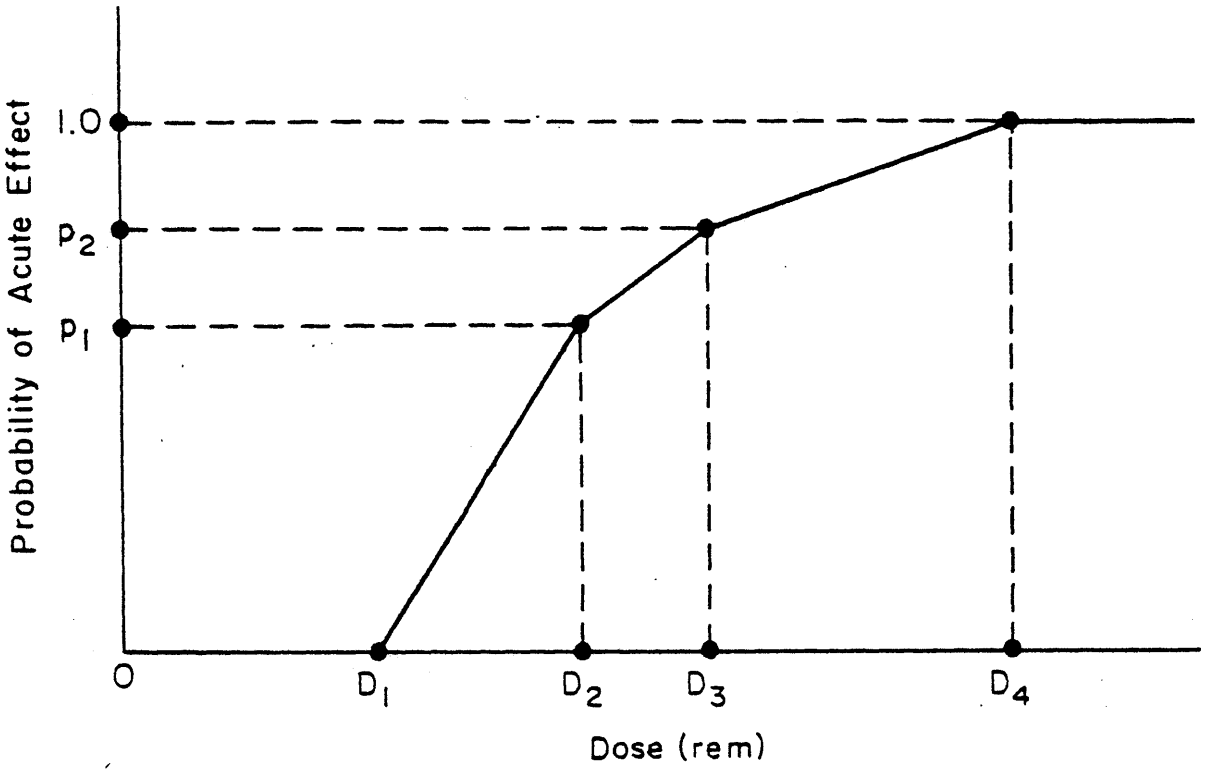

FIGURE B.1 ACUTE HEALTH EFFECTS DOSE RESPONSE CURVE (FROM REFERENCE 8)

TABLE B.3

Acute Health Effects Data for Dose Response Curve (Figure B.1)

Health Effect	Organ	Dose Break Points (rem)				Probability Values	
		D_1	D_2	D_3	D_4	P_1	P_2
Early fatalaties (death within a year after exposure)	bone marrow	320	400	510	615	0.03	0.5
	lower large intestine wall	2000	5000	5000	5000	1.0	1.0
	lungs	5000	14,800	22,400	24,000	0.24	0.73
Early illnesses (requiring serious medical attention)	whole body	55	150	280	370	0.30	0.80
	lung	3000	3000	6000	6000	0.05	1.0
	lower large intestine wall	1000	1000	2500	2500	0.05	1.0

Data from reference (8)

with the input coordinate values used in this study given in Table B.3. The Reactor Safety Study found that the important acute health effects were associated with the doses to bone marrow, lung, gastrointestinal tract, and the whole body. Appendix VI of the Reactor Safety Study should be consulted for details, and the doses given in Table B.3 are recommended values for the model.

The latent effects of early exposures are also calculated on an organ basis, though not necessarily the same ones as for the acute effects. The cloud gamma ray shine and the early ground exposure doses are accumulated for all the radio isotopes, and these are essentially instantaneous doses during the reactor accident sequence. The inhalation dose is accumulated over the life of the individual, and the model calculates a dose for 10 time periods after the accident. The "central estimate" for latent cancer fatalities due to early exposures is used in this study. This model is explained in detail in Appendix VI of the Reactor Safety Study, and it is an attempt to account for the effect of dose rate and possible thresholds. The data used in this study consisted of recommended values, and it is given in Table B.4. The dose conversions are used to determine the consequence magnitudes for various time periods, and for various organs utilizing an organ compensation factor. The total latent effects is the summation of cases over all organs and time periods.

TABLE B.4

Latent Health Effects Data

Health Effect	Organ	Time period dose conversion factors for latent health effects (cases/million man-rem) (time intervals in years)										Organ compensation factor
		<1	1-10	11-20	21-30	31-40	41-50	51-60	61-70	71-80	>80	
leukemia	Bone marrow	28.4	27.2	18.7	13.8	9.70	6.76	4.03	1.69	0.480	0.0	1.0
lung cancer	lung	22.2	22.2	22.2	14.5	8.13	3.99	1.50	0.220	0.0	0.0	0.5
bone cancer	skeleton	6.87	6.70	4.95	2.60	1.62	0.910	0.420	0.127	0.010	0.0	1.0
cancer of GI tract	lower large intestine wall	13.6	13.6	13.6	8.94	5.00	2.46	0.920	0.140	0.0	0.0	1.0
breast cancer	other organs	25.6	25.6	25.6	16.8	9.35	4.60	1.73	0.250	0.0	0.0	1000
other cancers		25.0	23.2	20.5	13.4	8.52	3.69	1.39	0.200	0.0	0.0	1.0
whole body effects	whole body	120.1	117.7	104.0	69.7	41.0	22.1	9.80	2.54	0.480	0.0	1.0

Data from reference (8)

B.3 Isotope Inventory and Release Characteristics for Light Water Reactors Used in the Consequence Model

The isotope inventory and release characteristics described in the Reactor Safety Study for light water reactors was used in this study to calculate the radiological hazard limits. The input parameters for the isotope inventory are given in Table B.5 and the PWR and BWR release characteristics are presented in Tables B.6 and B.7 respectively. The releases of radioactivity are described by isotope group release fractions, unlike the uniform releases assumed for the first wall induced activities. These tables are also helpful to compare the nature of the light water reactor releases to the characteristics assumed for first wall activity releases in this study.

TABLE B.5

Light Water Reactor Isotope Inventory

Isotope	Isotope Group	Activity (Ci x 10^{-4})
Co58	7	78.00
Co60	7	29.00
Kr85	1	56.00
Kr85m	1	2400
Kr87	1	4700
Kr88	1	6800
Rb86	4	2.600
Sr89	6	9400
Sr90	6	370.0
Sr91	6	11,000
Y90	8	390.0
Y91	8	12,000
Zr95	8	15,000
Zr97	8	15,000
Nb95	8	15,000
Mo99	7	16,000
Tc99m	7	14,000
Ru103	7	11,000
Ru105	7	7200
Ru106	7	2500
Rh105	7	4900
Te127	5	590.0
Te127m	5	110.0
Te129	5	3100
Te129m	5	530.0
Te131	5	1300
Te132	5	12,000
Sb127	5	610.0
Sb129	3	3300
I 131	3	8500
I 132	3	12,000
I 133	3	17,000
I 134	1	19,000
I 135	1	15,000
Xe133	4	17,000
Xe135	4	3400
Cs134	4	750.0
Cs136	6	300.0
Cs137	8	470.0
Ba140	8	16,000

TABLE B.5 continued

Isotope	Isotope Group	Activity (Ci x 10^{-4})
La140	8	16,000
Ce141	8	15,000
Ce143	8	13,000
Ce144	8	8,500
Pr143	8	13,000
Nd147	8	6,000
Np239	8	164,000
Pu238	8	5.700
Pu239	8	2.100
Pu240	8	2.100
Pu241	8	340.0
Am241	8	0.1700
Cm242	8	50.00
Cm244	8	2.300

Data from reference (8)

TABLE B.6

PWR Release Characteristics

Release category	Release probability (per year)	Time to release (hr)	Duration of release (hr)	Warning time (hr)	Sensible heat release rate (cal/s)	Release height (m)	Isotope group release fractions							
							1	2	3	4	5	6	7	8
PWR1A	5×10^{-7}	2.5	0.5	1.0	3.62×10^{7}	25.0	0.9	6×10^{-3}	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR1B	4×10^{-7}	2.5	0.5	1.0	1.19×10^{6}	25.0	0.9	6×10^{-3}	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR2	8×10^{-6}	2.5	0.5	1.0	1.18×10^{7}	10.0	0.9	7×10^{-3}	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR3	4×10^{-6}	5.0	1.5	2.0	4.20×10^{5}	10.0	0.8	6×10^{-3}	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR4	5×10^{-7}	2.0	3.0	2.0	7.00×10^{3}	10.0	0.6	2×10^{-3}	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR5	7×10^{-7}	2.0	4.0	1.0	2.10×10^{4}	10.0	0.3	2×10^{-3}	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR6	6×10^{-6}	12.0	10.0	1.0	0.0	10.0	0.3	2×10^{-3}	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR7	4×10^{-5}	10.0	10.0	1.0	0.0	10.0	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR8	4×10^{-5}	0.5	0.5	0.0	0.0	10.0	2×10^{-3}	5×10^{-6}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0.0	0.0
PWR9	4×10^{-4}	0.5	0.5	0.0	0.0	10.0	3×10^{-6}	7×10^{-9}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0.0	0.0

Data from reference (8)

TABLE B.7

BWR Release Characteristics

Release category	Release probability (per year)	Time to release (hr)	Duration of release (hr)	Warning time (hr)	Sensible heat release rate (cal/s)	Release height (m)	Isotope group release fractions							
							1	2	3	4	5	6	7	8
BWR1	$1x10^{-6}$	2.0	0.5	1.5	8.40×10^{6}	25.0	1.0	$7x10^{-3}$	0.4	0.4	0.7	0.05	0.5	$5x10^{-3}$
BWR2	$6x10^{-6}$	30.0	3.0	2.0	1.89×10^{6}	10.0	1.0	$7x10^{-3}$	0.9	0.5	0.3	0.1	0.03	$4x10^{-3}$
BWR3	$2x10^{-5}$	30.0	3.0	2.0	1.40×10^{6}	25.0	1.0	$7x10^{-3}$	0.1	0.1	0.3	0.01	0.02	$4x10^{-3}$
BWR4	$2x10^{-6}$	5.0	2.0	2.0	0.0	25.0	0.6	$7x10^{-4}$	$8x10^{-4}$	$5x10^{-3}$	$9x10^{-3}$	$6x10^{-4}$	$6x10^{-4}$	$1x10^{-4}$
BWR5	$1x10^{-4}$	3.5	5.0	0.0	0.0	150.0	$5x10^{-4}$	$2x10^{-9}$	$6x10^{-11}$	$4x10^{-9}$	$8x10^{-12}$	$8x10^{-14}$	0.0	0.0

Data from reference (8)