AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100

Silver Spring, Maryland 20910

301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

1 of 2

**In-Vessel ITER Tubing Failure Rates
for Selected Materials and Coolants**

T. D. Marshall^a
L. C. Cadwallader

Published March 1994

**Idaho National Engineering Laboratory
EG&G Idaho, Inc.
Idaho Falls, Idaho 83415-3880**

^aRensselaer Polytechnic Institute, Troy, New York 12180

Prepared for the
U.S. Department of Energy
Office of Energy Research
Under DOE Idaho Field Office
Contract DE-AC07-76ID01570

MASTER

ABSTRACT

Several materials have been suggested for fabrication of ITER in-vessel coolant tubing: beryllium, copper, Inconel, niobium, stainless steel, titanium, and vanadium. This report generates failure rates for the materials to identify the best performer from an operational safety and availability perspective. Coolant types considered in this report are helium gas, liquid lithium, liquid sodium, and water. Failure rates for the materials are generated by including the influence of ITER's operating environment and anticipated tubing failure mechanisms with industrial operating experience failure rates. The analyses define tubing failure mechanisms for ITER as: intergranular attack, flow erosion, helium induced swelling, hydrogen damage, neutron irradiation embrittlement, cyclic fatigue, and thermal cycling. K-factors, multipliers, are developed to model each failure mechanism and are applied to industrial operating experience failure rates to generate tubing failure rates for ITER. The generated failure rates identify the best performer by its expected reliability. With an average leakage failure rate of $3.1\text{e-}10 \text{ (m-hr)}^{-1}$ and an average rupture failure rate of $3.1\text{e-}11 \text{ (m-hr)}^{-1}$, titanium proved to be the best performer of the tubing materials. The failure rates generated in this report are intended to serve as comparison references for design safety and optimization studies. Actual material testing and analyses are required to validate the failure rates.

SUMMARY

ITER in-vessel tubing reliability is an important safety and operational availability concern since a Loss of Coolant Accident (LOCA) or Loss of Vacuum Accident (LOVA) can result from the ingress of coolant into the vacuum vessel. Such accidents may release large quantities of energy and radioactive isotopes and are, therefore, hazardous both to the machine and its operations personnel. To provide insight into the likelihood of LOCAs, LOVAs, and other tubing related accidents, the authors generate failure rates for the eight leading material candidates for ITER in-vessel coolant tubing.

Generating failure rates for a tubing system is a complicated procedure because the rates are dependent upon system specific operating factors. Such factors include the physical characteristics of the system, such as tubing length, diameter and wall thickness, and the stresses produced by the system's environment. Thus, the failure rate for a piping system has the integral features of the system incorporated into its value.

The intimate relation between a system and its failure rate presents a formidable problem when generating tubing failure rates for ITER. The problem arises because both the physical dimensions of the tubing system and operating environment within ITER are unknown. This situation suggests an inherent level of uncertainty associated with failure rates generated for a reactor design concept.

The level of uncertainty can be reduced by using conservative factors in the failure rate calculations and providing error factors for the generated failure rate. The authors use both methods, however, the level of uncertainty is reduced and not eliminated. Accordingly, failure testing of the materials analyzed in this report is necessary for validation of the generated failure rates.

The tubing materials considered in this report are beryllium, copper, ferritic steel, Inconel 600, niobium, stainless steel Type 316, titanium, vanadium and zirconium. This list includes materials currently being considered by the ITER design team: beryllium, copper, stainless steel 316, and vanadium. Ferritic steel and Inconel are included because of their favorable performance in previous fission power reactors and fusion technology experiments. Niobium is included because it has favorable strength properties at high temperatures, which would benefit the ITER design.

At the time of this report, no published operating experience data for niobium or vanadium was found. To obtain a perspective on these materials' failure rate as in-vessel tubing, their failure rates are cross referenced with zirconium and titanium, respectively. This cross reference is based on niobium's adjacent position to zirconium and vanadium's adjacent position to titanium in the periodic table. The four materials are refractory metals and are assumed to have similar failure mechanism resistance properties at elevated temperatures.

Individual analyses are done for niobium and zirconium so K-factors are generated for the materials. However, failure rates for the two materials are based on zirconium and titanium

operating experience failure rates. These failure rates are denoted as "quasi" failure rates and are intended to serve as an order-of-magnitude approximation until actual material testing can be performed.

The authors consider the tubing failure rate to have two primary factors: the influence of application specific failure mechanisms and the influence of the coolant type. Failure mechanisms defined for ITER in-vessel tubing are: cyclic fatigue, flow erosion, helium-induced swelling, hydrogen damage, intergranular attack, irradiation embrittlement, magnetic field interactions, and thermal cycling. The coolant types considered in the analyses are: helium gas, liquid lithium, liquid sodium, and water.

Each failure mechanism and coolant type is individually analyzed to determine their influence upon the materials' failure rate. This influence is incorporated into industrial operating experience failure rates for the materials by the application of K-factors. K-factors are used since they base the failure rates generated for ITER on actual operating experience failure rates.

The generated failure rates are in units of per meter hour, $(m-hr)^{-1}$, and represent the two accident scenarios possible for a tubing system: leakage and rupture. The leakage accident injects a critical volume of coolant into the vacuum vessel and causes a plasma disruption, the critical volume being that volume of coolant sufficient to trigger a disruption. The rupture accident injects coolant into the vacuum vessel at a rate equivalent to the normal operating flowrate. The authors subscribe to the leakage-before-rupture failure philosophy, which says that the tubing system will leak before rupturing.

ITER is still in the Engineering Design Activity (EDA) phase so there are limited system operating characteristic data. As mentioned earlier, the operating characteristics are an integral component of the failure rate. Therefore, a few basic operating assumptions are required to generate tubing failure rates. These assumptions are:

- ◆ In-vessel tubing has an inner diameter less than 19 mm and a wall thickness less than 2 mm.
- ◆ ITER has a pulse length of 1000 seconds.
- ◆ Coolant inlet temperature is in the range 20 to 400 °C. (The lower limit is for water coolant considerations.)
- ◆ Coolant outlet temperature is in the range 150 to 600 °C. (The lower limit is for water coolant considerations.)
- ◆ High pulse frequency.
- ◆ Long system downtime between pulse cycles.

- ◆ Coolant flows continuously, during pulse and no pulse cycles.

The coolant inlet and outlet temperatures are necessary for selecting the temperature range for studying the failure properties of the tubing materials. The water coolant system is assumed to have temperatures between 20 and 150 °C while the other coolant systems are assumed to have temperatures between 400 and 600 °C. The temperatures are based on ITER operating as a power reactor and requiring the higher temperatures for thermal efficiency. In the experimental operation of ITER, the assumed coolant temperatures may result in more conservative failure rates.

The last assumption, continuous coolant flow, is important since it implies that particulates in the coolant stream are not allowed to settle or conglomerate and enhance the activity of the failure mechanisms.

Tables 1 through 4 present the generated failure rates for ITER in-vessel tubing. The table gives leakage and rupture failure rates for each material with the various coolant types. The table also provides error factors for the generated failure rates. The error factor is a measure of the level of confidence associated with the failure rate. An error factor value less than or equal to three indicates a high level of confidence in the generated failure rate while higher error factor values indicate correspondingly decreasing levels of confidence.

The best performer of the material group is chosen by the material with the lowest average failure rate for all four coolant types. The analyses identified titanium as the best performer with an average leakage failure rate of $3.1\text{e-}10 \text{ (m-h)}^{-1}$ and average rupture failure rate of $3.1\text{e-}11 \text{ (m-h)}^{-1}$.

In summary, the authors use reliability prediction techniques to identify the best safety and operational availability performer from the list of suggested ITER in-vessel tubing materials. Using the criteria of lowest average failure rate, titanium is chosen as the best performer.

The failure rates produced in this report are intended to serve as comparison references for design safety and optimization studies. Actual material testing and analyses are required to validate the failure rates.

Table 1
Failure Rates for ITER Helium Coolant System

Tubing Material	Leakage (m-h)⁻¹	Rupture (m-h)⁻¹	Error Factor
Beryllium	6.5e-04	6.5e-05	10
Copper	7.5e-06	7.5e-07	3
Ferritic Steel	5.4e-09	5.4e-10	10
Ferritic Duplex Steel	9.2e-10	9.2e-11	10
Inconel 600	1.5e-07	4.0e-11	10
Niobium (quasi)	2.2e-08	2.2e-09	10
316 ss	1.4e-08	1.4e-09	3
Titanium	1.0e-09	1.0e-10	10
Vanadium (quasi)	1.2e-09	1.2e-10	10
Zirconium	1.1e-08	1.1e-09	3

Table 2
Failure Rates for ITER Liquid Lithium Coolant System

Tubing Material	Leakage (m-h)⁻¹	Rupture (m-h)⁻¹	Error Factor
Beryllium	6.5e-04	6.5e-05	10
Copper	n/a	n/a	n/a
Ferritic Steel	1.9e-09	1.9e-10	10
Ferritic Duplex Steel	9.2e-10	9.2e-11	10
Inconel 600	1.7e-07	4.7e-11	10
Niobium (quasi)	2.2e-08	2.2e-09	10
316 ss	1.4e-08	1.4e-09	3
Titanium	1.0e-09	1.0e-10	10
Vanadium (quasi)	8.9e-10	8.9e-11	10
Zirconium	1.1e-08	1.1e-09	3

Table 3
Failure Rates for ITER Liquid Sodium Coolant System

Tubing Material	Leakage (m-h)⁻¹	Rupture (m-h)⁻¹	Error Factor
Beryllium	6.5e-04	6.5e-05	10
Copper	n/a	n/a	n/a
Ferritic Steel	1.9e-09	1.9e-10	10
Ferritic Duplex Steel	9.2e-10	9.2e-11	10
Inconel 600	1.5e-07	4.0e-11	10
Niobium (quasi)	1.9e-08	1.9e-09	10
316 ss	1.4e-08	1.4e-09	3
Titanium	1.0e-09	1.0e-10	10
Vanadium (quasi)	1.0e-09	1.0e-10	10
Zirconium	1.5e-08	1.5e-09	3

Table 4
Failure Rates for Water Coolant System

Tubing Material	Leakage (m-h)⁻¹	Rupture (m-h)⁻¹	Error Factor
Beryllium	6.5e-04	6.5e-05	10
Copper	6.4e-06	6.4e-07	3
Ferritic Steel	5.4e-09	5.4e-10	10
Ferritic Duplex Steel	9.2e-10	9.2e-11	10
Inconel 600	1.5e-07	4.0e-11	10
Niobium (quasi)	2.2e-08	2.2e-09	10
316 ss	1.2e-08	1.2e-09	3
Titanium	9.5e-10	9.5e-11	10
Vanadium (quasi)	1.0e-09	1.0e-10	10
Zirconium	1.0e-08	1.0e-09	3

TABLE OF CONTENTS

ABSTRACT	ii
SUMMARY	iii
NOMENCLATURE	xi
1. INTRODUCTION	1
2. COOLANT TYPES	
2.1 Introduction.	6
2.2 Helium	6
2.3 Alkali Metals	7
2.3.1 Lithium	8
2.3.2 Sodium	8
2.4 Water	8
2.5 Summary	9
Chapter References	11
3. IN-VESSEL TUBING FAILURE MECHANISMS	
3.1 Introduction.	12
3.2 Creep	12
3.3 Cyclic Fatigue	13
3.4 Erosion	13
3.5 Helium Swelling	15
3.6 Hydrogen Damage	15
3.7 Irradiation Embrittlement	16
3.8 Magnetic Field Interactions.	17
3.9 Oxidation	17
3.10 Intergranular Attack	17
3.10.1 Intergranular Corrosion	18
3.10.2 Stress-Corrosion Cracking.	18
3.11 Thermal Cycling	20
3.12 Other Mechanisms	20
3.12.1 Electric Field Interactions	20
3.12.2 Geometry	20
3.12.3 Runaway Electrons	21
3.13 Summary	21
Chapter References	22

4. MATERIAL ANALYSIS

4.1	Introduction	.23
4.2	Beryllium	.23
4.2.1	Failure Mechanism Resistance	.24
4.2.2	Helium Coolant	.25
4.2.3	Liquid Metal Coolant	.25
4.2.4	Water Coolant.	.26
4.3	Copper	.26
4.3.1	Failure Mechanism Resistance	.26
4.3.2	Helium Coolant	.26
4.3.3	Liquid Metal Coolant	.27
4.3.4	Water Coolant.	.27
4.4	Nickel-Based Alloys (Inconel 600)	.27
4.4.1	Failure Mechanism Resistance	.27
4.4.2	Helium Coolant	.28
4.4.3	Liquid Metal Coolant	.28
4.4.4	Water Coolant.	.29
4.5	Niobium	.30
4.5.1	Failure Mechanism Resistance	.30
4.5.2	Helium Coolant	.31
4.5.3	Liquid Metal Coolant	.31
4.5.4	Water Coolant.	.32
4.6	Stainless Steel (Type 316)	.32
4.6.1	Failure Mechanism Resistance	.33
4.6.2	Helium Coolant	.33
4.6.3	Liquid Metal Coolant	.34
4.6.4	Water Coolant.	.35
4.7	Titanium	.35
4.7.1	Failure Mechanism Resistance	.35
4.7.2	Helium Coolant	.36
4.7.3	Liquid Metal Coolant	.36
4.7.4	Water Coolant.	.36
4.8	Vanadium.	.36
4.8.1	Failure Mechanism Resistance	.37
4.8.2	Helium Coolant	.37
4.8.3	Liquid Metal Coolant	.37
4.8.4	Water Coolant.	.38
4.9	Zirconium.	.38
4.9.1	Failure Mechanism Resistance	.39
4.9.2	Helium Coolant	.39
4.9.3	Liquid Metal Coolant	.39
4.9.4	Water Coolant.	.39
4.10	Chapter Summary	.40
	Chapter References	.43

5. DATA COMPILATION

5.1	Introduction	49
5.2	Beryllium	50
5.3	Copper	51
5.4	Ferritic Steel	52
5.5	Ferritic Steel Duplex Tubes	53
5.6	Ferritic Steel Double-Walled Tubes	53
5.7	Inconel 600	53
5.8	Niobium	55
5.9	Type 316 Stainless Steel	55
5.10	Titanium	56
5.11	Vanadium	56
5.12	Zirconium	56
5.13	Other Tubing Issues	57
5.13.1	Double-Walled Tubing	57
5.13.2	Welds	58
5.13.3	Tubes Plugging Closed	58
5.13.4	Multiple Tube Leaks	60
5.14	Chapter Summary	60
	Chapter References	65

6. RELIABILITY PREDICTION - K FACTORS

6.1	Introduction	68
6.2	K Factor Multipliers	68
6.3	K Factors	69
6.4	K3 Factors	71
	Chapter References	78

7. FAILURE RATES FOR ITER IN-VESSEL TUBING

7.1	Introduction	79
7.2	Failure Rate Tables	79
7.3	Failure Rate Graphs	81

8. CONCLUSIONS

APPENDIX A	Investigation of Selected Accident Frequencies for Helium, Water, and Liquid Metal ITER Coolants.	A-1
------------	--	-----

NOMENCLATURE

The following is a list of the acronyms, initialisms, and abbreviations (including units of measure) used in this document.

ACRONYMS, INITIALISMS, AND ABBREVIATIONS

bcc	body centered cubic
BCSS	Blanket Comparison and Selection Study
BMI	Battelle Memorial Institute
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium-Uranium Reactor
CDA	Conceptual Design Activity
dpa	displacements per atom
DBTT	Ductile-to-Brittle Transition Temperature
DNB	Departure from Nucleate Boiling
EBR-II	Experimental Breeder Reactor-II
EDA	Engineering Design Activity
ETR	Engineering Test Reactor
HCF	High Cycle Fatigue
HFIR	High Flux Isotope Reactor
ITER	International Thermonuclear Experimental Reactor
LCF	Low Cycle Fatigue
LMFBR	Liquid Metal Fast Breeder Reactor
LOCA	Loss of Coolant Accident
LOVA	Loss of Vacuum Accident
LWR	Light Water Reactor
MARS	Mirror Advanced Reactor Study
MHD	Magnetohydrodynamic force
MTR	Materials Testing Reactor
NSSS	Nuclear Steam Supply System
ORNL	Oak Ridge National Laboratory
PCA	Primary Candidate Alloy
PFC	Plasma Facing Components
ppm	parts per million
PWR	Pressurized Water Reactor
SCC	Stress Corrosion Cracking
SRC	Stress-Relief Cracking
TF	Toroidal Field
TFTR	Tokamak Fusion Test Reactor
UTS	Ultimate Tensile Strength

UNITS OF MEASURE

cm	centimeter(s)
cm ³	cubic centimeters(s)
Ci	curie(s)
d	day(s)
g	gram(s)
h	hour(s)
kg	kilogram(s)
L	liter(s)
m	meter(s)
MPa	megapascal(s)
m ²	square meter(s)
m ³	cubic meter(s)
min	minute(s)
mm	millimeters(s)
mRem	millirem(s)
ms	millisecond(s)
Pa	Pascal(s)
s	second(s)
yr	year(s)

1.0 INTRODUCTION

Every ITER component and system must ultimately answer the question of reliability. The issue is especially true for in-vessel piping systems since their failure can cause significant damage to the reactor and injury to operations personnel. Thus, reliability data can greatly help ITER design engineers in optimizing the reactor's design. The objective of the authors is to generate reliability data for the ITER in-vessel coolant tubing system.

The reliability of a system can be expressed in terms of a failure rate where the failure rate is the number of system failures in a given period. In-vessel coolant tubing for ITER must have a very low failure rate because its failure can have such serious repercussions. Tubing system failure can produce a Loss of Coolant Accident (LOCA), Loss of Flow Accident (LOFA), or Loss of Vacuum Accident (LOVA). To better understand the consequences of the accidents, a brief overview of each is required.

A LOVA normally results from a breach in the vacuum vessel or one of its ports. This breach allows air to enter the plasma chamber and disrupt the plasma, causing the plasma energy to be deposited in a localized area within the chamber. This intense and sudden deposition of energy and chemical reactions with air may release activation products formerly contained by the plasma facing components (PFCs), increase site dose levels, and damage or destroy the PFCs. The higher site dose levels will threaten the safety of operations personnel and the damaged PFCs will decrease the reactor's availability. It is important to note that the vacuum vessel may be breached by many means. A breach may occur from the recoil of tubing, "pipe whip," after it ruptures; impact of ruptured tubing with adjacent tubing, creating cascading tubing failures and multiple assaults upon the vessel; and loss of seal-tightness as tubing movement applies stress on the tubing-to-vessel seal.

A LOCA is the consequence of tubing leakage or rupture and allows coolant to enter the vacuum vessel. The LOCA can be internal or external, depending on the location of the leakage or rupture in reference to the vacuum vessel. While both the internal and external LOCA poses significant threats to the reactor by the loss of a heat removing medium, the internal LOCA also allows coolant reactions with the PFCs. These reactions work to release activation products that are normally contained by the PFCs.

The consequences of a LOCA are dependent upon the reactor coolant type. This is true because coolant-PFC reactions are defined by the coolant type. Theoretically with a water coolant type, a LOCA can over-pressurize the vacuum vessel and radioactivity confinement building due to the buildup of gases produced by the coolant-PFC reactions. This over-pressurization may eventually lead to an explosion if the coolant-PFC chemical reaction is particularly violent¹. The LOCA also creates high energy levels due to the internal energy of the coolant entering the vacuum vessel and the energy produced by the coolant-PFC

¹ Internal energy as used here refers to the thermodynamic energy available due to the pressure and temperature of the coolant.

reactions. Finally, a LOCA may cause cryogen boiling in the superconducting magnets since the vessel loses its heat removal capability.

The LOFA is similar to an external LOCA, the difference being that coolant is not lost from the system. A LOFA normally occurs from an obstruction in the coolant tubing. The major concern with a LOFA is the reactor's inability to remove heat from part or the whole of the vacuum vessel. As the temperature rises inside the vessel, the PFCs may lose their structural integrity. The elevated temperatures may also release activation products normally contained by the PFCs.

Of the three accidents described above, the LOCA is the most important from a tubing system design perspective. The LOFA is usually a maintenance concern, i.e., proper coolant filtration, coolant chemistry control, pump availability, and etc. The LOFA that is not caused by an external source, occurs from tubing movement after a LOCA. Therefore, ITER requires an in-vessel tubing system with the lowest possible probability of initiating a LOCA. This dictates an in-vessel tubing system with a low probability of leakage and rupture.

Generating a failure rate for a tubing system is a complicated procedure because the rate is dependent upon factors that are often specific to the system's application. Such factors include the physical characteristics of the system, such as tubing length, diameter and wall thickness, and the stresses produced by the system's environment. Thus, the failure rate for a piping system has the integral details of the system incorporated into its value.

The intimate relation between a system and its failure rate presents a formidable problem when generating tubing failure rates for ITER. The problem arises because both the physical dimensions of the tubing system and operating environment within ITER are unknown. This situation suggests an inherent level of uncertainty with the generated failure rates. The level of uncertainty can be reduced by using conservative factors in the failure rate calculations and providing error factors for the generated failure rate. The authors use both methods, however, the level of uncertainty is reduced and not eliminated. Accordingly, failure testing of the materials analyzed in this report is necessary for validation of the generated failure rates.

The tubing materials considered in this report are beryllium, copper, ferritic steel, Inconel 600, niobium, stainless steel Type 316, titanium, vanadium and zirconium. This list includes materials currently being considered by the ITER design team: beryllium, copper, stainless steel 316, and vanadium. Ferritic steel and Inconel are included are included in the analyses because of their favorable performance in previous fission power reactors and fusion technology experiments. Niobium is included because it has favorable strength properties at high temperatures, which would benefit the ITER design.

At the time of this report, published operating experience data for niobium or vanadium were not found. To obtain a perspective on these materials' failure rate as in-vessel tubing, their failure rates are cross referenced with zirconium and titanium, respectively. This cross

reference is based on niobium's adjacent position to zirconium and vanadium's adjacent position to titanium in the periodic table. The four materials are refractory metals and are assumed to have similar failure mechanism resistance properties at elevated temperatures.

Individual analyses are done for niobium and zirconium and K-factors for the materials are developed. However, failure rates for the two materials are based on zirconium and titanium operating experience failure rates. These failure rates are denoted as "quasi" failure rates and are intended to serve as an order-of-magnitude approximation until actual material testing can be performed.

The authors consider the tubing failure rate to have two primary factors: the influence of application specific failure mechanisms and the influence of the coolant type. Failure mechanisms defined for ITER in-vessel tubing are: cyclic fatigue, flow erosion, helium-induced swelling, hydrogen damage, intergranular attack, irradiation embrittlement, magnetic field interactions, and thermal cycling. The coolant types considered in the analyses are: helium gas, liquid lithium, liquid sodium, and water.

Each failure mechanism and coolant type are individually analyzed to determine their influence upon the materials' failure rate. This influence is incorporated into industrial operating experience failure rates for the materials by the application of K-factors. K-factors are used by the authors since the factors base the failure rates generated for ITER on actual operating experience failure rates.

The generated failure rates are in units of per meter hour, $(m-hr)^{-1}$, and represent the two accident scenarios possible for a tubing system: leakage and rupture. The leakage accident injects a critical volume of coolant into the vacuum vessel and causes a plasma disruption, the critical volume being that volume of coolant sufficient to trigger a disruption. The rupture accident injects coolant into the vacuum vessel at a rate equivalent to the normal operating flowrate. The authors subscribe to the leakage-before-rupture failure philosophy, which says that the tubing system will leak before rupturing.

ITER is still in the Engineering Design Activity (EDA) phase so there are limited system operating characteristic data. As mentioned earlier, the operating characteristics are an integral component of the failure rate. Therefore, a few basic operating assumptions are necessary to generate tubing failure rates. These assumptions are:

- ♦ In-vessel tubing has an inner diameter less than 19 mm and a wall thickness less than 2 mm.
- ♦ ITER has a pulse length of 1000 seconds.
- ♦ Coolant inlet temperature is in the range 20 to 400 °C. (The lower limit is for water coolant considerations.)

- ◆ Coolant outlet temperature is in the range 150 to 600 °C. (The lower limit is for water coolant considerations.)
- ◆ High pulse frequency.
- ◆ Long system downtime between pulse cycles.
- ◆ Coolant flows continuously, during pulse and no pulse cycles.

The coolant inlet and outlet temperatures are necessary for selecting the temperature range for studying the failure properties of the tubing materials. The water coolant² system is assumed to have temperatures between 20 and 150 °C while the other coolant systems are assumed to have temperatures between 400 and 600 °C. The temperatures are based on ITER operating as a power reactor and requiring the higher temperatures for thermal efficiency. In the experimental operation of ITER, the assumed coolant temperatures may result in more conservative failure rates.

The last assumption is important since it implies that particulates in the coolant stream are not allowed to settle or conglomerate and enhance the activity of the failure mechanisms.

The following chapters describe the analyses and procedures used to generate in-vessel tubing failure rates. Chapter 2 discusses the coolant types and their respective advantages and disadvantages. Chapter 3 defines the tubing failure mechanisms anticipated for ITER. Chapter 4 discusses the proposed tubing material and their respective failure resistance to the defined failure mechanisms and coolant types. Chapter 5 is the operating experience failure data compiled from published industrial literature. Chapter 6 is a discussion of K-factors and the variables used for this report. Chapter 7 presents the generated failure rates and Chapter 8 concludes and summarizes the results of this report.

Chapter References

1. Petti, D.A., Holland, D.F., Piet, S.J., McCarthy, K.A., and Cadwallader, L.C., Key Safety Issues Related To Coolant Selection For ITER, ITER/93/EN/SA-2, Idaho National Engineering Laboratory, February 5, 1993.
2. McGrath, Robert and et al., Plasma Facing Components Generic Facilities Review Panel (PFC-GFRP), Final Report, SAND93-1987, UC-423, Oct 1993, page 41.

2.0 COOLANT TYPES

2.1 Introduction

Many performance and safety issues for ITER will be decided by the coolant type. The primary function of the coolant is the efficient removal of heat from the first wall, diverter, and blanket area during ITER operation. The coolant system must also remove heat during baking and conditioning procedures¹. Coolant type influences tubing reliability since the coolant has intimate contact with the tubing material. Accordingly, failure rate generation must include an analysis of coolant type.

At the time of this report, several coolant types are candidates for ITER use: helium gas, liquid lithium, liquid sodium, and water. The objective of this chapter is to present an overview of the coolant types and their respective operational and safety concerns.

2.2 Helium

Helium is often suggested as a blanket coolant because it is chemically inert, essentially transparent to neutrons, and does not activate or experience radiolysis effects. An argument may be presented on the transparency to neutrons issue since the benefit of no activation products may be offset by the additional shielding requirement. Helium's chemical inertness implies minimal levels of corrosion in air or water and virtually no reactivity with PFCs. As an asphyxiant, helium rises in air, reducing hazards to operations personnel in leakage situations.

Gaseous helium coolant does not suffer from safety concerns associated with phase change. However, like most gases, helium has low atomic density and therefore low volumetric heat capacity. This dictates that high efficiency helium-based coolant systems operate with high flow rates, high system pressures, and a large coolant temperature differential. This requirement has a negative influence upon the tubing failure rate for the system since the high flow rate imparts additional stresses upon the tubing. However, helium's other benign properties as a coolant may compensate for the additional tubing load stresses imparted by the high flow rate. Reliability estimates for helium-cooled copper systems suggest an in-vessel LOCA frequency that compares very favorably with the one from a water-cooled system.²

Operating experience from helium-cooled fission reactors shows that helium does not produce any significant corrosion in its piping. This statement excludes the scenario where water enters the helium channel via leakage in the helium-to-water heat exchanger since this may result in significant corrosion problems. Pure helium (less than 0.01% impurities) should not create significant amounts of corrosion in the in-vessel tubing during ITER operation. However, if the helium becomes contaminated, then corrosion levels may increase. Experiments with high temperature, helium-cooled fission reactors show that corrosion problems due to helium impurities such as hydrogen and water vapor are not totally avoidable. Since these impurities may be oxidizing or reducing, depending upon the

nature of the contaminant, corrosive helium-metal reactions can occur in helium-cooled reactors.³

2.3 Alkali Metals

Nuclear reactors that have high power densities ($P_d \geq 200$ kW/l) require coolant types with exceptional heat-transfer capabilities. The liquid metals are ideal for such applications since they have a high thermal conductivity coefficient. Liquid metals, as a coolant type, allow elevated reactor operating temperatures without the need for high pressurization of the coolant system (system pressure for the Liquid Metal Fast Breeder Reactor (LMFBR) is slightly above atmospheric).⁴ The low system pressure of liquid metal coolant systems serves to decrease the tubing failure rate since the piping endures much less stress.

Liquid metal coolant types can potentially contribute two additional safety advantages to ITER: low LOCA depressurization effects and good natural circulation heat removal. Since they do not require a highly pressurized system to efficiently remove heat, liquid metal coolant systems will not experience a large pressure differential after a rupture occurs. This suggests that coolant will leave the system at a maximum rate equal to operating flow rates. Additionally, the tubing will not have the "whip" force of higher pressure systems so there is a lower probability of the ruptured tube lashing out and damaging either an adjacent tube or the vacuum vessel. The high thermal conductivity of liquid metals and their ability to remove heat at low pressure suggests a high heat removal rate from the reactor even when the coolant flow is due to natural circulation. This is quite significant since it implies a lower probability of hot spots in the tubing, which may exceed design limitations and cause the tubing to leak or rupture.

Although liquid metal coolants do offer the safety advantages discussed above, their total safety margin, from a tubing reliability perspective, is less than that of the other coolant types. The primary reason for this is liquid metal coolant incompatibility with most other metals. This incompatibility can be divided into two separate issues: degradation of the containment material's mechanical strength and mass transfer from the containment material.

Deterioration of the tubing's mechanical properties is caused by the microstructural and compositional changes that occur during the long-term, high-temperature exposure to the liquid metal. Liquid metals can influence the surface properties of the material through the mechanisms of embrittlement, oxidation, or carburization-decarburization phenomena. Each of these mechanisms can alter the near-surface deformation behavior of the material and thus affect mechanical properties such as fatigue crack propagation and creep ductility.

Erosion (mass transfer) of tubing material has been reported in nearly all experiments and applications that feature circulating liquid metal as the coolant type. The erosion effects of the liquid metal upon the tubing material can be uniform or selective dissolution, intergranular penetration, or interstitial-element transfer to and from the liquid metal. The corrosion effects can ultimately cause significant wall thinning and deposition (transport)

of corrosion products. Wall thinning implies a loss of tubing mechanical integrity and accordingly, a higher probability of failure. Corrosion product deposition implies the potential for coolant flow restriction, which may cause hot spots and areas of large pressure differential; both increase the probability of tubing failure.

The erosion problem varies in its degree of severity, depending upon tubing material-liquid metal combination and other variables. The more important of the variables include the liquid metal's purity and temperature; the tubing material's composition, microstructure, and exposure period; and coolant system operating parameters such as flow velocity, temperature differential, and temperature profile.

Liquid metal coolants may further affect tubing reliability by requiring complex maintenance procedures. As the liquid metal circulates through the blanket, it may capture neutrons and become radioactive. A radioactive coolant requires special handling to protect operations personnel and such special handling can conceivably affect maintenance procedures by limiting access to the tubing.

In this report, liquid metal coolants are represented by liquid lithium and liquid sodium. These two metals were chosen because of their suggested use at various periods of the EDA. The following subsections provide general comments on the two liquid metals.

2.3.1 Lithium

Lithium differs from other alkali metals because its chemical properties are more similar to those of alkaline-earth metals. Experiments show that liquid lithium is very aggressive in its attack on metals, more so than sodium or sodium-potassium alloys.⁵ Lithium's attack is greatly influenced by its contaminants so the severity or extent of lithium's attack may be governed to a certain extent.

2.3.2 Sodium

Liquid sodium has seen extensive use as a coolant type in various fission reactors and fusion technology experiments. The operating experiences from the reactors and experiments suggest that liquid sodium is very reactive and easily contaminated.³ Many of these contaminants enhance sodium's corrosive attack on the container material. The more corrosive of the contaminants are oxygen, carbon, nitrogen, and hydrogen. Oxygen increases the corrosion rate of attack, carbon and nitrogen cause excessive carburization, and hydrogen induces embrittlement. Collectively or independently, all four contaminants cause an increase in the tubing material's failure rate.

2.4 Water

Water coolant is defined in this report as the equivalent of light water. Of the coolant choices, water has the benefit of being used for cooling purposes in almost every industry. Thus, there is a wealth of information concerning water's performance as a coolant. Some

benefits of using water as a coolant type include the relative abundance, ease of handling and filtration, moderation effect upon neutrons, and compatibility with most tubing materials.

A major safety concern with using water as a coolant is voiding effects. Reactor operating experiences show that system temperatures above the Departure from Nucleate Boiling Range (DNBR) causes significant water voiding. Water voiding represents a loss of heat removal efficiency since the water vapor is coalescing into bubbles and the voids have a low heat conductivity coefficient. Coolant voiding causes localized hot spots in the tubing, causing it to lose its mechanical integrity and have a higher probability of failure.

In the temperature range for ITER operation, water coolant would experience a phase change, going from liquid to steam. However, water has higher thermal conductivity as a liquid so the coolant system must be pressurized to insure water remains in the liquid phase. As mentioned earlier, system pressurization has a negative influence upon tubing failure rates due to the additional stresses.

Water, in its purest form, is not especially corrosive or damaging to tubing material. Water does contain oxygen and therefore may be more corrosive with materials that are easily oxidized.

2.5 Summary

The in-vessel tubing failure rate is strongly influenced by the coolant type. The coolants considered in this report are helium gas, liquid lithium, liquid sodium, and light water. Each coolant has safety advantages and disadvantages, both for the coolant system and the tubing system.

Helium's advantages lie in its inertness and low corrosion levels. However, helium requires high flow rates for efficient heat removal and has poor natural circulation properties. The liquid metals have excellent thermal conductivity and natural circulation properties. They also require low system pressure, which implies good natural circulation characteristics. The liquid metals disadvantage lies in their incompatibility with most tubing materials. Light water has been used in many cooling applications and has an extensive operational database. Water is not particularly corrosive to most materials; proper chemical control can maintain minimal corrosion levels. Beyond the DNBR, water is susceptible to boiling and may create localized hot spots along the tubing.

Table 2-1 presents thermal conductivity coefficients for the coolants. The table shows that liquid sodium has the best thermal conductivity of the coolant types and water's thermal conductivity is approximately twice that of helium's. Table 2-2 presents general operating characteristics for the coolant types as used in fission reactors. The table shows that water coolant has the highest flow rate and system pressure of the coolant types, but has the lowest power density and efficiency. Helium coolant has the lowest power density, system pressure, and flow rate of the coolant types.

Table 2-1
Thermal Conductivity of Coolant Types⁴

Coolant	Temperature °K	Density kg/m ³	Thermal Conductivity W/m-K
Helium	700	0.0703	0.278
Sodium	700	850	70
Water	573	714	0.54

Table 2-2
Properties of Reactor Coolants⁶

Coolant	Thermal Output MWt	Efficiency %	Power Density kW/l	Pressure MPa	Flow Rate 10 ⁻⁶ kg/hr	Temperature	
						Inlet °C	Outlet °C
Helium HTGR	3000	39	8.4	5	5	337	755
Helium GCFR	2530	39.5	297	8.6	10	332	642
Sodium LMFBR	2410	39	380	1.4	50	380	552
Water PWR	3411	33.7	104	15.5	62	300	332

Chapter References

1. Petti, D., Piet, S., McCarthy, K., Cadwallader, L., Key Safety Issues Related to Coolant Selection for ITER, ITER/93/EN/SA-2, Idaho National Engineering Laboratory, February 1993.
2. Cadwallader, L. C., Investigation of Selected Accident Frequencies for Helium, Water, and Liquid Metal ITER Coolants, ITER/US/93/EN/SA-1, Idaho National Engineering Laboratory, January 1993.
3. Berry, Warren, Corrosion in Nuclear Applications, John Wiley and Sons, New York 1971.
4. Glasstone, S. and Sesonske, A., Nuclear Reactor Engineering, Third Edition, Krieger Publishing Company, Florida, 1991.
5. Kelman, LeRoy, Wilkinson, W., and Yaggee, F., Resistance of Materials to Attack by Liquid Metals, ANL-4417, Argonne National Laboratory, 1950.
6. Duderstadt J., and Hamilton, L., Nuclear Reactor Analysis, John Wiley, & Sons, New York, 1976.

3.0 IN-VESSEL TUBING FAILURE MECHANISMS

3.1 Introduction

A tubing system can have many failure mechanisms, the number heavily dependent on the operating environment. The term failure mechanism is defined in this report as any mechanism that increases the probability of tubing failure. Failure mechanisms include chemical, nuclear, and mechanical processes.

The operating environment of the tubing system is the most important variable when considering failure mechanisms since the environment can enhance the activity or attack of the failure mechanisms. For example, consider the operating environment for ITER and the failure mechanism creep. Creep is a function of temperature, i.e., it displays greater effects at higher temperatures, and ITER cooling systems are expected to operate in a high temperature regime ($400 \leq T \leq 600^\circ\text{C}$). Accordingly, creep is anticipated to be a significant failure mechanism.

The number of possible failure mechanisms for in-vessel tubing decreases for ITER considerations since many mechanisms will not occur, e.g., microbiological attack. Failure mechanisms that thrive in high temperature, cyclic, and vacuum environments are considered in this report. Mechanisms enhanced by the presence of hydrogen, helium, or oxygen, are also important since these three gases can cause substantial attack upon the tubing material's mechanical integrity.

Failure mechanisms for ITER in-vessel tubing are chosen from literature for fission Nuclear Power Steam Systems (NPSS), fossil fuel boiler plants, natural gas pipelines, and sea water distillation systems. The literature is reviewed to identify common piping failure mechanisms with ITER-specific failure mechanisms taken as a subset. The following sections present a brief discussion on the in-vessel tubing failure mechanisms anticipated for ITER. The failure rates, which are generated in a later chapter, will be based on the expected influence of the defined failure mechanisms upon the tubing system's mechanical integrity.

3.2 Creep

One of the more critical variables that decides the integrity of high temperature components is their creep behavior.¹ Generally creep implies the elongation of the component along an axis as induced by thermal loads and/or stresses. Creep can also be induced by irradiation of the material. For brevity, the authors define the term, creep, as the combination of thermally-induced and radiation-induced creep.

Materials under thermally-induced creep will slowly, continuously, and plastically deform until they eventually fail. Deformation of the material has been shown to be a function of time, temperature, and applied load (stress). Thermally-induced creep affects the physical dimensions of the material and therefore threatens the material's mechanical integrity.

Depending on the material, the resulting failure may be limited either by deformation or by fracture.

Radiation-induced creep is a common phenomenon in materials subject to neutron bombardment. Stresses are introduced in the materials as neutrons displace atoms from their location in the lattices and by the production of helium gas from (n, α) reactions. The stresses are relieved in the materials by the process of irradiation creep.² Radiation-induced creep has been shown to have a dependence on the material's degree of cold work and the irradiation temperature.³ That is, the phenomena occur less frequently and to a smaller extent with cold worked materials that operate in relatively low temperature (< 300 °C) environments.

3.3 Cyclic Fatigue

Repetitive or continually fluctuating loads (stresses) weaken the mechanical integrity of a material and will ultimately result in the material's failure. Material failures induced by cyclic loading are termed cyclic fatigue failures. Cyclic fatigue can cause materials to fail at stress levels much lower than those required for single load failure.

Cyclic fatigue can occur in any location where cyclic tensile stresses are operative. Systems used discontinuously or otherwise operated cyclically are especially susceptible to cyclic fatigue. Rapid startup or shutdown of a system can create vibrational stresses, alternating bending loads, and fluctuating thermal stresses. Therefore, startups and shutdowns also contribute to a system's susceptibility to cyclic fatigue.

There is a failure threshold for cyclic fatigue and this threshold is dependent upon the magnitude of the applied stress. Cyclic fatigue is also categorized by the number of cycles required for failure. Two types of cyclic fatigue are distinguished: High-Cycle Fatigue (HCF) and Low-Cycle Fatigue (LCF). HCF is traditionally classified as a failure occurring above 10^4 cycles while LCF is a failure occurring below that value.

An important distinction between HCF and LCF is the behavior of cracks in the material before failure occurs. In HCF, most of the fatigue life is spent in crack initiation, whereas in LCF most of the life is spent in crack propagation. For ITER in-vessel tubing, cyclic fatigue is expected to be mostly HCF. This assumption is based on ITER having a high frequency of pulse cycles and tubing materials fabricated to tolerate crack initiation, but prevent crack propagation.

3.4 Erosion

In this report, erosion failure is defined for water and helium coolants as the failure caused by impingement of coolant particles upon tubing walls as coolant circulates through the system. For liquid metal coolants, erosion failure is defined as the failure due to mass transfer from the tubing walls. For all four coolant types, the erosion failure mechanism promotes tubing failure by weakening the tube's internal integrity.

In water cooled systems, erosion by impingement attack is easily identified by visual inspection of the tubing. Impingement attack is characterized by a roughened internal surface of the tubing. The roughness and U-shaped pits in the tubing are invariably directional in nature and have undercutting in the direction of coolant flow. The attack proceeds to penetrate into the tubing and cause failure by leakage or rupture if conditions are sufficiently severe and remain unchanged.

The causes of impingement attack are high coolant velocity, the presence of entrained gases (including air that may have infiltrated the system), and entrained solids. Both gases and solids within the coolant increase the speed of attack and lower the threshold velocity required for attack. Gases trapped in the coolant may also enhance the erosion failure mechanism by influencing the corrosiveness of the coolant.

Some operating experience failure data suggest that erosion is an insignificant failure mechanism for water and steam cooled tubing systems.⁴ However, it is expected that this data is biased by the effects of extensive water chemistry control. Other cooling applications with water project that high temperatures and velocities will greatly increase the erosion process⁵. The authors believe that erosion will occur in water and helium coolant systems, the magnitude of erosion depends on tubing material characteristics and coolant purity.

In liquid metal coolant systems, erosion is generally related to the solubility of the tubing material with the coolant. Erosion in such system usually occurs as the result of thermodynamic imbalance at the interfaces between liquid and solid metal; the result being mass transfer. Mass transfer in liquid metal coolant systems is caused by the presence of gradients. The three gradients are: concentration, temperature, and energy.⁷ The authors consider the erosion failure mechanism for liquid metal coolant systems to be the culmination of all three mass transfer types.

The velocity and temperature of liquid metal coolants have been shown to be important variables for erosion by mass transfer. More specifically, a material's solution rate in liquid metal is velocity dependent. This dependency implies that the transfer of atoms from the solid metal into the immediately adjacent liquid metal is more rapid than their subsequent migration through the liquid metal. Experiments with flowing liquid metals where liquid diffusion controlled the solution rate, show that the corrosion rate varies as $(v)^{0.8}$ where v is the average flow velocity. As a statement to the general applicability of this observation, the velocity dependence has also been observed in systems that were not diffusion limited.

The other erosion controlling variable, temperature, is important because solubilities and diffusion rates generally increase with increasing temperature. Temperature gradient mass transfer has been noted to occur even when the solid material is only sparingly soluble in the liquid metal.

Mass transfer by temperature gradients has proven to be a considerable problem in circulating loop systems. In such systems, the material goes into solution in the hot zones

of the loop and precipitates out when its solubility limit is exceeded in the cold zone of the loop. This usually results in aberrant stresses in the coolant tubing and decreased heat removal capabilities. It is worthwhile to note that the subsequent tubing failure resulted more from deposited metal plugging in the cold zone than from wall reduction by material dissolution in the hot zone.⁶

3.5 Helium Swelling

Helium gas is produced in materials when incident neutrons react with the material to produce alpha particles via the (n,α) reaction. In most metals, this helium is thermodynamically insoluble and tends to congregate into bubbles. As the helium bubbles migrate through the material, they coalesce and create compressive and tensile stresses depending on the component geometry and initial loading. The helium bubbles may also link at the material's grain boundaries and cause intergranular failure. In either case, the presence of helium gas within the material increases the material's susceptibility to failure by reducing its mechanical integrity.

It can logically be argued that ITER designers will optimize tubing design parameters to reduce the effects of helium swelling. However, when one considers the number of neutrons that will bombard ITER's first wall and blanket area, helium production and subsequent induced swelling appear inevitable. Accordingly, the failure mechanism of helium swelling is included in this report.

3.6 Hydrogen Damage

Hydrogen damage is a general term that refers to material damage caused by the presence of or interaction with hydrogen. Hydrogen can exist as coolant impurities or be produced by (n,d) and (n,t) reactions, plasma driven implantation, or radiolytic decomposition.

Hydrogen damage can be classified into four distinct types:⁵

- a. Hydrogen blistering
- b. Hydrogen embrittlement
- c. Decarburization in steels
- d. Hydrogen attack

Hydrogen blistering results from the penetration of hydrogen into a material. When the depth of hydrogen penetration is limited to a few mils (25 or 50 μm), blistering of the material occurs as the hydrogen coalesces into bubbles. The result of the blistering is local deformation and in extreme cases, complete destruction of the tubing wall.

Hydrogen embrittlement is also caused by penetration of hydrogen into a material, but penetration to a greater depth. The failure mechanism of hydrogen embrittlement is not completely understood, however, the effects are frequently observed. Hydrogen embrittlement is generally observed at a slow strain rate and may be reversible or

irreversible depending upon the material.⁵ Embrittlement may be caused by: stress induced precipitation from solid solutions supersaturated with hydrogen; interactions of hydrogen atoms with moving dislocations; diffusion of hydrogen in uniform stress, thermal, or electrical fields; or hydrogen produced in stress corrosion cracking reactions.⁵

Decarburization is a phenomenon associated primarily with steels.⁵ Steels in a high temperature environment that contains moist hydrogen have been noted to experience the loss of carbon atoms. The loss of these atoms, decarburization, lowers the tensile strength of steels and make them susceptible to failure at loads lower than design specifications.

Hydrogen attack refers to the high temperature interaction between a hydrogen atom and one of the materials in an alloy. An example of hydrogen attack is the disintegration of oxygen-containing copper, Electrolytic Tough Pitch (ETP) copper, in the presence of hydrogen. The dissolved O_2 from the disintegrated copper forms H_2O with the H_2 and causes swelling.⁷

Hydrogen damage from radiolysis occurs primarily in water-cooled systems. The radiolysis generates free hydrogen that subsequently damages coolant tubing in a manner similar to hydrogen embrittlement. Damage usually occurs in regions of high heat flux, heat-transfer, welds, and other devices that disrupt the coolant's flow.

For failure rate analyses, the authors group the hydrogen damage types under the general term, hydrogen damage. As an example of hydrogen damage, consider the following case for a steel tubed boiler system.⁷

In boiler systems, concentrated sodium hydroxide typically dissolves the magnetic iron oxide on the steel tube. With the protective covering now destroyed, water can directly react with the iron in the tube to evolve atomic hydrogen. The sodium hydroxide itself may also react with the iron to produce hydrogen. If atomic hydrogen is liberated, it is capable of diffusing into the steel. Some of this diffused atomic hydrogen will combine at grain boundaries or inclusions in the metal to produce molecular hydrogen. Since molecular hydrogen is not capable of diffusing through the steel, the gas accumulates, primarily at grain boundaries. Eventually, gas pressures will cause separation of the metal at its grain boundaries, producing discontinuous intergranular micro-cracks. As micro-cracks accumulate, tube strength diminishes until the stress imposed by boiler pressure exceeds the tensile strength of the remaining, intact metal. At this point a longitudinal burst may occur.

3.7 Irradiation Embrittlement

In the study of irradiation effects in metals, the single most important variable, besides the neutron flux, is temperature.⁸ This is because temperature determines whether the defects caused by irradiation will have the mobility necessary for interactions and the energy required to produce additional secondary effects. In this regard, the melting point of the

material becomes important since it has been shown that the temperature must be above half the absolute melting point (homologous temperature = T/T_m) for many solid-state reactions to occur.

The thermal and mechanical history of the material can also influence its post-irradiation behavior. For example, cold worked materials have shown less susceptibility to radiation-induced damage than non-cold worked materials.⁸ The difference is expected to be a result of the considerable concentration of dislocations introduced in cold worked specimens. The large concentration of dislocations is believed to mask the effect of radiation-induced defects.⁸

Currently there is no universally accepted procedure for predicting radiation effects upon material hardness. However, it is known that the lower the melting point of a material, the lower the experiment/operation temperature must be for an observable radiation effect, i.e., as T_{mp} decreases so does $T_{\text{observed effects}}$. This suggests that materials with high melting points will experience fewer radiation hardening effects.⁸

3.8 Magnetic Field Interactions

At the time of this report, the question arose as to possible influences of ITER's magnetic fields upon in-vessel tubing failure rates. This section answers the question using the analysis results of Attaya.⁹ Attaya analyzed the magnetic fields induced in ferromagnetic steels and the resultant forces. The analysis considered the magnetic loading on a segment of coolant piping for the Mirror Advanced Reactor Study (MARS). The conclusions were: 1) the magnitude of the resultant magnetic force is small in comparison with the coolant pressure and 2) the distribution of the magnetic force is not uniform. The non-uniformity of the force is expected to give the once circular piping an ultimate oval cross section as the pipe is deformed by thermally-induced creep.

Since the phenomenon has been identified only for ferromagnetic materials, this failure mechanism is not expected to be significant for the other tubing materials. The failure mechanism is included in the analyses of this report, but in a limited role. Additional research is necessary to reach a better understanding of this phenomenon.

3.9 Oxidation

Oxidation would not generally be considered as a failure mechanism for ITER due to the machine's vacuum environment. However, maintenance practices may require the vacuum vessel to be vented or oxygen may exist within the machine as a coolant impurity. Oxygen can also exist as the by-product of interactions between the tubing material and the liquid metal coolant. Oxidation affects tubing by causing scaling, spalling, and embrittlement.

3.10 Intergranular Attack

The authors define intergranular attack as the combination of two indirectly related failure

mechanisms: stress-corrosion cracking and intergranular corrosion. Stress-Corrosion Cracking (SCC) is included because it has been observed in most materials, regardless of coolant type. Intergranular corrosion is included because it is a significant failure mechanism for alloyed materials. Both mechanisms promote failure by attacking the grain boundaries of the tubing material. The two mechanisms are further discussed in the following subsections.

3.10.1 Intergranular Corrosion

Intergranular corrosion is generally a result of atomic incompatibilities at the grain boundaries of polycrystalline metals. The incompatibilities cause the stored strain energy of the metal to be higher in the grain boundary regions than in the grains. Since these high-energy grain boundaries are more chemically reactive than the grains, the boundaries are susceptible to corrosion attack.

Intergranular corrosion requires an environment that has high temperatures and stresses. When these two conditions are present, intergranular corrosion occurs by depleting one of the alloying element within the material. This depletion occurs at the grain boundaries and can locally change the composition of a normally corrosion resistant material, making it susceptible to corrosive attack.

3.10.2 Stress-Corrosion Cracking

Stress corrosion cracking (SCC) is an extremely important failure mode because it occurs in many materials. SCC is observed mainly in highly stressed areas such as expansion transition zones.¹⁰ In principle, SCC can occur wherever a corrosion agent and sufficient tensile stresses coexist. SCC is not a surface attack, instead it propagates a system of intergranular or transgranular cracks through the material's matrix over a period of time.¹¹

The chemical compositions of environments conducive to SCC are highly specific to the material alloy and general patterns have not been observed. SCC has been observed to be influenced by alloy composition, environment type, temperature, and material stress level. The stress levels that produce SCC are normally well below the yield strength of the material, and residual stresses as well as applied stresses may produce failure. The lower the stress level, the longer the time required to produce cracking. There are indications of a threshold stress level, below which SCC will not occur. However, this threshold level is not always determinable.

SCC occurs mostly in water cooled tubing so this discussion is primary for water coolant. The term SCC refers to material failure resulting from a synergistic interaction of tensile stress and a specific corrodant. The tensile stresses may be either applied, such as those caused by internal pressure, or residual, such as those induced by welding. Gross attack of the metal is not necessary for this phenomenon and, in fact, does not characteristically accompany it.

In water coolant systems, there are two other factors to the SCC process: the ability of sodium hydroxide or alkaline-producing salts (i.e., salts whose solution in water produces a base) to attack the material and a mechanism for concentration. Sodium hydroxide and alkaline-producing salts are rarely present at corrosion levels in the system environment. Therefore, a means of concentrating them must be present. The three basic concentration mechanisms are:

- ◆ **Departure from Nucleate Boiling (DNB).**⁷ The term nucleate boiling refers to a condition in which discrete bubbles of steam nucleate at points on a metal surface. Normally, as these steam bubbles form, minute concentrations of water solids will develop at the metal surface, usually at the interface of the bubble and the water. As the bubble separates from the metal surface, the water will redissolve soluble solids such as sodium hydroxide. At the onset of DNB, the rate of bubble formation exceeds the rinsing rate. Under these conditions, sodium hydroxide, as well as other dissolved solids or suspended solids, will begin to concentrate. The presence of concentrated sodium hydroxide and other concentrated corrosives will compromise the thin film of magnetic iron oxide, causing metal loss. Under the conditions of fully developed DNB, a stable film or blanket of steam will form. Corrosives then concentrate at the edges of this blanket, causing metal loss at the perimeter. The metal at the interior of the blanket is left relatively intact.
- ◆ **Deposition.** A situation similar to DNB occurs when deposits shield the metal from the bulk water. Steam that forms under these thermally insulating deposits escapes and leaves a corrosive residue that can deeply gouge the metal surface.
- ◆ **Evaporation at water line.** Where a water line exists, corrosives may concentrate by evaporation, resulting in gouging along the water line.

The above discussion concentrates on SCC as it applies to water coolant and for conditions where coolant phase change is anticipated. While not completely representative of helium and liquid metal coolants or directly applicable ITER, the discussion is included for thoroughness. For example, one would not consider designing a nucleate boiling operating point for ITER since this could easily lead to a serious accident scenario. However, consideration of evaporation at a water line is important for a water-cooled design since the water level may decrease during maintenance or off-line procedures. If the cooling system is kept "solid"; that is, filled with liquid water, then oxidation and corrosion are inhibited. If the coolant is kept flowing, even at low flow rates, oxidation and corrosion are greatly reduced.

Stress-Relief Cracking (SRC) refers to the formation of intergranular cracks in the coarse-grained regions of heat-affected zones. SRC generally occurs in the weld metal of a welded assembly when it is reheated to relieve residual stresses or placed into service at elevated temperatures. SRC accompanies the relaxation of residual stresses by creep and is a manifestation of poor creep ductility. SRC usually occurs in steels alloyed with strong carbide-formers to increase the steel's high temperature strength.⁷

3.11 Thermal Cycling

Startup and shutdown of high temperature systems create transient temperature gradients. As these transients are repeated, the differential thermal expansion during each transient creates a thermally induced cyclic stress in the systems. Fatigue damage by thermal cycling depends on the nature and frequency of the transient and the thermal gradient of the system. Systems subjected to thermally induced stresses generally operate within their creep range. Thus, damage due to both fatigue and creep have to be considered.

Fast system startups from low temperatures can create severe thermal stresses. The effects of start-stop cycles can be illustrated by considering the high pressure rotor of a steam turbine.¹ During start-up, the surface of the rotor heats up faster than the bulk and attempts to expand. However, surface expansion rates are limited by bulk expansion rates. This imbalance between the rotor's surface and bulk expansion rates creates a surface compressive strain. The compressive strain eventually relaxes to zero strain, but it leaves a tensile residual stress when the entire rotor reaches steady operating temperatures. Upon shutdown, the process is reversed. Efforts are taken to minimize the stress by slow heat up, using a turning gear to give initial momentum to the rotor at start-up, and etc. to prolong turbine life.

3.12 Other Mechanisms

This section discusses tubing failure mechanisms that may be possible, but are excluded from the failure analyses. Reasons for exclusion include lack of appropriate data and anticipated design optimizations by the ITER engineers.

3.12.1 Electric Field Interactions

At the time of this report, the question arose concerning possible influences of ITER's electrical fields upon in-vessel tubing failure rates. The authors had the objective of addressing this issue, however, literature was not found on the subject. Further research and testing is necessary for better understanding.

3.12.2 Geometry

The geometry (configuration) of a tubing system can significantly influence the system's failure rate. A tubing system that features spiraling or other obtuse geometrical turns of the tubing will have a higher failure rate since such complex geometries introduce additional stresses. Complex piping geometries will also have to contend with irradiation and thermally-induced stresses. These stresses will cause the tubing to swell or elongate and may produce external damage to the tubing as it attempts to move against the restraints of its mounting and positioning fixtures.

The failure analyses of the authors do not consider tubing system geometry effects. The exclusion is based on the assumption that ITER design engineers will optimize tubing

fixture designs to allow controlled tubing movement.

3.12.3 Runaway Electrons

The issue of runaway electrons is a viable tubing concern since the phenomenon can occur in tokamak reactors. Runaway electrons adversely affect tubing reliability through the action of the electrons. These high energy electrons can potentially tunnel through the tubing wall and cause a coolant leak. This leak may result in a LOCA and/or plasma disruption.

At the time of this report, there were no published operating experience data on the effects of runaway electrons upon tubing reliability performance. However, the authors believe the accident to be a plausible scenario unless the ITER design team specifically designs against it. Without such design precautions, runaway electrons may significantly increase the tubing failure rates generated in this report. In our failure rate analyses we, the authors, do not consider the effects of runaway electrons upon tubing reliability, however further research and testing is required for a better understanding of the phenomenon.

3.13 Summary

The tubing failure mechanisms defined in this chapter form the basis for the generation of ITER in-vessel tubing failure rates. The ten most critical failure mechanisms are: creep, cyclic fatigue, erosion, helium-induced swelling, hydrogen embrittlement, intergranular attack, irradiation embrittlement, magnetic field interactions, oxidation, and thermal cycling.

For the analyses by the authors, the failure mechanisms are considered to be independent, that is, each mechanism operates without the influence of other failure mechanisms. In reactor applications, this assumption will not necessarily be true. For example, the failure mechanism of irradiation has been shown to influence the failure mechanism of creep.

The assumption of failure mechanism independence allows the mechanism to directly influence the tubing failure rate. Since tubing operating experience predicts that the identified failure mechanisms will exist for ITER, allowing the mechanisms to influence in-vessel tubing failure rates helps tailor the rates for ITER. This assumption also allows the failure analyses of this report to be readily upgraded as additional material failure data becomes available.

4.13 Chapter References

1. Viswanathan, R., Damage Mechanisms and Life Assessment of High-Temperature Components, ASM International, Ohio, 1989.
2. Grossbeck, M. L. and Horak, J. A., "Irradiation Creep in Type 316 Stainless Steel and U.S. PCA With Fusion Reactor He/dpa Levels," Journal of Nuclear Materials, pgs 155-177, 1988.
3. Olander, Donald R., Fundamental Aspects of Nuclear Reactor Fuel Elements, Technical Information Center, Energy Research and Development Administration, U.S. Department of Energy, 1976.
4. Port, Robert D., The Nalco Guide to Boiler Failure Analysis, McGraw-Hill, New York, 1991.
5. Fontana, Mars G., Corrosion Engineering, 3rd Edition, McGraw-Hill, New York, 1986.
6. Gurinsky, D. H. and et al., "Corrosion in Liquid Metal Systems," Third International Conference on the Peaceful Uses of Atomic Energy, Vol 9, Reactor Materials, 1965.
7. Berry, Warren, Corrosion in Nuclear Applications, John Wiley and Sons, NY, 1971.
8. Kircher, John F. and Bowman, Richard E., Effects of Radiation on Materials and Components, New York: Reinhold Publishing Corporation, 1964.
9. Attaya, H., Kulcinski, G. L., and Wolfer, W. G., "Analysis of Forces on Ferromagnetic Components Used in Magnetic fusion Reactors," Journal of Nuclear Materials, 122 & 123, 1984.
10. Mavko, Borut and Cizelj, Leon, "Failure Probability of Axially Cracked Steam Generator Tubes", Nuclear Technology, Vol. 98, May 1992.
11. Collins, J. A., Failure of Materials in Mechanical Design, New York: John Wiley & Sons, Inc., 1981.

4.0 MATERIAL ANALYSIS

4.1 Introduction

Tubing materials for ITER must exhibit superior performance in the areas of thermal conductivity and compatibility with various coolants. Additionally, the materials should have high levels of resistance to the anticipated failure mechanisms: creep, cyclic fatigue, erosion, helium swelling, hydrogen and irradiation embrittlement, and oxidation. This chapter discusses the effects of the failure mechanisms upon the proposed tubing materials. The materials considered are: beryllium, copper, Inconel 600, niobium, 316 stainless steel, titanium, vanadium, and zirconium.

The failure nomenclature of this chapter may lead to confusion since all of the terms are prefaced with the word failure. The following discussion is intended to eliminate confusion between the terms.

The authors use the terms failure mechanisms, failure data, operating experience failure rate, and ITER tubing failure rate. Failure mechanisms are chemical or nuclear processes that attack the tubing material and reduce its mechanical integrity. Failure data are information from commercial plants or laboratory experiments that report how a material responds when exposed to a particular failure mechanism. An operating experience failure rate is the rate of failure for a material as determined by its performance in industry or laboratory applications. ITER tubing failure rates are material failure rates generated using operational experience failure rates.

The failure data presented in this chapter is from a variety of sources. These sources include material testing for fusion applications and material performance in commercial fission power plants. The different failure data sources preclude presenting the information in a format that allows direct comparison of the materials' resistance to the failure mechanisms. Analysis of the materials is performed by presenting published results on material failure resistance properties and summarizing these in a table provided in the conclusion section of this chapter.

Each section in this chapter discusses one of the proposed tubing materials. The sections provide general comments on the material and then specific information on resistance to failure mechanisms and behavior characteristics in the suggested coolant types. Since the failure data presented in this report are dependent upon published failure data, there may be instances where information is not available for a particular failure mechanism. In such cases, general comments are provided for the failure mechanism.

4.2 Beryllium

There are very few reported data for operating experiences with beryllium tubing. Beryllium tubing has been manufactured by Brush Wellman Incorporated,¹ however, Brush Wellman was unable to provide any failure data. In this report, failure data for beryllium tubing is

judged with support from operating experiences with beryllium neutron shields in experimental reactors. These data should be applicable until actual failure data for beryllium tubing becomes available.

4.2.1 Failure Mechanism Resistance.

The introduction of helium or any gas atom into the matrix of beryllium has been seen to cause swelling of the bulk material.² Neutron reflectors composed of beryllium and used in nuclear reactors have required replacement in times as short as six months simply as a consequence of radiation-induced and helium swelling³.

To better understand the effects of helium swelling on irradiated beryllium, Rich⁴ studied a block of material cut from the beryllium reflector of the Materials Testing Reactor (MTR). The reflector had been exposed to a fluence of 7.6×10^{21} n/cm² ($E_n > 1$ MeV) at a temperature of 70 °C. Rich reported the following:

- 1.) Considerable embrittlement of the beryllium block due to atomic displacements by fast neutrons.
- 2.) Helium produced by fast neutrons remains in solution at temperatures below 500 °C.
- 3.) Helium precipitates at temperatures above 500 °C and forms bubbles. These bubbles cause an increase in material's volume that will rarely exceed 30 percent.
- 4.) The presence of the bubbles at grain boundaries will cause the beryllium to embrittle and to become permeable to gases.

These results suggest that beryllium has limited resistance to the failure mechanisms of helium swelling and irradiation embrittlement

Beryllium has also shown limited resistance to the failure mechanism of oxidation. The oxidation that normally forms on beryllium is BeO. The kinetics of this oxidation have been studied by several researchers.^{5,6,7} At temperatures between 350 to 950 °C and for short exposure times, the oxidation kinetics are seen to follow a parabolic rate law. Longer exposure times at higher temperatures produce oxidation kinetics that have a more linear rate law.

The oxidation kinetics are dependent upon beryllium's activation energy at the specific temperature. Activation energy is the chemical energy required to begin the oxidation process and is usually given in units of kcal/mole. Table 4-1 presents some noted reaction rates (weight gains) for beryllium.⁶

Table 4-1
Reaction Rates for High Temperature Beryllium

Temperature °C	Activation Energy kcal/mole	Reaction Rate g/m ² -min
700*	n/a	0.62 to 8
930	110	0.55
1050	110	72.0
1065	10.6	350
1295	10.6	550

*The data for beryllium at 700 °C is from Antil.⁹ Antil does not provide the associated activation energy and his reaction rates (in units of g/m²) depend on the purity of the beryllium.

4.2.2 Helium Coolant

Beryllium's compatibility with helium has been studied by Cobbs.¹⁰ Cobbs studied high purity beryllium and did not notice any incompatibility issues up to temperatures of 600 °C. Published experiences with beryllium in impure helium were not found. The authors assume that the maximum rate of beryllium attack in impure helium is equivalent to exposure to each impurity individually.

4.2.3 Liquid Metal Coolant

In liquid sodium, beryllium reduces to a sodium oxide. The chemical reaction is:

The beryllium oxide that forms from the reaction, BeO, is neither adherent nor protective. Thus, the oxide is free to migrate through the liquid sodium. Berry¹¹ reports that gettering the liquid sodium with calcium or thorium reduces the oxygen content of the sodium and therefore the corrosion rate of beryllium.

Stang¹² has also studied beryllium behavior in liquid sodium. Stang performed stress-rupture tests for beryllium in static liquid sodium at 538 °C. His results suggest that intergranular attack is not a significant factor for beryllium in liquid sodium.

Beryllium's behavior in liquid lithium has been studied by Kelman¹³. Kelman placed beryllium samples in a liquid lithium bath at 600 °C for six days. His results indicate that beryllium is not significantly attacked by liquid lithium below temperatures of 600 °C.

4.2.4 Water Coolant

In the presence of water, beryllium develops a hydrated form of BeO , $\text{Be}(\text{OH})_2$. If the water is maintained with a high level of purity and heavy metal ion contamination is avoided, the $\text{Be}(\text{OH})_2$ is inactive and beryllium exhibits acceptable corrosion behavior.¹¹ The purity of the water can be maintained by demineralizing the water and adding HNO_3 to maintain pH levels between 5.5 and 6.5.¹⁴

Peoples¹⁵ reports that beryllium's corrosion behavior in high-temperature ($T > 250^\circ\text{C}$) water is erratic. Peoples studied vacuum-cast, extruded, and sintered beryllium samples. The samples were placed in tanks of demineralized water at temperatures of 274°C and 315°C for two weeks. The samples tested at 274°C exhibited corrosion rates of 7.9 to $26.1\ \mu\text{m}/\text{yr}$ with pit penetration rates of 25.4 to $4394.2\ \mu\text{m}/\text{yr}$. The samples tested at 315°C initially exhibited similar varying degrees of localized attack, but then began to partially disintegrate after 38 hours of exposure.

4.3 Copper

Copper differs from the other proposed tubing materials in several ways. First, there is a large operational database on copper tubing so much is known concerning its advantages and disadvantages. Second, copper satisfies many of the specifications for successful tubing: high thermal conductivity coefficient, formability and machinability, and sufficient strength levels. However, the same factors that make copper a good tubing choice also make it a poor tubing choice. Copper's high thermal conductivity makes it susceptible to creep. Its ease in formability and machinability indicate that it is a soft metal and therefore does not possess high strength levels at elevated temperatures.

4.3.1 Failure Mechanism Resistance

As a consequence of its ductility, copper is believed to be especially susceptible to damage from erosion. The material's ductility may also make it more vulnerable to the failure mechanisms of creep and irradiation embrittlement. The high thermal conductivity of copper suggests that thermal cycling fatigue may be significant, however, the material's ductility should provide better than average performance under regular cyclic loads.

4.3.2 Helium Coolant

No information is available at the time of this report concerning copper's behavior in a helium-cooled environment. Aside from the high system pressures, the authors do not believe that there are any special problems with using copper tubing with helium coolant. Thus, copper tubing is expected to have satisfactorily reliability performance in a helium-cooled environment. Additional material testing and research can be used to validate this assumption.

4.3.3 Liquid Metal Coolant

Generally, copper is not selected as a material for use in a liquid metal environment since unalloyed copper is resistant only in the range of 300 to 400 °C.¹¹ Laboratory experimentations by Eichelberger^{16,17} and Singer¹⁸ show copper to be relatively soluble in sodium. Additionally, many of the common copper alloys were found to be preferentially attacked in sodium.

Osborg¹⁹ has studied copper behavior in liquid lithium. He placed copper specimens in static liquid lithium at 300 °C for six days. The copper was reported to have dissolved in the lithium without compound formation.

4.3.4 Water Coolant

Oxygen has a dual role in the corrosion of copper. In nearly pure aerated water, a thin, protective layer of cuprous oxide and cupric hydroxide forms. However, high water velocities can disturb the corrosion product layer and allow copper attack. Attack can also be caused by oxygen as it interacts with the copper atoms in the tubing wall.

4.4 Nickel-Based Alloys (Inconel 600)

There are a variety of nickel alloys that may be suggested as tubing material. For instance, there is Type 316, Inconel X-750, Hastelloy, and Monel. The authors reviewed published literature and discovered that the nickel alloy most frequently used in fusion technology experiments was Inconel 650. Although Inconel 650 was more often used, better and more consistent failure data were found for Inconel 600. To use this data, the authors assume that Inconel 600's performance as in-vessel tubing will be indicative of the Inconel 600 alloy series. Inconel 600 is a nickel based alloy that includes chromium (15Cr-7Fe-bal Ni) for better strength properties.

5.4.1 Failure Mechanism Resistance

Unalloyed nickel is relatively resistant to oxidation effects up to temperatures of 600 °C.¹¹ Nickel oxidation kinetics follow a parabolic rate law up to 1000 °C and at partial pressures of one atmosphere (atm).^{20,21} Chromium additions greater than 14 weight percent produce a marked improvement in the nickel's resistance to oxidation.¹¹

Studies of Inconel 600 oxidation resistance have been performed by Baker.²² Baker reports that Inconel 600 exposed to 2 MPa air at 954 °C for 10,000 hours experiences oxide penetrations of 150 µm. At a higher temperature of 1000 °C and a 9400 hour exposure, the oxide penetration of Inconel 600 is 280 µm. Baker also provides an oxide penetration rate for Inconel 600 at lower temperatures. At 704 °C and a 5000 hour exposure, the oxide penetration rate of Inconel 600 is 1.3 µm/yr.

4.4.2 Helium Coolant

Inconel's performance in a helium coolant system is taken from information provided by Baker.²² Baker reports that Inconel 600 exposed for 1500 hours to helium gas with a temperature of 760 °C and contaminants of 2500 parts per million (ppm) CO and 2500 ppm H₂ experienced a weight gain of 5.5 g/m².

Bokros²³ has studied the effects of helium impurities upon Inconel's resistance to intergranular attack. His results state that unalloyed nickel experiences grain boundary attack at temperatures greater than 498 °C. Inconel 600 was studied and seen to exhibit internal oxidation at temperatures above 704 °C.

Neutron irradiation of nickel-alloys in impure helium has been studied by Bokros²³ and Wunderlich.²⁴ Fluxes of 2×10^{13} n/cm²-s ($E_n > 1$ MeV) and 1×10^{13} n/cm²-s ($E_n < 0.04$ eV) did not produce any significant attack on the alloys maintained at 510 °C. At 816 °C, the nickel alloys were slightly attacked (approximately 64 µm of grain boundary penetration after 243 hr of exposure).¹¹ The irradiated specimens also displayed more weight gains than unirradiated specimens. This weight gain is believed to result from the mass transport of carbon in the gas coolant.²⁵

4.4.3 Liquid Metal Coolant

Nickel-based alloys have good high-temperature strength levels and are thus favored for liquid sodium applications that use temperatures above 650 °C. However, this temperature range is where the corrosion of nickel-base alloys becomes accelerated. Corrosion rates of nickel-base alloys in liquid sodium are reputed to significantly increase for temperatures above 700 °C.²⁶

Corrosion of Inconel in liquid sodium is very much a velocity dependent phenomenon.^{27,28} Aside from the sodium's velocity, its oxidation concentration also influences the corrosion attack of Inconel. Liquid sodium with oxygen concentrations between 0 and 500 ppm produces minimal corrosion attack of Inconel 600.²⁹ However, oxygen concentrations above 1500 ppm produce corrosion rates that are approximately twice those at 500 ppm oxygen.³⁰

Stang¹² reports that nickel-base alloys carburize in liquid sodium. Inconel 600 was seen to decarburized in high oxygen sodium, however, not in high-purity (low oxygen content) sodium.³¹ Some loss of strength occurred as the result of this decarburization in Inconel 600, however, the loss was not considered to significantly reduce the material's mechanical integrity.¹¹

In his study of irradiated Inconel 600 specimens in liquid sodium, Miller³² concluded that irradiation has little effect upon the corrosion attack on the material. Miller's work was based on Inconel 600 specimens exposed to liquid sodium at 816 °C and neutron doses

of 4×10^{17} n/cm² ($E_n < 0.04$ eV and $E_n > 1$ MeV) for 200 hours. After the exposure, the specimens exhibited only etching and slight intergranular attack to a depth less than 13 μ m.

Miller's work has been confirmed by separate studies by ANL on the Experimental Breeder Reactor II (EBR-II). Operating experience with EBR-II exposed Inconel 600 to a fluence of 7.2×10^{20} n/cm² ($E_n > 1$ MeV) in flowing liquid sodium between 510 and 530 °C. The material was examined after exposure and did not exhibit a correlation between irradiation exposure and corrosion behavior.

Nickel-base alloys generally suffer extreme solution attack in liquid lithium since nickel is highly soluble in lithium. While they do not believe Inconel 600 will suffer such a severe attack, Freed³³ and Hoffman³⁴ suggest that Inconel 600 is susceptible to gross mass transfer effects when exposed to liquid lithium.

Wilkinson³⁵ provides better detailed information on Inconel 600 and liquid lithium combinations. His studies placed Inconel 600 in a liquid lithium bath at 300 °C for six days. After the exposure, the Inconel did not show indications of intergranular attack or significant changes in weight or dimensions. The tests were repeated at a liquid lithium temperature of 600 °C and the Inconel did show signs of solubility (2.1% by weight), intergranular penetrations and small increases in dimensions.

4.4.4 Water Coolant

Inconel 600 has successfully replaced LWR heat exchanger tubing (austenitic stainless steel) that suffered from intergranular attack amplified by the chloride levels of the water. Laboratory studies of corrosion rates for Inconel 600 in static water at 316 °C show that the material has a corrosion rate well below 1 g/m²-month.³⁶ The corrosion rate of Inconel 600 is dependent upon surface preparation. Lowest corrosion rates are observed with material that has been bright annealed after machining.¹¹

Intergranular attack of Inconel 600 in water has been studied by Boujant,³⁷ Coriou,³⁸ and Copson.³⁹ Boujant initiated and propagated intergranular attack in Inconel 600 through simulated BWR environments. The results of his studies showed that the intergranular attack of Inconel 600 was primarily caused by oxidizing species from radiolytic decomposition. He added NH₃ (concentration greater than 10 ppm) to the water phase of the simulation and noted that intergranular attack was significantly reduced.

Coriou studied Inconel 600 specimens placed in high purity water at temperatures between 350 and 650 °C. He noted that small amounts of dissolved oxygen in the water, together with crevices in the stressed areas, caused intergranular attack of the material. Copson performed similar tests, however, he was able to define a critical oxygen concentration for intergranular attack. This critical concentration was reported as 1 ppm.

4.5 Niobium

Niobium is often suggested as a fusion material because it has high strength. Niobium does produce Nb-62, Nb-90, Nb-92, and Nb-94m when irradiated and all of these are waste management concerns.

4.5.1 Failure Mechanism Resistance

Makin and Minter⁴⁰ have studied the influence of irradiation upon niobium's mechanical properties. The researchers considered niobium irradiated to a fluence of 1×10^{20} n/cm² ($E_n > 1$ MeV) at a temperature of 16 °C. The post irradiated niobium was seen to have an increase of 111 MPa in yield strength (from 420 to 531 MPa). In additional analyses, the researchers noted that recovery of irradiation hardening occurs at annealing temperatures of 350 °C. After an hour at 600 °C, the annealing process was complete. The effects of the annealing process were a very large reduction in uniform elongation, from 17.0 percent in the unirradiated condition to approximately 6.0 percent after irradiation.

High cycle fatigue behavior of niobium at room temperature has been studied by Ferro, Mazetti, and Montalenti.⁴¹ The conclusion of their study was that the fatigue limit (approximately 226 MPa) is reached before 10^7 cycles and the ratio of the fatigue limit to Ultimate Tensile Strength (UTS) is 0.5 to 0.8.

The effects of hydrogen on the HCF behavior of room temperature niobium have been investigated by Chung and Stoloff.⁴² Improvement in fatigue life of un-notched niobium was reported with increasing hydrogen content. However, the high cycle fatigue life was degraded by hydrogen if a notch were present. Chung also reported that crack propagation was very rapid in hydrogenated niobium at room temperature. HCF life of Nb-H alloys is reported to sharply decrease when the load frequency is decreased.

Doner⁴³ has done limited LCF testing on niobium and its alloys. Doner's research shows niobium displaying a two-stage hardening at low strain amplitudes. The role of hardening in the second stage increases with increasing strain amplitude, and a cyclic stress-strain curve independent of prior history was found to exist.

LCF work by Chung and Stoloff⁴² suggests that hydrogen additions have a detrimental effect on the LCF life of niobium in air. Cyclic hardening was observed in the first few test cycles and was followed by small amounts of softening after saturation. The observation was true for both unalloyed niobium and niobium-hydrogen alloys with a maximum hydrogen content of 1180 ppm.

Research by Leadbetter⁴⁴ shows that oxygen interstitial serve to improve the yield stress of niobium. The effect of oxygen on the fatigue life of niobium has been researched by Enrietto, Sinclair, and Wert.⁴⁵ The studies found that niobium is extremely susceptible to strain aging, both in tension and in fatigue, when small amounts of oxygen are present in solid solution. Maximum strengthening occurred at 400 °C in fatigue and at 500 °C in

tension. A rapid increase in the rate of strain aging was also reported with increasing oxygen content.

Similarly, research by Donoso⁴⁶ on slow strain-rate embrittlement of niobium by oxygen at elevated temperatures shows a ductile-to-brittle-to-ductile fracture transition at a critical oxygen content of 0.5 atomic percent. A ductility reduction with decreasing strain-rate and increasing oxygen concentration was also reported.

Creep properties of niobium have been studied by Ransow and Argent.⁴⁷ The researchers studied the effects of oxygen and carbon on the creep strength of niobium between 800 and 1200 °C. Their conclusions were that oxygen has little influence on the creep strength above 1050 °C, but at lower temperatures it reduces the creep rates.

Barnore⁴⁸ has studied the effects of helium on the tensile and creep behavior of niobium. He reports that as the helium content in niobium increases, the material strength will also increase for the temperature range of 25 and 800 °C. Barnore further reported that steady state creep rates progressively decrease with increasing helium concentrations.

4.5.2 Helium Coolant

Unalloyed niobium and niobium alloys with zirconium (< 5%) and titanium (< 10%) have been observed by Shoemaker⁴⁹ to carburize in impure helium. The helium was contaminated with CO at 1 atm and had temperatures between 400 and 550 °C. Weight gains from the carburization were reported to be low, however, the niobium reacted with the CO to form a surface carbide layer and the nearby oxygen dissolved in the material causing embrittling effects. Similar behavior was observed with niobium in helium containing 5×10^{-3} atm CO and 5×10^{-3} atm H₂.

Irradiation effects upon niobium's failure resistance have been studied by Shoemaker⁴⁹ and Kosiba.⁵⁰ The researchers exposed Nb-1Zr specimens to a flux of 2×10^{13} n/cm²-s ($E_n < 0.04$ eV) for 243 hours in impure helium and gettered helium; both were maintained at 816°C. The study showed the niobium alloy to have higher hardness levels after exposure to impure helium than for exposure to gettered helium.

4.5.3 Liquid Metal Coolant

Berry¹¹ has determined that the corrosion behavior of niobium in liquid sodium depends on the oxygen content in each of the two materials. The oxygen partitions between the niobium and the liquid sodium in a more-or-less fixed ratio at high temperatures. In laboratory experiments with the liquid sodium at 600 °C, niobium containing 70 ppm oxygen did not pickup oxygen until the oxygen content of the sodium exceeded 1000 ppm. Experiments with niobium initially containing 950 ppm oxygen showed the niobium losing oxygen to the liquid sodium, which initially contained 50 ppm oxygen.

In a similar analysis, Hoffman⁵¹ reported that niobium's corrosion resistance in liquid sodium can range from excellent to poor, dependent upon the oxygen concentration in the liquid sodium. The excellent behavior was observed with pure niobium in static sodium at 982 °C. The poor performance was for niobium in flowing sodium at 600 °C with 30 ppm oxygen (penetration rate of 10.1 mm/year).

Niobium alloys containing zirconium or titanium have been seen to be quite resistant to liquid sodium at temperatures between 1000 and 1315 °C.¹¹ Static and loop tests in liquid sodium at temperatures of 1090 to 1266 °C show that Nb-1Zr has good corrosion resistance properties. The Nb-1Zr alloy did exhibit slight solubility when exposed to the liquid sodium at the given temperatures, however, the alloy's overall performance was termed "excellent".^{52,53}

Cleary⁵⁴ reports that niobium is only slightly soluble in liquid lithium, however, he does not provide the temperature range or testing conditions. DiStefano⁵⁵ states that niobium is resistant to corrosion attack in liquid lithium up to temperatures of 800 °C. He noted that small quantities of oxygen in the niobium caused rapid penetrations at grain boundaries or along crystallographic planes where complex corrosion products were formed and oxygen was lost to the lithium.⁵⁶

In his analysis of niobium in liquid lithium, Freed³³ reported that temperatures of 816 up to 928 °C cause localized attack when the oxygen concentration of the niobium exceeded 500 ppm.

4.5.4 Water Coolant

Niobium is reported to have excellent mechanical strength properties in water with a temperature range of 600 to 800 °C.¹¹ However, its use has not progressed beyond the laboratory environment so there is little information available on its in-reactor performance.

From laboratory experiences, the corrosion rate law for unalloyed niobium has been approximated as being cubic during initial exposures in high-temperature water.¹¹ In the presence of the high-temperature water, niobium forms a nonprotective oxide film, Nb₂O₅. Over time the oxide film spalls and the corrosion kinetics become linear.⁵⁷

4.6 Stainless Steel (Type 316)

Austenitic stainless steels have been used extensively in experimental fusion reactor applications. Accordingly, the stainless steels have the most developed operational database for fusion when compared with the other tubing materials. Review of published operating experience literature suggested that Type 316 austenitic stainless steel was the best performer for ITER goals than other types of stainless steels. Therefore, the austenitic stainless steels are represented in this report by Type 316 stainless steel (18Cr-8Ni-Mo).

4.6.1 Failure Mechanism Resistance

At elevated temperatures, helium gas from (n, α) reactions make austenitic stainless steels particularly vulnerable to intergranular corrosion at temperatures of 510 to 788 °C.³³ This vulnerability results from heat sensitization of the material. Heat sensitization causes depletion of the chromium near the grain boundaries as chromium carbide is precipitated. The chromium-poor regions then corrode because of local galvanic cell action.³³

To reduce the susceptibility of austenitic stainless steels to intergranular corrosion, a variety of methods may be used. Collins³³ suggests limiting the carbon content below 0.03 weight percent, adding stabilizers to prevent depletion of the chromium near the grain boundaries, or employing a high-temperature solution heat treatment to produce a more homogeneous alloy.

Olander⁵⁸ reports that helium causes severe embrittlement of stainless steel. The extent of helium embrittlement was described as a function of neutron fluence ($E_n > 1$ MeV), material composition, and operating temperature.

Operating experience and laboratory observations by Cobbs⁵⁹ and Darras⁶⁰ noted that oxidation of stainless steel results in considerable scaling of the steel. The amount of scaling was determined to depend on exposure time and temperature. For Type 316 stainless steel in air at ambient pressure, the maximum exposure temperature without excessive scaling is believed to be 899 °C.⁶¹

Austenitic stainless steels exhibit very low corrosion rates in oxidizing media.¹¹ The steel's excellent resistance to oxidation is due to a protective oxide film that develops during exposure to the oxidizing media. Attack of the steels does not become significant until the solution's oxidizing power becomes great enough to locally breach the protective oxide. The solution's pH alone does not control attack. The presence of chloride and other strong "depasivating" chemicals deteriorate the material's corrosion resistance.¹¹

4.6.2 Helium Coolant

The oxidation rate kinetics of stainless steel in contaminated helium has proven to be parabolic.²⁷ Berry¹¹ reports that temperatures above 900 °C "breakaway" oxidation occurs in a relatively short period of time.

The helium contaminants that significantly affect Type 316 stainless steel are CO₂ and CO. In his experiments that exposed stainless steel to impure helium (3000 ppm CO) at 1010°C, Wunderlich³¹ reported that the steel exhibited significant intergranular attack by the CO.

4.6.3 Liquid Metal Coolant

The corrosion resistance of austenitic stainless steels in sodium at temperatures greater than 706°C is believed to be marginal.⁶² Berry¹¹ reports that the corrosion rate is proportional to the velocity of the liquid sodium.

In his analyses of liquid sodium-stainless steel combinations, Baker²² reports that Type 316 stainless steel exposed to liquid sodium acquires thin Cr₂O₃ films. The films cause small weight gains by the material, but primarily serve to increase the material's resistance to attack by shielding it from the sodium.

Liquid lithium has been safely confined by austenitic stainless steel up to a temperature of 500 °C.⁶³ Temperatures of 600 °C are believed possible if corrosion-inhibiting calcium or aluminum additives are incorporated.⁶³ With such additives, the corrosion rate is expected to be less than 0.7 nm/year.

The effects of liquid lithium on the tensile, creep, fatigue-life, and crack-growth properties of austenitic steels have been studied by Raeder⁶³ and Baker²². Most of the reported data is based on tests using static liquid lithium pots or tube specimens filled with liquid lithium. The significant results of the tests can be summarized as follows:

- a.) Tensile and creep strengths of austenitic stainless steels are not significantly affected by either static or flowing lithium.
- b.) Additions of hydrogen (≤ 40 ppm) in lithium have little or no effect on the tensile properties of austenitic stainless steels.
- c.) Nitrogen content in lithium plays an important role in controlling the fatigue properties of the austenitic stainless steels. Low-cycle fatigue life of austenitic steels in low-nitrogen flowing lithium (< 100 ppm) is superior to the fatigue life in air. Limited data also indicate that the fatigue life is independent of strain rate. However, fatigue life is significantly reduced in flowing lithium containing > 1000 ppm nitrogen due to intergranular attack of the material.
- d.) Aluminized layers formed by additions of aluminum to lithium have been found to decrease the mass transfer of Type 316 stainless steel in flowing lithium by a factor of 5.

Parkinson and Sisman⁶⁴ reported that type 316 stainless steel exhibited less than 0.5 mils of penetration after 100 hours of exposure to 538°C lithium flowing at 0.61 m/s. However, mass transfer of the steel becomes significant at liquid lithium temperatures above 600°C. Hoffman⁶⁵ reports that air-contaminated lithium further accelerates the mass transfer process.

Irradiation effects on Type 316 were also study by Parkinson and Sisman.⁶⁴ Their conclusions were that irradiation does not appear to greatly affect the corrosion process. The specimens they studied were irradiated to a fluence of 7×10^{16} n/cm² ($E_n > 1$ MeV) in 538 °C liquid lithium. Material analyses afterward showed that corrosion penetrations were less than 13 µm after 160 hours.

4.6.4 Water Coolant

Austenitic stainless steels normally perform well in water cooled systems due to their chromium content. Oxygen combines with the chromium to form a highly adherent and protective oxide film, Cr₂O₃. If the film is ruptured in certain oxidizing environments, it rapidly heals and prevents any substantial corrosion to occur. When the chromium concentration of the austenitic stainless steel is greater than 11%, the Cr₂O₃ layer is not continuous. Below the 11% threshold, corrosion resistance to oxygenated water is similar to unalloyed steel.

4.7 Titanium

Titanium is an unique material in that it is resistant to most modes of attack. This resistance is due to the formation of a tightly adherent chemical inert oxide film. In most corrosive media, this film readily reforms when it is broken or damaged.

Titanium also has high strength levels and is believed to be a low activation material. These properties make titanium a material of choice for in-vessel tubing.

4.7.1 Failure Mechanism Resistance

General intergranular attack of titanium alloys has been observed in very few media. Instead, the concern with titanium is hydrogen pickup. The absorption of hydrogen by titanium usually results in embrittlement. Hydrogen uptake is generally limited to situations where all of the following conditions are met: the pH of the medium is greater than 12 or less than 3, the protective oxide film has been damaged by abrasion, the temperature is greater than 77 °C, and a mechanism for generating hydrogen is present.⁶⁵

Fontana⁶⁷ in his study of titanium, says:

- a.) Hydrogen or tritium enhances the susceptibility of titanium alloys to cracking, especially with increased oxygen content,
- b.) Age hardening, increasing oxygen content, and decreasing temperature enhance hydrogen embrittlement in titanium alloys,
- c.) Titanium in contact with a corroding metal, such as steel, absorbs hydrogen and becomes embrittled.

Irradiation effects upon titanium's performance are extrapolated from experiences with titanium and boron control rods. Berry¹¹ and Belouin,⁶⁸ in separate analyses, reported that titanium displayed small weight gains (0.8 to 1.6 g/m² after 3000 hour exposure in 360 °C water) and suffered no significant loss in corrosion resistivity.

4.7.2 Helium Coolant

No information is available at the time of this report concerning titanium's behavior in a helium cooled environment.

4.7.3 Liquid Metal Coolant

Limited research has been conducted for titanium with liquid sodium. Lyon⁶⁹ does provide a few insightful comments. He suggests that titanium should be compatible with liquid sodium if the sodium is maintained with a high level of purity. Hydrogen impurities in the sodium will result in embrittlement of the titanium.

Titanium's behavior in liquid lithium has been studied by DiStefano.⁷⁰ He reports that high purity titanium resists attack by high purity liquid lithium up to temperatures of 800 °C. It was also noted that maximum oxygen concentrations of 2000 ppm in the titanium did not adversely affect its corrosion resistance to the lithium. However, oxygen concentrations in the liquid lithium were observed to cause embrittlement of the titanium.

4.7.4 Water Coolant

Titanium has recently been employed as tubing for heat exchanger in several nuclear power stations.⁷¹ The heat exchangers are generally used for limited periods and/or during emergency periods. These are situations that produce high corrosion rates due to stagnant water and possibly low water line. Titanium tubing has proven to be highly resistant to corrosion attack in these environments.⁷¹ At the time of this report, there have been no reported incidents of corrosion damage for condensers equipped with titanium tubes.⁷¹

A factor that usually influences the failure rate of a condenser is the coolant's velocity. Lindemann⁷¹ reports that the limiting water velocity for titanium tubes far exceed that for tubes manufactured with other materials. Titanium's performance in this area is due to the formation of thin, adhesive, and passive oxide layers that are self-curing even with low O₂ content. In his studies of titanium, Lindemann⁷¹ reports using maximum water velocities of 36.6 m/s without producing noticeable damage to the titanium.

4.8 Vanadium

Vanadium-based alloys offer tubing material advantages such as high strength levels at elevated temperatures, good corrosion resistance in lithium, and good resistance to radiation damage. The V-15Cr-5Ti alloy, originally developed under the fast breeder

reactor program, is normally the vanadium alloy of choice. The titanium fraction provides improved resistance to radiation damage and the chromium addition enhances high temperature material properties.

4.8.1 Failure Mechanism Resistance

An operational concern with vanadium-based alloys is their reactivity with hydrogen, nitrogen, and oxygen. As a strong hydride forming material, vanadium is embrittled when dissolved hydrogen reacts with it to form brittle hydride compounds. Fontana⁶⁷ suggests that vanadium's high strength properties may make the material more susceptible to crack propagation as a result of hydrogen embrittlement.

Vanadium's behavior after irradiation has been studied by Baker.²² Most of the irradiation data comes from operating experiences in the Liquid Metal Fast Breeder Reactor (LMFBR). Overall, vanadium alloys have shown a resistance to irradiation induced void swelling and loss of ductility.

The vanadium alloys used in LMFBRs were V-15-Cr-5Ti, V-20Ti, and V-15Ti-7.5Cr. Irradiation damage was induced by neutron ($E_n > 1$ MeV) and ion bombardment. At a temperature of 525 °C, V-15-Cr-5Ti had a peak swelling of 20 dpa and V-20Ti had 35 dpa. Further experiments by Baker show that titanium concentrations of only a few percent greatly suppress swelling in vanadium alloys when compared to unalloyed vanadium. V-15Cr-5Ti and V-15Ti-7.5Cr irradiated to 425 °C exhibited swelling of only 30 dpa. Irradiation at temperatures between 550 and 600 °C produced swelling levels of 40 dpa.

Radiation creep characteristics of V-20Ti were studied by exposing the alloy to a fluence of 2×10^{22} n/cm² ($E_n > 1$ MeV) at temperatures of 585 and 650 °C. Examination of the alloy after irradiation showed no effects upon the minimum creep rate of the unirradiated alloy.

4.8.2 Helium Coolant

The thermodynamic and kinetic processes for vanadium and VCrTi exposed to helium with low impurity concentrations have been evaluated in the Blanket Comparison and Selection Study (BCSS) Interim Report.²² The conclusion of the BCSS was that oxidation of vanadium alloys will be excessive if the alloys are exposed to helium that has impurity concentrations greater than 0.1 ppm and operating temperatures above 500 °C.

4.8.3 Liquid Metal Coolant

Vanadium is reputed to have low solubility in liquid sodium.¹¹ However, vanadium is very reactive with nonmetallic liquid sodium impurities such as oxygen, carbon, and nitrogen.¹¹ The critical oxygen concentration in liquid sodium, as related to vanadium oxidation varies with the sodium's temperature. At 600 °C, the critical concentration is 0.6 ppm while it increases to 6 ppm at 700 °C.⁷² Oxygen concentrations above 6 ppm cause significant embrittlement of the vanadium. In his analyses of vanadium alloys in liquid sodium, Smith⁷²

reported that vanadium-titanium alloys exhibited embrittled surface layers when exposed to sodium containing 5 to 10 ppm oxygen at 500 to 600°C.

Corrosion data for vanadium alloys in liquid lithium are primarily from three sources: the Pratt-Whitney Corporation, Oak Ridge National Laboratory (ORNL), and Argonne National Laboratory (ANL). The Pratt-Whitney data is from forced-circulation loop tests conducted in the late 1950's and early 1960's as part of the space power program. Data from ORNL is for capsule tests conducted in the early 1960s as part of the high temperature materials program. Argonne's data was generated during the late 1970's and early 1980's as part of the fusion materials program.

The three data sources focused primarily on unalloyed vanadium. The vanadium was exposed for 1170 hours in a liquid lithium forced circulation loop with a velocity of 4 m/s, maximum temperature of 870 °C, and loop ΔT of 220 °C. The reported corrosion of the vanadium was less than 1×10^{-6} g/cm².

4.8.4 Water Coolant

Baker²² has studied the use of vanadium in pressurized water-cooled systems. His conclusion was that vanadium has poor performance in such an environment because it is highly susceptible to the corrosion agents that exist from the chemistry control and natural environment of water-cooled systems. Vanadium performance was also limited by the material's affinity for hydrogen pickup and subsequent embrittlement.

Baker does support the use of vanadium alloys from the VCrTi group in pressurized water-cooled systems. His analyses showed that these alloys are more resistive to the corrosion agents that attack unalloyed vanadium. Corrosion rates for VCrTi alloys in 288 °C water for 500 hours were reported to be an order of magnitude lower than those for other common tubing materials.

4.9 Zirconium

Zirconium is available in two grades: reactor grade, which is hafnium-free; and commercial grade, which contains 2.5 percent hafnium. For improved corrosion resistance in high-purity water or steam at elevated temperatures, reactor-grade zirconium is alloyed with tin, iron, chromium, and nickel. The alloy is marketed as Zircaloy-2 (1.5Sn-0.12Fe-0.10Cr-0.05Ni additions to sponge grade zirconium). Sponge grade zirconium is produced by calcium or sodium reduction of $ZrCl_4$. Since Zircaloy-2 has displayed good performance in fission reactors as fuel cladding, the authors use it to draw general conclusions about zirconium-based tubing.

4.9.1 Failure Mechanism Resistance

In his limited analyses on the oxidation of zirconium, Berry¹¹ reported that the material can suffer significant oxidation effects. Most notably, zirconium is susceptible to oxidation from interactions with oxygen and CO₂.

Irradiation effects upon zirconium material properties have been studied by Kircher.³ Kircher reports that unalloyed zirconium at 280 °C and subjected to a neutron flux of 3×10^{12} n/cm²-s ($E_n < 0.04$ eV) experiences a significant decrease in creep rate. Chocki⁷³ has done similar work for Zircaloy-2. He observed an increased creep rate for temperatures above 480 °C and high stresses ($103 \leq \text{stress} \leq 138$ MPa). Experiments by Kircher³ on cold-worked and annealed Zircaloy-2 also show an increase in creep rate after irradiation at elevated temperatures.

The burst strength of irradiated Zircaloy-2 tubing has been tested by Kahle.⁷⁴ Kahle suggests that tubing burst strength is appreciably increased by irradiation, however, subsequent fractures occur in a ductile manner.

4.9.2 Helium Coolant

Zirconium in a helium-cooled environment has been studied by Cobbs.^{10,50,61} In high purity helium, Cobbs reports that zirconium is generally resistant to corrosion attack. However, impurities in the helium affect zirconium's corrosion resistance. Zirconium's decreased resistance in impure helium is due to the material's affinity for gettering gaseous contaminants.

4.9.3 Liquid Metal Coolant

Zirconium's solubility in liquid sodium has been observed by Kane⁷⁵ to depend on the sodium's oxygen concentration. Kane reported significant material solubility at a liquid sodium temperature of 722 °C and oxygen concentration of 0.09 ppm. Material solubility was also observed to increase at a temperature of 1380 °C and an oxygen concentration of 3.6 ppm. When the oxygen concentration of the liquid sodium exceeded 15 ppm, the material exhibited significant mass transfer effects regardless of the sodium's temperature.

Literature was not found for experiences with zirconium in liquid lithium.

4.9.4 Water Coolant

The corrosion behavior of zirconium in water-cooled systems has been observed to be dependent on the formation of a very thin, dense, and adherent oxide film.⁷⁶ This film is impervious to ion diffusion and will prevent corrosion as long as it remains intact. If the film is damaged, a sudden increase is seen in the localized corrosion rate. The corrosion rate has also exhibited a proportionality with temperature.

Berry¹¹ reports that the corrosion rate of Zircaloy-2 in high-temperature water is not affected by water velocities below 10 m/s. He further reports that Zircaloy-2 has a higher tolerance for impurities than does unalloyed zirconium. However, concentrations of nitrogen above 100 ppm or oxygen above 2700 ppm significantly reduce Zircaloy-2 corrosion resistance in water above 360 °C.

Berry also studied the effects of irradiation on the corrosion behavior of Zircaloy-2 in water. In high quality steam at 340 °C and a neutron flux of 3×10^{13} n/cm²-s ($E_n > 1$ MeV), corrosion rates increased by a factor eight above those for unirradiated samples. The uptake of hydrogen from the corrosion reactions was seen to result in eventual embrittlement of Zircaloy-2.

4.10 Chapter Summary

The preceding sections provide available failure data for the piping materials. These data presented in the sections are an integral component of this report since they will define the multipliers that will generate ITER in-vessel tubing failure rates. This summary section categorizes the failure data so that critical failure rate information is extracted.

The authors identified ten failure mechanisms for in-vessel tubing. The identification criterion was the frequency of occurrence in published tubing operating experiences. The intent of this chapter was to obtain information from industry operating experiences on the resistance characteristics of the tubing materials to the identified failure mechanisms. Occasionally, available failure data does not address each of the ten failure mechanisms. When the reviewed failure data does not address a failure mechanism, an assumption is made on the material's minimal expectant resistance to that particular mechanism. This assumption is based on the mechanical properties and general knowledge on the material.

The authors assume that tubing materials can have one of the three possible levels of resistance to the individual failure mechanisms: poor, acceptable, or excellent. The level of resistance comments on the magnitude of attack upon a material's mechanical integrity when it is exposed to the particular failure mechanism.

A resistance level of "poor" implies that the material is highly susceptible to failure when exposed to the failure mechanism for greater than 60 hours at temperatures above 400 °C. An "acceptable" level of resistance implies minimal attack upon the material by the failure mechanism during exposures up to 8.76×10^3 hours (1 year) and temperatures between 400 and 600 °C. A material that has an "excellent" level of resistance to a failure mechanism suffers no attack by the mechanism during exposures up to 8.76×10^3 hours and temperatures between 500 and 600 °C. Table 4-2 summarizes the levels of failure mechanism resistance.

Table 4-2
Material Levels of Resistance to Failure Mechanisms

Failure Mechanism Rate of Attack	Material Level of Resistance	Exposure	
		Temperature Range (°C)	Time Range (hr)
severe	poor	< 500	< 60
normal	fair	500 - 600	≤ 8.76E+3
none	excellent	500 - 600	≤ 8.76E+3

For representational purposes, the levels of resistance are assigned the following symbolic values: poor = -1, fair = 0, and excellent = +1. These symbols are used to categorize the materials. Table 4-3 shows the relation between the symbolic values and the levels of resistance. The symbolic representation of the levels of resistance can be used to categorize the tubing materials. This categorization will be used in a Chapter 6 to generate tubing failure rates. Table 4-4 presents the levels of resistance categorization for the tubing materials. The categorization is taken from the failure data presented in this chapter.

Table 4-3
Symbolic Representation of Levels of Resistance

Symbolic Representation	Level of Resistance
-1	poor
0	fair
+1	excellent

Table 4-4
Levels of Failure Resistance for Tubing Materials

	Be	Cu	Inconel	Nb	316 ss	Ti	V	Zr
Failure Mechanism								
Creep	0	-1	0	0	0	0	0	0
Cyclic Fatigue	-1	-1	0	-1	0	0	0	0
Erosion	0	0	0	0	0	0	0	0
Helium Swelling	-1	0	0	+1	-1	0	0	0
Hydrogen Embrittlement	0	-1	0	0	-1	-1	-1	0
Intergranular Attack	0	0	0	0	-1	0	0	0
Irradiation Embrittlement	-1	-1	0	-1	0	0	0	0
Magnetic Fields Interactions	+1	-1	-1	+1	+1	+1	+1	+1
Oxidation	-1	-1	0	-1	-1	-1	-1	-1
Thermal Cycling	0	0	0	0	0	0	0	0
Coolant Type								
Helium Gas	0	0	0	0	0	0	-1	0
Liquid Lithium	0	-1	0	0	0	0	0	0
Liquid Sodium	0	-1	0	+1	0	0	0	-1
Water	0	0	0	0	+1	+1	0	-1

Chapter References

1. Brush Wellman Inc., Designing With Beryllium, Product Brochure, BWI, Cleveland, OH, October 1993.
2. Kircher, John F. and Bowman, Richard E., Effects of Radiation on Materials and Components, New York: Reinhold Publishing Corporation, 1964.
3. Clairborne, H. C., Effects of Fast Neutron Reactions in the Beryllium Reflector of the HTR, ORNL-60-3-10, Oak Ridge National Laboratory, March 1960.
4. Rich, J. B., Redding, G. B., and Barnes, R. D., The Effect of Heating Neutron Irradiated Beryllium, United Kingdom Atomic Energy Authority, AERE-M/R 2685, September 1958.
5. Gulbransen, E. A., and Andrew, K. F., Journal of the Electrochemistry Society, 11, 383, 1950.
6. Cubicciotti, D., Journal of American Chemistry Society, 72, 2084, 1950.
7. Williams, J., and Munro, W., "The Reaction of Beryllium with Carbon Dioxide in the Temperature Range 500 to 700 °C," AERE-M/M-108, February 1956.
8. Bradshaw, W., and Wright, E. S., "High-Temperature Corrosion of Beryllium in Air," U.S. Report LMSD-288232, January 1960.
9. Antil, J.E., and Higgins, J.K., "Oxidation of Beryllium, "Proceedings of the US/UK Meeting on the Compatibility problems of Gas-Cooled Reactors, Oak Ridge National Laboratory, February 24-26, 1960; USAEC Report TID-7597.
10. Cobbs, J.H., "Advanced Materials Development," Paper presented at Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958, USAEC Report TID-7564.
11. Berry, Warren, Corrosion in Nuclear Applications, John Wiley and Sons, New York 1971
12. Stang, J.H., et al., "Compatibility of Liquid and Vapor Alkali Metals with Construction Materials," DMIC Report 227, April 15, 1966.
13. Kelman, LeRoy R., Wilkinson, Walter D., and Yagee, Frank L., Resistance of Materials to Attack by Liquid Metals, ANL-4417, Argonne National Labortaory, 1950
14. Stonehouse, A.J., and Beaver, W.W., "Beryllium Corrosion and How to Prevent It," Material Protection, 4 (1), 24-28 January, 1965.
15. Peoples, R.A., Bulkowski, H., "Corrosion of Beryllium in 525 °F and 600 °F Distilled Water," USAEC Report BMI-HAP-101, September 12, 1949.

16. Eichelberger, R.L., and McKisson, R.L., "The Solubility of Cu, Mo, Nb, Fe, Ni, and Cr in High-Purity Sodium," Proceedings of the International Conference on Sodium technology and Large Fast Reactor Design, November 7-9, 1968, Part I. Sessions on Sodium Technology; USAEC Report ANL-7520, Part I.
17. Eichelberger, R. L., and McKisson, R. L., "Solubility of Copper in Sodium," USAEC Report AI-AEC-12671, May 31, 1968.
18. Singer, R. M., and Weeks, J. R., "On the Solubility of Cu, Ni, and Fe in Liquid Sodium," Proceedings of the International Conference on Sodium Technology and Large Fast Reactor Design, November 7-9, 1968, Part I. Sessions on Sodium Technology; USAEC Report ANL-7520, Part I.
19. Osborg, H., Lithium, Theoretical Studies and Practical Applications, The Electrochemical Society, 1935.
20. Kofstad, Per, High-Temperature Oxidation of Metals, John Wiley & Sons, New York 1966.
21. Shreir, L., ed., Corrosion of Metals and Alloys, John Wiley & Sons, New York 1963.
22. Baker, C., Blanket Comparison and Selection Study - Final Report, ANL/FPP-84-1, Argonne National Laboratory, 1984.
23. Bokros, J. C. and Shoemaker, H. E., "Reactor Materials Compatibility with Impurities in Helium," USAEC Report GA-1508.
24. Wunderlich, J. W. and Baker, N. E., "Exposure of HTGR Candidate Core Plate and Thermal Insulation Materials to Impure Helium at 1650 °F to 1850 °F for 3000 Hours," USAEC Report GAMD-7377, December 29, 1966.
25. Kosiba, W. L., "Irradiation Effects on Carbon Mass Transport by Impurities in Helium," Proceedings of US/UK Meeting on the Compatibility Problems of Gas-Cooled Reactors, Oak Ridge National Laboratory, February 24-26, 1960; USAEC Report TID-7597.
26. Frye, J. H., Jr., Manly, W. D., and Cunningham, J. E., "Metallurgy Division Annual Progress Report for Period Ending October 10, 1957," USAEC Report ORNL-2422, December 13, 1957.
27. Coriou, H., et al., "Stress-Corrosion Cracking of Inconel in High-Temperature Water," 3^e Colloque de Metallurgie sur la Corrosion, 1959, North Holland Publishing Company, 1960.
28. Roy, P., Wozadlo, G. P., and Comprelli, F. A., "Mass Transport and Corrosion of Stainless Steels in Flowing Sodium Systems at 1300 °F," Proceedings of the International Conference on Sodium Technology and Large Fast Reactor Design, November 7-9, 1968,

Part I. Sessions on Sodium Technology; USAEC Report UNL-7520.

29. DeVan, J. H., "Corrosion of Iron- and Nickel-Base Alloys in High-Temperature Sodium and NaK," Proceedings of the Symposium on Alkali Metal Coolants - Corrosion Studies and System Operating Experience - Held by the International Atomic Energy Agency, Vienna, November 28 - December 2, 1966.

30. Chamberlain, Harry V., Coleman, John H., Kovacic, Evan C., and Shoudy, Albert A., "Wastage of Steam-Generator Materials by Sodium-Water Reactions," Proceedings of the International Conference of Sodium Technology and Large Fast Reactor Design, November 7-9, 1968, Part I. Sessions on Sodium Technology; USAEC Report ANL-7520, Part I.

31. Ovier, R. B., et al., "Effect of Environment on Creep Properties of High-Temperature Alloys," paper presented at ANP-Materials Meeting, November 16-18, 1954; USAEC Report ORNL-2685, March 26, 1959.

32. Miller, N.E. and et al., "Carbon Transport and Corrosion in High-Temperature Gas Cooled Reactors," Proceedings of the Conference on Corrosion of Reactor Materials, Salzburg, Austria, June 4-8, 1962, USAEC Report GA-2630.

33. Freed, M.S. and Kelly, K.J. (Eds.), "Corrosion of Columbium Base and Other Structural Alloys in High Temperature Lithium," Report No. PWAC-355, Pratt & Whitney Aircraft - CANEL, Division of United Aircraft Corporation (June 30, 1961) (Declassified in June, 1965).

34. Hoffman, E. E. and Harrison, R. W., "The Compatibility of Refractory Metals with Liquid Metals," Proceedings of a Symposium on Metallurgy and Technology of Refractory Metals, held in Washington, D. C., April 25-26, 1968.

35. Wilkinson, W. and et al., "Resistance of Materials to Attack by Lithium," unpublished data and preliminary data reported in ANL Metallurgy Division Quarterly Progress Reports for Period July Through December 1949, ANL-4364 and 4399.

36. Galonian, G. E., and Tymchyn, H. L., "Corrosion of Inconel in 600 °F Static Water," USAEC Report KAPL-2047, May 4, 1959.

37. Boujant, P. and LeSurr, J., "Some Observations of Intergranular Corrosion on Iron and Nickel Alloys in High Purity Water," Paper presented at the 24th National Association of Corrosion Engineers Conference, AECL-2800, March 1968, (April 1968).

38. Coriou, H., et al., "Sensitivity to Stress Corrosion and Intergranular Attack of High-Nickel Austenitic Alloys," Corrosion, 22 (10), October 1966.

39. Copson, H. R., Economy, G., "Effect of Some Environmental Conditions on Stress-Corrosion Behavior of Nickel-chromium-Iron Alloys in Pressurized Water," Paper presented at the International Nickel Power Conference, Lausanne, Switzerland, 1967; also

Corrosion, **24** (3), March 1968.

40. Makin, J. J. and Minter, F. J., "Mechanical Properties of Irradiated Niobium," Acta Metallurgy, **7**, 361, June 1959.

41. Ferro, A., Mazzetti, P. and Montalenti, G., Phil. Mag., Vol. 12, 1965.

42. Chung, D. W. and Stoloff, N. S., Met. Trans. A, Vol. 9A, 1978.

43. Doner, M., Diprimio, J. C. and Salkovitz, E. I., Acta Metallurgica, Vol. 21, 1973.

44. Leadbetter, M. J. and Argent, B. B., Journal of Less Common Metals, Vol. 3, 1961.

45. Enrietto, J. F., Sinclair, G. M., and Wert, C. A., Columbium Metallurgy, New York: Interscience, 1961.

46. Donoso, J. R. and Reed-hill, R. E., Metallurgy Transactions, Vol. 7A, July 1976.

47. Ransow, J. and Argent, B., Journal of the Institute of Metals, Vol. 95, 1967.

48. Brissow, G. and Argent, B. B., J. of the Inst. of Metals, Vol 91 1962-63.

49. Shoemaker, H.E., "Carbon Transport in Gas-Cooled Reactors," in Proceedings of the US/UK Meeting on the Compatibility Problems of Gas-Cooled Reactors, Oak Ridge National Laboratory, February 24-26, 1960; USAEC Report TID-7597.

50. Kosiba, W. L., "Irradiation Effects on Carbon Mass Transport by Impurities in Helium," Proceedings of US/UK Meeting on the Compatibility Problems of Gas-Cooled Reactors, Oak Ridge National Laboratory, February 24-26, 1960; USAEC Report TID-7597.

51. Davis, M., and Draycott, A., "Compatibility of Reactor Materials in Flowing Sodium," Second International Conference on the Peaceful Uses of Atomic Energy, Vol. 7, Reactor Technology, United Nations, Geneva, 1958.

52. Gurinsky, D. H. and et al., "Corrosion in Liquid Metals Systems," third International Conference on the Peaceful Uses of Atomic Energy, Vol 9, Reactor Materials, 1965.

53. Anthrop, Donald F., "Solubilities of Transition Metals in Liquid Alkali and Alkaline Earth Metals, Lanthanum and Cerium - A Critical Review of the Literature," USAEC Report UCRL-50315, September 20, 1967.

54. Cleary, R. E., et al., "Solubility of Refractory Metals in Lithium and Potassium, USAEC Report TIM-850, November 1965.

55. DiStefano, J. R., "Corrosion of Refractory Metals by Lithium," USAEC Report ORNL-3551, March 1964.

56. DeVan, J. H., Litman, A. P., DiStefano, J. R., and Sessions, C. E., "Lithium and Potassium Corrosion Studies with Refractory Metals," Proceedings Series, Alkali Metal Coolants, International Atomic Energy Agency, Vienna, 1967.
57. Cox, B., and Johnston, T., "The Oxidation and Corrosion of Niobium," Report from Chemistry Division, Atomic Energy Research Establishment, Harwell, Berkshire, England, March 1961.
58. Olander, Donald, Fundamental Aspects of Nuclear Reactor Fuel Elements, Technical Information Center, Energy Research and Development Administration, U.S. Department of Energy, 1976.
59. Cobbs, J., "Advanced Materials Development," Paper presented at Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21, 22, 1958, USAEC Report TID-7564.
60. Darras, R., "Corrosion Problems in Gas Cooled Reactors," IGIS-8 (RD/W), Translation from Bulletin of Information Scientifiques et Techniques New Series, No. 21, September 1958.
61. "Steels for Elevated Temperature Service", United States Steel, 1966.
62. Brasunas, Anton De S., and Stansbury, E. E., "Symposium on Corrosion Fundamentals", The University of Tennessee Press, 1956.
63. Raeder, J., Borrass, K., Buende, R., Danner, W., Klingelhofer, R., Lengyel, L., Leuterer, F., Soll, M. Controlled Nuclear Fusion: Fundamentals of its Utilization for Energy Supply, New York: John Wiley & Sons Ltd, 1986.
64. Parkinson, W. and Sisman, O., "Liquid Metal Loops Irradiated in the ORNL Graphite Reactor and the LITR", USAEC Report ORNL-2630, June 1, 1959.
65. Hoffman, E. E., "Corrosion of Materials by Lithium at Elevated Temperatures," USAEC Report ORNL-2924, October 27, 1960.
66. Schutz, R. and Covington, L., Guidelines for Corrosion Testing of Titanium, ASTM Special Technical Publication 728, October 1979.
67. Fontana, Mars, Corrosion Engineering, McGraw-Hill, New York 1986.
68. Belouin, D. and Merten, U., "The Corrosion Behavior of Some Zirconium and Titanium-Base Boron Alloys in High Temperature Water," USAEC Report, KAPL-M-UM-6, June 15, 1956.
69. Lyon, R. (ed.), Liquid Metal Handbook, 2nd Edition, U.S. Government Printing Office, June 1952.

70. DiStefano, J., "Corrosion of Refractory Metals by Lithium," USAEC Report, ORNL-3551, March 1964.
71. Lindemann, J., "Manufacture and Properties of Titanium Tubes for Power Station Condensers," Titanium and Titanium Alloys, Vol. 3, New York: Plenum Press, 1982.
72. Smith, D. and Kassner, T., "Application of Thermodynamic and Kinetic Parameters of the V-O-Na System to the Sodium corrosion of Vanadium Base Alloys," Corrosion by Liquid Metals, J. Draley and J. Weeks (Eds.) Plenum Press, New York 1970.
73. Chocki, L. J., In-Reactor Creep Measurements of Zircaloy, 2HW-64814, April 1960.
74. Kahle, V. E., Burst Testing of Irradiated Zircaloy-2 Tubing, GE-HAPO, HW-64168, March, 1960.
75. Kane, R. and Boyd, D., "Use of Titanium and Zirconium In Chemical Environments," Industrial Applications of Titanium and Zirconium, ASTM Special Technical Publication 728, October 1979.
76. Industrial Applications of Titanium and Zirconium, ASTM Special Technical Publication 728, October 1979.

5.0 DATA COMPILATION

5.1 Introduction

This chapter presents the operating experience failure rates compiled for the tubing materials from various industries, primarily fission and commercial power production. For each of the failure rates, a brief discussion on the associated operating environment is provided. This discussion is intended to provide background information on the failure rate.

The compiled failure rates are in units of failure per meter-hour, $(m-hr)^{-1}$ and represent the two failure modes possible for tubing: leakage and rupture. The leakage mode is defined as the amount of coolant leakage into the vacuum vessel that is sufficient to trigger a plasma disruption.

The leakage amount to yield a plasma disruption has been set by experience on the Tore Supra machine, where 0.5 liter of water from a burned out cooling tube caused a disruption.¹ Merrill² has investigated coolant induced disruptions and found that as little as 30 grams of coolant (about 0.03 liter of water entering the plasma chamber in a second) can cause a disruption twice as intense as a typical disruption. Therefore, very small leaks are defined as the threshold for the leakage failure mode. The rupture failure mode is defined as the injection of coolant from the tubing into the vacuum vessel at a rate equivalent to the operational flow-rate.

Sometimes, only one of the two tubing failure rates (leakage or rupture) is given in the published literature. As mentioned earlier, tubing is more likely to fail by leakage than by rupture. This philosophy suggests that the leakage failure rate is larger than the rupture failure rate. For the cases where the literature does not provide both leakage and rupture failure rates, the two rates are assumed to be related by a factor of 10%. The 10% factor is derived from piping reliability research by Thomas.³ The validity of the 10% factor has been checked by considering the ratio of leakage to rupture failure rates from operating experience. Published failure rates for Inconel 600 show that the ratio has a 4% factor. This suggests that the assumption of a 10% factor should be adequately conservative for the other materials.

The failure rates presented in this chapter are given an associated error factor. This factor is intended to convey the level of confidence given to the failure rate. The error factor is formally defined as the square root of the quotient of the upper bound (95 percentile) failure rate and the median (50 percentile) failure rate. For high quality (low uncertainty) data, error factors typically have a value around 3. Data of medium quality are seen to have a value of about 5. Data that have a low level of confidence have an error factor with values of 10 or more. When error factors are not given, the authors assign values based on this statistical guidance.

For many of the materials, operating experience failure rates for liquid sodium and liquid lithium coolant systems are not found. Fortunately, John⁴ has determined that the

differences between the two liquid metals are not significant once one accounts for material compatibility issues. He assumes that liquid sodium component failure rates also apply to liquid lithium components. Thus, when review of published operating experiences does not provide failure rates for liquid lithium or liquid sodium, the given failure rate for one is assumed applicable to the other.

The following sections primarily present operating experience failure rates for single-walled tubing. There are also subsections on duplex and double walled tubing failure rates. Duplex tubing features two tubing materials intimately joined so that there is continuous contact between the two materials throughout the length of the tubing. Duplex tubing has only one flow path. Double-walled tubing is tubing manufactured with the primary tube enclosed by a secondary tube, creating an annular flow channel separate from the main flow path.

During the latter development of this report, ferritic steels (2.25% chromium, 1% molybdenum) were suggested as a potential tubing material. To address this issue, the authors accepted the responsibility of generating in-vessel tubing failure rates for ferritic steels. Unlike the other tubing materials, which are categorized by the levels of resistance to the identified tubing failure mechanisms, the failure rates generated for ferritic steels are based on the failure categorization of a similar steel, austenitic Type 316. This correlation with austenitic steels implies a greater level of uncertainty for the ferritic steel in-vessel failure rate, however, the objective of the failure generation was to obtain an order of magnitude approximation.

This chapter also includes failure rate data for tubing related components such as welds and multiple leakage. The presented failure rates are intended to serve as a reference and are not used in determine in-vessel tubing material failure rates.

The failure rates compiled in this chapter are based on equipment designed in accordance with applicable design standards, such as the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

5.2 Beryllium

Review of published literature did not provide operating experience failure rates for beryllium tubing. Thus, an extrapolation or cross reference is necessary to represent beryllium tubing's probability of failure. This extrapolation is based on beryllium's tendency to swell and embrittle during neutron irradiation.

Beeston⁵ reports that beryllium reflectors in fission test reactors function as completely brittle members. This information can be used with research results by Carroll⁶ to cross reference operating experience failure rates for beryllium tubing. In his reliability analysis of polyvinyl chloride (PVC) pipe, Carroll reported that PVC is more brittle than most engineering materials. Thus, as a first approximation, operating experience tubing failure rates for beryllium are assumed to be of the same magnitude as those for all diameters of

PVC pipe. Beryllium's actual performance as a tubing material should be better than the failure rates assumed from PVC. Therefore, the PVC pipe failure rates are actually a conservative upper bound for beryllium rates until testing provides better values.

The authors assume a pipe length of 3 meters when providing failure rates for PVC pipe. The average rupture failure rate for such pipe is given as $3\text{e-}07 \text{ (m-hr)}^{-1}$.⁷ An error factor of 10 is associated with this failure rate since beryllium's performance is being inferred from that of PVC pipe.

Chemical engineers data⁷ does not provide leakage failure rates for PVC pipe so the leakage failure rate for PVC pipe is assumed to be an order of magnitude larger (10% factor) than the rupture failure rate. Thus, the average leakage failure rate for beryllium tubing is $\{(3\text{e-}07) \times 10\} 3\text{e-}08 \text{ (m-hr)}^{-1}$. Like the leakage failure rate, the error factor associated with the rupture failure rate has a value of 10.

Operating experience failure rates for beryllium with liquid metals or helium were not found during the review of published literature. Therefore, until material testing provides better values, the leakage and rupture failure rates for water coolant are applied to helium, liquid sodium, and liquid lithium coolants.

5.3 Copper

Published operating experiences^{8,9} for the resistive magnet system in the Tokamak Fusion Test Reactor (TFTR) were reviewed to compile failure rates for copper tubing. The magnets use copper tubing as hollow magnetic conductors as well as a cooling channel. Operating experiences for copper tubing in air-conditioning units and residential plumbing were also reviewed, however, they lacked sufficiently documented failure information to obtain failure rates.

The TFTR toroidal field coils are used to determine failure rates for copper tubing. In TFTR, there are 20 toroidal field (TF) magnet coils. Each coil has approximately 540 meters of copper tubing for a total tubing length of $\{(540 \text{ m}) \times (20 \text{ coils})\} 10,800 \text{ meters}$.⁹ In the ten years of TFTR operation, there have been six reported leaks in the magnet cooling tubing.⁹ However, the leaks have occurred at joint brazes rather than through tubing walls.

The coolant channels in the copper tubing have a cross sectional area of $143\text{e-}6 \text{ m}^2$ and an average wall thickness of $4.2\text{e-}3 \text{ meters}$. Water circulates through the coolant channels with a velocity of 3 m/s, pressure of 1.9 MPa, inlet temperature of 10 °C, and outlet temperature of 27 °C.⁹

According to published TFTR failure experience data^{8,9}, there have been no tubing leaks with the copper conductor. This absence of tubing leaks is expected to be a consequence of the tubing wall thickness. To generate failure rates for a system that has not experienced a failure, one typically uses the 50% Chi-square distribution with two degrees

of freedom.¹⁰ Applying the formula yields a leakage failure rate of $\{1.39/[2 \times (10800 \text{ m}) \times (5931 \text{ hr})]\} 1.1\text{e-}8 \text{ (m-hr)}^{-1}$. An error factor of 3 is assumed for this failure rate since the tubing information from TFTR is assumed accurate.

During the operation of TFTR, there have been no published accounts of rupture in the copper conductor. The rupture failure rates for copper tubing in water coolant is calculated using the 10% ratio between leakage and rupture failure rates. Using this ratio and the leakage failure rate yields a rupture failure rate of $\{[1.1\text{e-}8 \text{ (m-hr)}^{-1}]/10\} 1.1\text{e-}9 \text{ (m-hr)}^{-1}$. Similar to the leakage failure rate, the error factor associated with the rupture failure rate is 3.

Operating experience failure rates for copper with helium was not found during the review of published literature. Therefore, until material testing provides better values, the leakage and rupture failure rates for water coolant will be applied to the helium coolant. Copper is readily soluble in liquid sodium and liquid lithium, therefore, failure rates are not generated for those material-coolant combinations.

5.4 Ferritic Steel

Cadwallader¹¹ reported leakage failure rates for ferritic steels in helium-coolant systems. His analyses were for ferritic steel pipe so adjustments are required to make the failure rate applicable to tubing. Pipes have a larger wall thickness than tubing so they should be more resistance to leakage and rupture. Thus, the leakage failure rate, $4\text{e-}11 \text{ (m-h)}^{-1}$, given by Cadwallader is increased by a factor of 10 to yield a leakage failure rate of $\{[4\text{e-}11 \text{ (m-h)}^{-1}] \times 10\} 4\text{e-}10 \text{ (m-h)}^{-1}$. The error factor associated with this failure rate is 10 due to uncertainties in extrapolating from pipe to tubing.

The rupture failure rate for ferritic steel tubing in helium is calculated using the 10% ratio between leakage and rupture failure rates. Using this factor and the tubing leakage failure rate yields a rupture failure rate of $\{[4\text{e-}10 \text{ (m-hr)}^{-1}]/10\} 4\text{e-}11 \text{ (m-hr)}^{-1}$. Similar to the leakage failure rate, the error factor associated with the rupture failure rate is 10.

The operating environment for ferritic steel tubing in the helium coolant systems featured a system pressure of 6 MPa, coolant velocity of 4 to 5 m/s, coolant temperature of 730 °C, and moderately levels of helium impurities.

Operating experience failure rates for ferritic steel tubing in liquid sodium coolant is from the Fermi-1 liquid sodium-cooled fission reactor that operated from 1963 until 1972. The reactor had 3 steam generators that featured ferritic steel tubing. Each steam generator had 1200 tubes. Each tube had a length of 24.5 meters, outer diameter of $15.9\text{e-}3$ meters, and wall thickness of $1\text{e-}3$ meters. The tubes were single walled and internally exposed to steam at 6 MPa and with a velocity of 10 m/s. Externally, the tubes were exposed to liquid sodium flowing at 1 m/s and with a temperature of 440 °C.¹²

From plant data, the total number of plugged tubes was 52. Here it is assumed that a plugged tube is one that was leaking. Plant data from Fermi-1 suggest that the reactor operated below current power plant availability averages. Therefore, plant availability is assumed to be 50%. For its ten year operational life, this assumption gives Fermi-1 an operating time of $\{(365 \text{ days per year}) \times (24 \text{ hours per day}) \times (10 \text{ years}) \times (0.5 \text{ availability})\}$ 43.8×10^3 hours. Total tubing length is calculated as $\{(3 \text{ steam generators}) \times (1200 \text{ tubes per steam generator}) \times (24.5 \text{ m per tube})\}$ 88.2×10^3 meters. Thus, the leakage failure rate is calculated as $\{52 / [(43.8 \times 10^3 \text{ hr}) \times (88.2 \times 10^3 \text{ m})]\}$ $1.35 \times 10^{-10} \text{ (m-hr)}^{-1}$. An error factor of 10 is associated with this failure rate due to the uncertainty in plant availability.

The rupture failure rate for ferritic steel tubing in liquid sodium is calculated using the 10% ratio between leakage and rupture failure rates. Using this factor and the tubing leakage failure rate yields a rupture failure rate of $\{1.35 \times 10^{-10} \text{ (m-hr)}^{-1} / 10\}$ $1.35 \times 10^{-11} \text{ (m-hr)}^{-1}$. Similar to the leakage failure rate, the error factor associated with the rupture failure rate is 10.

5.5 Ferritic Steel Duplex Tubes

Using operating experiences from liquid metal fission reactors, John⁴ calculates leakage failure rates for duplex walled ferritic steel tubing. John calculated a leakage failure rate of $6.8 \times 10^{-11} \text{ (m-hr)}^{-1}$ for ferritic tubing in fusion reactor steam generators. An error factor of 10 is associated with the failure rate since it was calculated for steam generator tubing.

The rupture failure rate for ferritic steel duplex walled tubing is calculated using the 10% factor between leakage and rupture failure rates. Using this factor and the tubing leakage failure rate yields a rupture failure rate of $\{6.8 \times 10^{-11} \text{ (m-hr)}^{-1} / 10\}$ $6.8 \times 10^{-12} \text{ (m-hr)}^{-1}$. Similar to the leakage failure rate, the error factor associated with the rupture failure rate is 10.

5.6 Ferritic Steel Double-Walled Tubes

Operating experiences for double-walled tubing were not found during the review of published literature. However, it is believed that inferences may be taken from the stainless steel pressure tube concept applied in the Canadian Deuterium-Uranium (CANDU) reactors. CANDU reactors use the pressure tube concept where the fuel cladding functions like an interior tube. CANDU operating experiences have been favorable, with only one pressure tube rupture occurring during 150 reactor-years of operation.¹³

Since actual failure rates were not found for double-walled tubing, corresponding failure rates for ITER in-vessel tubing are not generated.

5.7 Inconel 600

Failure rate sources for Inconel 600 include tubing used in fission power plant steam generators, heat exchangers, and condensers. Steam generator tubing is chosen to provide the base failure rate since this tubing is believed more similar to ITER in-vessel tubing: thin walled, close packed, and exposed to radiation fields. Caution must be

exercised when using steam generator tubing to calculate failure rates for ITER since the steam generator tubes are subjected to additional failure modes by being exposed to water and water/steam on both sides of the tubes.

Tatone¹⁴ reports that Inconel 600 suffers more degradation in steam generator service than other nickel-base alloys. Inconel 600 tubes with an outer diameter of $22\text{e-}3$ meters and a wall thickness of $1.3\text{e-}3$ meters are given a leakage failure rate of $2.9\text{e-}8$ (m-hr)⁻¹. Tatone does not provide an error factor with his calculated failure rate. However, the number of plugged tubes and system operating time are known so the error factor is given a value of 3.

With the failure rates that he calculates for Inconel 600, Tatone also specifies the influence percentages of the failure mechanisms. The failure mechanism percentages are: stress corrosion cracking (id), 45%; stress corrosion cracking (od), 24%; fretting, 12%; pitting, 10%; phosphate wastage, 4%; denting, 4%; erosion, 0.4%; fatigue, 0.1%; and other modes, 4%.¹⁴

For ITER, the outer diameter failure mechanisms can be removed since they will not occur in ITER's vacuum conditions. Factoring the 24% from stress corrosion cracking, 4% from phosphate wastage, and 0.4% from denting yields a new leakage failure rate of $\{[1 - (.24 + .04 + 0.004)] * 2.9\text{e-}8$ (m-hr)⁻¹ $\}$ $2.1\text{e-}8$ (m-hr)⁻¹. The fretting failure mechanism is retained since there may be fretting concerns with the in-vessel tube banks for ITER.

Inconel 600 tubing rupture failure rates are given by Adams.¹⁵ Adams' rupture failure rates are for U-tube steam generators from fission power plants. The rate he reports is $1\text{e-}02$ (reactor-yr)⁻¹. Assuming that a steam generator has 3600 tubes and each tube has a length of 20 meters, the total length of tubing in the steam generator is $\{3600 \times 20 \text{ m}\}$ 72000 m. A fission power plant generally has four steam generators so the total length of tubing for the steam generator system is $\{(72\text{e}+3 \text{ m}) \times (4)\}$ 288e+3 meters. Average plant availability is usually 70%. Total plant operating time is calculated as $\{(365 \text{ days}) \times (24 \text{ hours}) \times (0.7 \text{ availability})\}$ 6.1e+3 hours. Using the failure rate, total tubing length, and reactor operating hours to calculate a rupture failure rate yields $\{[1\text{e-}2 \text{ failures}]/[(288\text{e}+3 \text{ m}) \times (6.1\text{e}+3 \text{ h})]\}$ $5.7\text{e-}12$ (m-hr)⁻¹. An error factor of ten is associated with the calculated failure rate due to the assumptions on plant availability, tubing length, and the number of tubes.

Assumptions for the Inconel 600 operating environment are: high pressure differential, 8 MPa; high radiation levels, 50 Rad/hr¹⁶; low water temperatures, 280 °C; and moderate flow velocities, 3-4 m/s in tubes and 5-6 m/s in the shell.

Since both leakage and rupture failure rates are given for this material, calculation of the relation factor is possible. This calculation is intended to comment on the assumed 10% relation factor. The ratio between the given leakage and rupture failure rates is $\{2.1\text{e-}08/5.7\text{e-}12\}$ 5090 or approximately 5000. Tatone¹⁴ states that 0.5% of the steam generator tubes that are plugged actually have leaks. He suggests that tubes may be plugged when nondestructive testing indicate that the tubes have a high probability of leakage. Using the

0.5% factor on the leakage to rupture failure rate ratio yields $\{5000 \times 0.005\}$ 25. Thus $\{25/5000\}$ or 4% of all leaks are ruptures. This calculation shows that using a 10% factor to relate leakage failure rates to rupture failure rates is fairly conservative.

Operating experience failure rates for Inconel 600 with liquid metals or helium was not found during the review of published literature. Therefore, until material testing provides better values, the leakage and rupture failure rates for water coolant are applied to helium, liquid sodium, and liquid lithium coolants.

5.8 Niobium

Operating experience failure rate data for niobium tubing were not found in the review of published literature. Quasi failure rates for niobium are generated using operating experience failure rate data for zirconium.

5.9 Type 316 Stainless Steel

Failure rate sources for Type 316 include tubing used in fission power plant steam generators, heat exchangers, and condensers. Steam generator tubing is chosen to provide the base failure rate since this tubing is believed more similar to ITER in-vessel tubing: thin walled, close packed, and exposed to radiation fields. Caution must be exercised when using steam generator tubing to calculate failure rates for ITER since steam generator tubes are subjected to additional failure modes by being exposed to water and water/steam on both sides of the tubes.

Based on steam generator tubing studies by Tatone¹⁴, Type 316 tubing has a leakage failure rate of $1.0\text{e-}08 \text{ (m-hr)}^{-1}$. This failure rate represent tubes with a length of 20 meters and average plant availability of 70%. Tatone provides the number of steam generator tubes and system operating time so an error factor of 3 is associated with the failure rate.

In his analysis, Tatone discovered that the leakage failure rate was composed of two failure mechanisms: stress corrosion cracking, 70%; and phosphate wastage, 30%. Since these two failure mechanisms are not anticipated in the vacuum environment of ITER, the failure rate is reduced by a factor of 10. This reduction yields a new leakage failure rate of $1.0\text{e-}09 \text{ (m-hr)}^{-1}$. Similar to above, an error factor of 3 is associated with the failure rate.

The rupture failure rate for Type 316 tubing is calculated using the 10% factor between leakage and rupture failure rates. Using this factor and the tubing leakage failure rate yields a rupture failure rate of $\{[1.0\text{e-}09 \text{ (m-hr)}^{-1}]/10\}$ $1.0\text{e-}10 \text{ (m-hr)}^{-1}$. Similar to the leakage failure rate, the error factor associated with the rupture failure rate is 3.

Operating experience failure rates for Type 316 with liquid metals or helium was not found during the review of published literature. Therefore, until material testing provides better values, the leakage and rupture failure rates for water coolant are applied to helium, liquid sodium, and liquid lithium coolants.

5.10 Titanium

Failure rates for titanium tubing are provided by operating experiences of titanium tubed heat exchangers in coal-fired power plants.^{17,18,19} The number of titanium tubes in the reference heat exchanger was 24,200. Each tube has an outer diameter of 22.2e-3 meters and a length of 11 meters. The operating environment for the heat exchanger featured a water pressure of 0.8 MPa, steam pressure of 0.007 kPa, and water velocity of 2 m/s. The heat exchanger also had an availability of 70% during an operating period of 30 years. The number of leaks during the operating period was three.

The leakage failure rate can be calculated by using its definition. Inserting all of the variables yields a leakage failure rate of $\{[3 \text{ leaks}] / [(11 \text{ m}) \times (24,200 \text{ tubes}) \times (0.7 \text{ availability}) \times (365 \text{ days per year}) \times (24 \text{ hour per day}) \times (30 \text{ years})]\}$ 6.1e-11 (m-hr)⁻¹. An error factor of 10 is associated with the failure rate due to the translation from a coal-fired environment to a fusion environment.

The rupture failure rate for titanium tubing is calculated using the 10% ratio between leakage and rupture failure rates. Using this factor and the tubing leakage failure rate yields a rupture failure rate of $\{[6.1\text{e-}11 \text{ (m-hr)}^{-1}] / 10\}$ 6.1e-12 (m-hr)⁻¹. Similar to the leakage failure rate, the error factor associated with the rupture failure rate is 10.

Operating experience failure rates for titanium with liquid metals or helium was not found during the review of published literature. Therefore, until material testing provides better values, the leakage and rupture failure rates for water coolant are applied to helium, liquid sodium, and liquid lithium coolants.

5.11 Vanadium

Operating experience failure rate data for vanadium tubing were not found in the review of published literature. Quasi failure rates for vanadium are generated using operating experience failure rate data for titanium.

5.12 Zirconium

Zirconium's performance as a tubing material in a water coolant system is extrapolated from its performance as fuel rod cladding in LWRs. Fuel rods in LWRs typically have a length of 3.7 meters, diameter of 10.7e-3 meters, and a wall thickness of 0.6e-3 meters. The fuel rods are subjected to an environment that has water velocities of 2-3 m/s, radiation levels of 3 MRad/hr, system pressures of 6 to 7 MPa, and temperatures of 400 °C.¹⁶

Operating experience in over 5 million General Electric BWRs give fuel rod cladding a reliability of 99.996% per rod.²⁰ Assuming a 70% plant availability and one reactor year operating time, the leakage failure rate is calculated as $\{[1 - .99996] / [(365 \text{ days per year}) \times (1 \text{ year}) \times (24 \text{ hours per day}) \times (0.7 \text{ availability}) \times (3.7 \text{ meters per rod}) \times (1 \text{ rod})]\}$ 1.8e-9

$(\text{m-hr})^{-1}$. An error factor of three is assigned to the failure rate since there is a high level of confidence associated with these data.

The rupture failure rate for zirconium tubing is calculated using the 10% ratio between leakage and rupture failure rates. Using this factor and the tubing leakage failure rate yields a rupture failure rate of $\{[1.8\text{e-}9 (\text{m-hr})^{-1}]/10\}$ $1.8\text{e-}10 (\text{m-hr})^{-1}$. Similar to the leakage failure rate, the error factor associated with the rupture failure rate is 3.

The Fermi-1 sodium-cooled reactor used zirconium fuel rods. The fuel rods had an outer diameter of $4\text{e-}3$ meters, a length of 0.8 meters and a total number per core loading of 13,000.¹⁶ The cladding on the fuel rods was $1.27\text{e-}3$ meters. The operating environment of Fermi-1 subjected the zirconium cladding to wall temperatures of 430°C , system pressure of 0.8 MPa, coolant velocities of 2 to 3 m/s, and radiation levels of 10 kRad/hr.¹⁸

Fermi-1 fuel rods did not experience any leakage failures during the operating lifetime of the reactor so the Chi-square distribution is used with 2 degrees of freedom to estimate a leakage failure rate. Assuming plant availability of 50% and an one year operating period yields a leakage failure rate of $\{[1.39][(2) \times (13,000 \text{ tubes}) \times (0.8 \text{ m length}) \times (0.5 \text{ availability}) \times (365 \text{ days per year}) \times (1 \text{ year}) \times (24 \text{ hours per day})]\}$ $1.5\text{e-}8 (\text{m-hr})^{-1}$. The error factor associated with the failure rate is 3 since these data are exact.

The rupture failure rate for zirconium tubing in liquid sodium is calculated using the 10% factor between leakage and rupture failure rates. Using this factor and the tubing leakage failure rate yields a rupture failure rate of $\{[1.5\text{e-}8 (\text{m-hr})^{-1}]/10\}$ $1.5\text{e-}9 (\text{m-hr})^{-1}$. Similar to the leakage failure rate, the error factor assigned to the rupture failure rate is 3.

5.13 Other Tubing Issues

This section presents data researched for tubing issues applicable to all material types.

5.13.1 Double-Walled Tubing

The BCSS²¹ provides general guidance for failure rate calculations with double walled tubes. The study suggests that the external leakage reliability of double walled tubes can be calculated by the formula:

$$\text{Reliability} = 1 - (\text{number of failed tubes in a time period}/\text{number of tubes in service})^2.$$

This reliability formula is essentially the square of the single-walled tubing failure rate.

The BCSS' reliability formula does not include common mode failures, such as radiation damage, environmental effects, material compatibility, and other factors common to adjacent tubes. These common modes must be included in a reliability analysis since their exclusion leads to unreasonably low failure rates.

Piping reliability work by the Environmental Protection Agency²² (EPA) for waste piping used a combination of expert opinion and operating experience to develop a multiplication factor of 0.5 to calculate the reliability of double-walled piping. The 0.5 factor is applied to the failure rate of single walled piping to generate failure rates for double walled piping. The EPA's double-walled multiplication factor does include the common mode effects of vibration, annulus and outer wall corrosion, and material compatibility.

For ITER double walled tubing, neither of the above methods appears directly applicable since the waste piping's factor seems too high and the BCSS formula does not include common mode effects. Consideration of ITER's rigorous in-vessel operating environment inclines one to chose the waste piping multiplication factor of 0.5. However, this factor includes issues such as buried pipe corrosion, corrosive waste fluids, and physical impacts that are specific to waste piping. As a solution, the authors suggest a factor of 0.1 on single walled tubing failure rates to estimate the failure rates for double walled tubing. The 0.1 factor is set by conservative engineering judgment based on reductions from the 0.5 factor.

The 0.1 value is believed to be conservative because it considered the difference between duplex and single walled ferritic steel tubing. The development of the 0.1 factor is as follows. A leakage failure rate for ferritic steel duplex tubing in a liquid metal coolant was given earlier in this chapter as $6.8\text{e-}11 \text{ (m-hr)}^{-1}$. A leakage rate for ferritic steel single wall tubing in a liquid metal coolant was also given as $1.35\text{e-}10 \text{ (m-hr)}^{-1}$. Dividing the duplex tubing failure rate value by the single walled failure rate yields a factor of $\{(6.8\text{e-}11)/(1.35\text{e-}10)\} 0.005$. The 0.5 factor suggested by the waste piping study is two orders of magnitude greater the 0.005 value. As a conservative approach, the authors chose a factor that was one order of magnitude greater than the 0.005 and then rounded up to the nearest tenth. This yields the multiplication factor of 0.1. This factor is believed to be acceptable until further study is conducted or new failure rate data becomes available.

5.13.2 Welds

In-vessel tubing welds are given failure rates in work performed by Buende and et al.²³ Table 5-1 summarizes his calculations. The failure rates in the table are generally applicable to most welded material.

Buende also provides guidance on scaling weld leakage failure rates. These results are given in Table 5-2. The provided multiplication factors are used to tailor basic weld failure rates into failure rates that reflect given operating conditions.

5.13.3 Tubes Plugging Closed

Besides the failure modes of leakage and rupture, tubing can potentially fail because of plugging. Tube plugging often occurs as a result of corrosion product growth in the tubing or excessive impurities in coolant stream.

Table 5-1
Failure Rates for Weld Types

Weld Type	Average Failure Rate (m-hr)⁻¹	Maximum Failure Rate (m-hr)⁻¹
Longitudinal	3E-07	3E-06
Pipe Butt	3E-08	1E-07
Pipe Bend	1E-08 (bend-hr) ⁻¹	4E-06 (bend-hr) ⁻¹
Circumferential	3E-09	1E-08

Table 5-2
Multiplication Factors for Weld Failure Rates

Desired Failure Rate	Multiplication Factor	Base Failure Rate
large leak	0.1	small leak
rupture	0.01	small leak
high stress loaded mat'l	10	low stress loaded mat'l
high quality control	0.1	next lower quality control procedure for weld leakage
maintenance weld	10	shop weld
field weld	3	shop weld

It is expected that the ITER design and maintenance teams will insure cleanliness of the coolant channels during reactor operation since this maintains heat transfer efficiency and limits neutron activation within the coolant system. Accordingly, tube plugging rates are expected to be low. Guidance from the Reactor Safety Study²⁴ indicates that plugging in small diameter pipes (less than 76.2 mm diameter) is as frequent as pipe ruptures, so the tubing rupture failure rates presented in this chapter can be used for the plugging failure mode until more detailed analyses are performed.

5.13.4 Multiple Tube Leaks

The issue of multiple tubes simultaneously failing is presented here due to related concerns for steam generator tubing in the fission power industry. Bloch²⁵ has calculated that the probability of a second tube failing within an hour of an adjacent tube's rupture is approximately 1e-04.

It should be noted that while the above failure rate does include the influences of common tubing failure mechanisms, it does not include the phenomena of pipe whip or coolant jetting. For ITER, the analyses for these phenomenons must be performed separately. More specifically, the American National Standards Institute (ANSI) requires that all high energy piping (water temperature greater than 93 °C or coolant pressure greater than 1.93 MPa) have pipe whip analyses performed.²⁶

5.14 Chapter Summary

The preceding sections presented leakage and failure rates obtained from the review of published literature for the proposed tubing materials. Information on the failure rates' operating environment was provided to give background on the failure rates.

The Tables 5-3 through 5-6 present the operating experience failure rates and their associated error factors. This information will be used in subsequent chapters to generate tubing failure rates for ITER.

Table 5-3
Operating Experience Failure Rates for Helium Coolant Type

Helium Coolant	Leakage (m-h)⁻¹	Rupture (m-h)⁻¹	Error Factor
Beryllium	3.0e-06	3.0e-07	10
Copper	4.9e-08	4.9e-09	3
Ferritic Steel	4.0e-10	4.0e-11	10
Ferritic Steel, Duplex	4.0e-10	4.0e-11	10
Inconel 600	2.1e-08*	5.7e-12	10
Niobium (quasi)	1.8e-09	1.8e-10	10
Stainless Steel 316	1.0e-09	1.0e-10	3
Titanium	6.1e-11	6.1e-12	10
Vanadium (quasi)	6.1e-11	6.1e-12	10
Zirconium	1.8e-09	1.8e-10	3

*Value may be inflated due to inclusion of incipient leaks.

Table 5-4
Operating Experience Failure Rates for Liquid Lithium Coolant Type

Lithium Coolant	Leakage (m-h)⁻¹	Rupture (m-h)⁻¹	Error Factor
Beryllium	3.0e-06	3.0e-07	10
Copper	4.9e-08	4.9e-09	3
Ferritic Steel	4.0e-10	4.0e-11	10
Ferritic Steel, Duplex	4.0e-10	4.0e-11	10
Inconel 600	2.1e-08*	5.7e-12	10
Niobium (quasi)	1.8e-09	1.8e-10	10
Stainless Steel 316	1.0e-09	1.0e-10	3
Titanium	6.1e-11	6.1e-12	10
Vanadium (quasi)	6.1e-11	6.1e-12	10
Zirconium	1.8e-09	1.8e-10	3

*Value may be inflated due to inclusion of incipient leaks.

Table 5-5
Operating Experience Failure Rates for Liquid Sodium Coolant Type

Sodium Coolant	Leakage (m-h)⁻¹	Rupture (m-h)⁻¹	Error Factor
Beryllium	3.0e-06	3.0e-07	10
Copper	4.9e-08	4.9e-09	3
Ferritic Steel	1.4e-10	1.4e-11	10
Ferritic Steel, Duplex	6.8e-11	6.8e-12	10
Inconel 600	2.1e-08*	5.7e-12	10
Niobium (quasi)	1.8e-09	1.8e-10	10
Stainless Steel 316	1.0e-09	1.0e-10	3
Titanium	6.1e-11	6.1e-12	10
Vanadium (quasi)	6.1e-11	6.1e-12	10
Zirconium	1.8e-09	1.8e-10	3

*Value may be inflated due to inclusion of incipient leaks.

Table 5-6
Operating Experience Failure Rates for Water Coolant Type

Water Coolant	Leakage (m-h)⁻¹	Rupture (m-h)⁻¹	Error Factor
Beryllium	3.0e-06	3.0e-07	10
Copper	4.9e-08	4.9e-09	3
Ferritic Steel	4.0e-10	4.0e-11	10
Ferritic Steel, Duplex	4.0e-10	4.0e-11	10
Inconel 600	2.1e-08*	5.7e-12	10
Niobium (quasi)	1.8e-09	1.8e-10	10
Stainless Steel 316	1.0e-09	1.0e-10	3
Titanium	6.1e-11	6.1e-12	10
Vanadium (quasi)	6.1e-11	6.1e-12	10
Zirconium	1.8e-09	1.8e-10	3

*Value may be inflated due to inclusion of incipient leaks.

Chapter References

1. K. D. Dietz and et al., "Experience With High Heat Flux Components In Large Tokamaks," Fusion Engineering and Design, 16, 1991, pages 229-251.
2. B. J. Merrill and S. C. Jardin, "Coolant Ingress Induced Disruption Calculations for ITER," Fusion Technology, 19, 1991, pages 1278-1283.
3. H. M. Thomas, "Pipe and Pressure Vessel Failure Probability," Reliability Engineering, 2, 1981, pages 83-124.
4. H. John, H. Schnauder, E. Bogusch, J. Wehling, "Reliability Investigations and Improvements of the Cooling System of a Self-Cooled Liquid Metal Breeder Blanket," Fusion Technology 1992. Proceedings of the 17th Symposium on Fusion Technology, Rome Italy, September 14-18, 1992, Elsevier Science Publishers, 1993, pages 1399-1403.
5. J. M. Beeston et al., "Surveillance Testing and Property Evaluation of Beryllium in Test Reactors," Nuclear Metallurgy, 19, August 1973, Proceedings of the Symposium on Materials Performance in Operating Nuclear Systems, CONF-730801, pages 59-87.
6. M. M. Carroll, "Polyvinyl Chloride (PVC) Pipe Reliability and Failure Modes," Reliability Engineering, 13, 1985, pages 11-21.
7. Guidelines for Process Equipment Reliability Data. With Data Tables, Center for Chemical Process Safety, New York, American Society of Chemical Engineers, 1989, page 186.
8. J. B. Tobias, "Brazing of Large Section Water-Cooled Copper Conductor of TFTR," Proceedings of the 8th Symposium on Engineering Problems of Fusion Research, San Francisco, California, November 13-16, 1979, pages 117-123.
9. Final Safety Analysis Report, Tokamak Fusion Test Reactor, Amendment 2, DTSD-FSAR-17, Princeton Plasma Physics Laboratory, April 22, 1993, Chapter 4, Section 4.1.3.6.3; "TF Coil Cooling-Water Leaks," and Chapter 5.
10. Manual on Reliability Data Collection for Research Reactor PSAs, IAEA-TECDOC-636, International Atomic Energy Agency, Vienna, January 1992, page 32.
11. L. C. Cadwallader, "Investigation of Selected Accident Frequencies for Helium, Water, and Liquid Metal ITER Coolants," International Thermonuclear Experimental Reactor Engineering Design Safety task, ITER/US/93/EN/SA-3, January 27, 1993.
12. F. Boni and P. S. Otten, "Fermi and Hallam Steam Generators," Nucleonics, 17, June 1961, pages 58-61.

13. D. Mosey, "Sudden Pressure Tube Failure in Pickering Nuclear Generating Station Unit 2," Nuclear Safety, 26, 1985, pages 349-358.
14. O. S. Tatone and R. L. Tapping, "Steam Generator Tube Performance Experience with Water-Cooled Nuclear Power Reactors During 1985," Nuclear Safety, 30, 1989, pages 382-399.
15. J. P. Adams and M. B. Sattison, "Frequency and Consequences Associated with a Steam Generator Tube Rupture Event," Nuclear Technology, 90, May 1990, pages 168-185.
16. J. Sejvar, "Normal Operating Radiation Levels in Pressurized Water Reactor Plants," Nuclear Technology, 36, 1977, pages 48-55.
17. M. M. El-Wakil, Powerplant Technology, McGraw-Hill, New York, 1984.
18. M. M. El-Wakil, Nuclear Energy Conversion, American Nuclear Society, La Grange Park, Illinois, 1978, pages 313-320.
19. S. A. Kurtz, R. L. Ward, and D. J. Schumerth, "A Titanium Tubed Modular Condenser Changeout at the Ravenswood Generating Station," PWR-Vol. 12, Performance Monitoring and Replacement of Heat Exchanger Components and Materials, American Society of Mechanical Engineers, 1990, pages 35-43.
20. W. E. Bailey et al., "Recent GE BWR Fuel Experience," Proceedings of the International Topical Meeting on LWR Fuel Performance, Fuel for the 90's, sponsored by the American Nuclear Society, Avignon, France, April 21-24, 1991, pages 26-35.
21. D. L. Smith et al., Blanket Comparison and Selection Study. Final Report, ANL/FPP-84-1, Argonne National Laboratory, September 1984, pages 9-5 to 9-7.
22. Hazardous Waste Tank Failure (HWTF) and Release Model: Description of Methodology, Appendices A, B, C, D, and E, EPA/530/SW-86/012, sponsored by the US Environmental Protection Agency, performed by Pope-Reid Associates, Inc. under report number PB-86-192945, January 1986.
23. R. Buende, S. Fabritsiev, and V. Rybin, "Reliability of welds and brazed joints in blankets and its influence on availability," Fusion Engineering and Design, 16, 1991, pages 59-72.
24. N. C. Rasmussen et al., Reactor Safety Study. An Assessment of Accident Risks in US Commercial Nuclear Power Plants, WASH-1400, NUREG/75-014, US Nuclear Regulatory Commission, August 1974, Appendix III, Failure Rate Data.
25. M. Bloch et al., "Probabilistic Risk Assessment Related to Multiple Ruptures of Steam Generator Tubes," Nuclear Technology, 84, 1989.

26. Design basis for protection of light water nuclear power plants against the effects of postulated pipe rupture, ANSI/ANS-58.2-1988, American National Standards Institute and the American Nuclear Society, October 6, 1988.

6.0 RELIABILITY PREDICTION - K FACTORS

6.1 Introduction

Reliability prediction is the process of estimating the reliability of an engineering system during the design phase. Reliability prediction can also be performed during the system's operational phase, before a significant amount of operating time has been accumulated. The reliability prediction technique used by the authors is based on the Thomas¹ method.

The Thomas method applies weighing factors to failure rates from an operating system to generate failure rates for a system without operating experience. Other, more general methods use K-factor multipliers on the failure rates. Usually, the system without operating experience is still in the design phase.

K-factors are intended to account for the small differences between the operating system and the system without operating experience. These differences can be variations in component parameters and/or system operating environment. The smaller the variations, the more accurate the K-factors model the system without operating experience. Similarly, large variations between the two systems yield large uncertainty levels with the extrapolated failure rates.

Most of the failure rate data used in this report are from the nuclear industry so the data are somewhat applicable to the fusion environment. However, the fusion reactor's operating environment is vastly different from that of the fission reactor. Thomas' method allows the development of conservative K-factors to help reduce the uncertainty levels associated with using non-fusion operating experience failure rates for ITER.

The sections in this chapter discuss the derivation and application of K-factors for the generation of ITER in-vessel tubing failure rates. These K-factors are applied to operating experience failure rates in the next chapter to generate failure rates for ITER.

6.2 K Factor Multipliers

As with most engineering models, reliability prediction involves engineering judgment. These judgments are required in several areas of determining component operating parameters. The following equation identifies the operating parameters required for component reliability prediction:²

$$\text{Generated Failure Rate} = (\text{Generic Failure Rate}) \times K1 \times K2 \times K3 \times K4 \times K5 \times K6$$

where the generic, unmodified failure rate is characteristically found in a handbook, a compilation of reliability data, or in the literature on equipment operating experiences. In this report, the generic failure rates are the failure rate data presented in the Data Compilation chapter.

The generic failure rate is generally considered unstressed by special factors, that is, operating within its design parameters. This assumption implies that incurred failures are from causes inherent in system operation. These causes are termed random events. Each of the six K factors in the above equation represents a random event. Below are definitions for the random events.

- K1 The applied stress that challenges the design envelope. Generally, K1 is defined as electrical voltages or thermal stresses.
- K2 The ratio of minor, or tolerable failures, to catastrophic failures.
- K3 The environmental stress factor. K3 models system-specific failure mechanisms.
- K4 The maintenance factor. K4 models the anticipated quality of maintenance procedures, inspections, and/or tests. K4 also includes expected maintenance difficulties associated with highly toxic, irradiated materials.
- K5 The system complexity factor. K5 is based on the system's geometry and architecture. K5 also models interaction(s) between multiple systems.
- K6 The cycling effects factor.

The individual K factors often have values between 0.1 and 3.0. However, this range is highly dependent on the component type, generic data source, and the K factor category.

6.3 K Factors

Failure rates for ITER in-vessel tubing are generated using the equation:

$$\text{Generated Failure Rate} = (\text{Generic Failure Rate}) \times K1 \times K2 \times K3 \times K4 \times K5 \times K6$$

K3 models the failure mechanisms developed and discussed throughout this report. K3 will be given specific values for the materials in the next section. This section gives values to the other K factors in the equation. These factors will then be used with the K3 values in the following chapter to generate in-vessel tubing failure rates.

K1. The authors assume that the ITER design team will use components that are compatible with the rigorous and demanding operating environment within ITER. It is further assumed that the first wall and blanket will be replaced at scheduled intervals. This assumption implies that the components have a preplanned service life. The two assumptions suggest that abnormal stress loads are not anticipated for the tubing. Therefore, K1 is given a value of 1.0.

K2. K2 is used to identify failures of a particular type. Since the failure rates calculated here are failure mode specific (i.e. leakage or rupture), this factor is not needed. Therefore, K2 is given a value of 1.0.

K4. Many factors must be considered to develop a K4 value. Such factors include the issues of remote versus hands-on maintenance and the techniques of testing and inspection.

Remote maintenance may be required by the high levels of radioactivity associated with the piping material and the in-vessel environment. While remote maintenance does protect maintenance personnel, it limits inspection procedures. Often, maintenance personnel have more diagnostic tools available to them during "hands on" inspection and such inspections may be more informative.

Testing and inspection issues are also addressed with this factor. Piping and tubing can be pressure tested and during operation may be viewed with television cameras for visual inspection. Both of these factors allow a reduction in the K4 value. For the High Flux Isotope Reactor (HFIR) risk assessment, a failure rate reduction factor of 0.05 was used for reactor piping that was 200% pressure tested annually and experienced little thermal cycling because of its low operating temperature.³

At the time of this report, little information is available on the maintenance, testing, and inspection procedures for ITER. However, the authors previously made the assumption that the ITER design team will pursue the highest quality and performance in every aspect of ITER design and operation. Therefore, K4 is given an optimistically low value of 1.5.

K5. K5 models complicated electronic equipment such as radar units and satellite link navigation systems. In this report, ITER in-vessel tubing is not considered to have design complexities of the same order of magnitude as the aforementioned electronic equipment. Therefore, K5 is given a value 1.0.

K6. K6 models on-off cyclic behavior. The authors define cyclic fatigue as a tubing failure mechanism so it is included under the K3 factor. Thus, K6 is given a value of 1.0.

The above discussion has assigned values to all of the K factors except K3, which is discussed in the following section. Table 6-1 presents the values given to the K factors.

Table 6-1
In-Vessel Tubing K Factors

K1	K2	K4	K5	K6
1.0	1.0	1.5	1.0	1.0

6.4 K3 Factors

The Material Analysis chapter categorized the failure mechanism resistance levels for each of the tubing materials. The results of the categorization are presented in Table 6-2. These resistance levels will be used to determine K3 factors for the tubing materials. First, a consideration of ITER's operating environment is necessary.

The failure rates presented in the Data Compilation chapter are generally from operating environments that are more benign than the one anticipated for ITER. ITER's operating environment for in-vessel tubing features moderate wall loadings ($\leq 0.5 \text{ MW/m}^2$), elevated temperatures ($\leq 300 \text{ }^\circ\text{C}$ for water coolant and $400 \text{ }^\circ\text{C}$ for other coolants), large neutron fluxes ($\leq 2.2 \times 10^{14} \text{ n/cm}^2\text{-s}$, $E_n > 1.0 \text{ MeV}$), and high system pressures ($\leq 6 \text{ MPa}$, for water coolant type). Few, if any, of the compiled failure rates are for components whose operating environment is similar to those specifications. Thus, it is reasonable to assume that a component's reliability, with all other variables held constant, will decrease in an ITER-like environment.

The above assumption implies that even high-grade fission components may have lower reliability values in ITER. This consequence may result from two major factors: cyclic nature and helium-to-dpa ratio of tokamaks. ITER, by design, is a pulsed machine and this will introduce additional fatigue factors and stresses that are not present in the steady state environment of a fission reactor. The higher helium-to-dpa ratio suggests that fusion components may be more susceptible to helium-induced swelling than their fission counterparts. Considering these two factors, the best reliability value for the tubing materials in ITER may be less than the corresponding value in the original operating environment. This limitation is modeled by setting the best resistance level from Table 6-2 to a value greater than unity. This value also defines the upper limit of the K factor scale.

It can be argued that setting the best resistance level from Table 6-2 to a value greater than unity may be overly conservative since many fission components are designed with a conservative safety margin. However, the authors feel that since ITER is not yet fully characterized and a complete knowledge of tubing performance in a fusion machine is unknown, a conservative approach is necessary to extrapolate known failure rates to ITER conditions. It is further the authors' intentions that these extrapolations can be updated as more information becomes available.

The upper and lower limits of the K factor scale are defined as 1.1 and 3.0, respectively. The limits are set by using the variance information ($0.1 < K < 3.0$). The lower limit, 1.1, states that a failure mechanism reduces non-fusion tubing reliability by 10% due to ITER's rigorous in-vessel operating environment, as explained in the two paragraphs above. The 10% factor is chosen as an approximation to include the makeup of the tubing material operating experience failure rate data in Chapter 5.0. For tubes in condensers and heat exchangers, the 10% factor should be a good representation of the decrease in tubing reliability. For steam generator tubes and fuel rods, the 10% factor may be a more conservative approximation. However, the failure rates generated in this report are

intended to be order of magnitude values for design trade-off studies and preliminary ITER safety work. Accordingly initial approximations may be conservative until further information becomes available.

The 3.0 upper limit says that a failure mechanism activity reduces non-fusion tubing material reliability by 300% when the original operating environment is compared with ITER's. The 3.0 upper limit is chosen with the assumption that the ITER design team will not use materials whose performance is reduced by a margin greater than 300%.

The K factor scale requires further division to better define the K3 factor. The scale is presented in Table 6-3. In Table 6-3, the resistance levels in Table 6-2 are subdivided and given individual K factors. The following paragraphs describe the K factors in Table 6-3.

The "poor" level of failure resistance is assigned two possible K values, 1.5 and 3.0. The 3.0 value is our lower limit and is used for failure mechanisms that severely and immediately attack tubing material upon exposure. The 1.5 value is used for failure mechanisms that severely attack tubing materials when given the right operating conditions. These conditions usually are high impurity concentrations in the coolant stream or elevated temperatures ($T > 500\text{ }^{\circ}\text{C}$).

The "acceptable" level of failure resistance also has two possible K values, 1.3 and 1.1. The 1.1 value is used when failure data were not found for the failure mechanism or available data indicated that the failure mechanism is not a significant factor. The 1.3 value is given to failure mechanisms whose attack is reported in the failure data as being slightly worse than average. The value is also used for failure mechanisms that are without published failure data, but might be more probable in ITER's environment than the original environment.

Applying the methodology described above to the data in Table 6-3 yields the data in Table 6-4. Table 6-4 presents the K values for each of the failure mechanisms and coolant types for the tubing materials. The row in Table 6-4 titled "Total For Failure Mechanisms" is the total influence expected upon the tubing material's failure rates due to the failure mechanisms. This value is the product of the individual failure mechanisms and is one component of the K3 factor, the other being the coolant performance. For example, ITER in-vessel tubing made of Inconel 600 will cause the generic Inconel 600 failure rate to increase by a factor of 4.2 due to the anticipated activity of the tubing failure mechanisms.

Table 6-4 also provides K values for the coolant types. These values are combined with the total K values for the failure mechanisms to generate the K3 factor. Tables 6-5 through 6-8 combine the total K values for the failure mechanisms with the K values for the coolant types.

At this point, all of the K factors have been defined and given values. ITER in-vessel tubing failure rates can now be generated using the failure rate prediction equation given earlier in this chapter. The next chapter presents the results of the failure rate calculations.

Table 6-2
Levels of Failure Resistance for Tubing Materials

	Be	Cu	Inconel	Nb	316 ss	Ti	V	Zr
Failure Mechanism								
Creep	0	-1	0	0	0	0	0	0
Cyclic Fatigue	-1	-1	0	-1	0	0	0	0
Erosion	0	0	0	0	0	0	0	0
Helium Swelling	-1	0	0	+1	-1	0	0	0
Hydrogen Embrittlement	0	-1	0	0	-1	-1	-1	0
Intergranular Attack	0	0	0	0	-1	0	0	0
Irradiation Embrittlement	-1	-1	0	-1	0	0	0	0
Magnetic Fields Interactions	+1	-1	-1	+1	+1	+1	+1	+1
Oxidation	-1	-1	0	-1	-1	-1	-1	-1
Thermal Cycling	0	0	0	0	0	0	0	0
Coolant Type								
Helium Gas	0	0	0	0	0	0	-1	0
Liquid Lithium	0	-1	0	0	0	0	0	0
Liquid Sodium	0	-1	0	+1	0	0	0	-1
Water	0	0	0	0	+1	+1	0	-1

Table 6-3
K Factor Scale for Failure Mechanism Resistance Levels

Failure Resistance	poor		acceptable		excellent
Symbolic Representation	-1		0		+1
K Factor Value	3	1.5	1.3	1.1	1

Table 6-4
K Values for Tubing Materials and Coolant Types

	Be	Cu	Inconel	Nb	316 ss	Ti	V	Zr
Creep	1.1	3.0	1.1	1.1	1.1	1.1	1.1	1.3
Cyclic Fatigue	3.0	1.5	1.1	1.5	1.1	1.1	1.1	1.1
Erosion	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1
Helium Swelling	3.0	1.1	1.3	1.0	1.5	1.3	1.1	1.1
Hydrogen Embrittlement	1.1	1.5	1.1	1.3	1.5	3.0	3.0	1.1
Intergranular Attack	1.1	1.3	1.1	1.1	1.5	1.1	1.1	1.1
Irradiation Embrittlement	3.0	1.5	1.1	1.5	1.1	1.1	1.1	1.1
Magnetic Fields Interactions	1.0	3.0	1.5	1.0	1.0	1.0	1.0	1.0
Oxidation	3.0	1.5	1.1	1.5	1.5	1.5	1.5	1.5
Thermal Cycling	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1
Total for Failure Mechanisms	130.5	78.8	4.2	6.4	8.2	10.4	8.8	3.8
Coolant Type								
Helium Gas	1.1	1.3	1.1	1.3	1.1	1.1	1.5	1.1
Liquid Lithium	1.1	3.0	1.3	1.3	1.1	1.1	1.1	1.1
Liquid Sodium	1.1	3.0	1.1	1.1	1.1	1.1	1.3	1.5
Water	1.1	1.1	1.1	1.3	1.1	1.1	1.3	1.1

Table 6-5
K3 Factors for Tubing Materials and Water Coolant Type

Coolant	Failure Mechanism K Values	Coolant K Value	K3 Factor
Beryllium	130.5	1.1	144
Copper	78.8	1.1	87
Inconel 600	4.2	1.1	5
Niobium	6.4	1.3	8
316 S.S.	8.2	1.1	9
Titanium	10.4	1.1	11
Vanadium	8.8	1.3	11
Zirconium	3.8	1.1	4

Table 6-6
K3 Factors for Tubing Materials and Helium Coolant Type

Coolant	Failure Mechanism K Values	Coolant K Value	K3 Factor
Beryllium	130.5	1.1	144
Copper	78.8	1.3	102
Inconel 600	4.2	1.1	5
Niobium	6.4	1.3	8
316 S.S.	8.2	1.1	9
Titanium	10.4	1.1	11
Vanadium	8.8	1.5	13
Zirconium	3.8	1.1	4

Table 6-7
K3 Factors for Tubing Materials and Liquid Lithium Coolant Type

Coolant	Failure Mechanism K Values	Coolant K Value	K3 Factor
Beryllium	130.5	1.1	144
Copper	78.8	3	236
Inconel 600	4.2	1.3	5
Niobium	6.4	1.3	8
316 S.S.	8.2	1.1	9
Titanium	10.4	1.1	11
Vanadium	8.8	1.1	10.
Zirconium	3.8	1.1	4

Table 6-8
K3 Factors for Tubing Materials and Liquid Sodium Coolant Type

Coolant	Failure Mechanism K Values	Coolant K Value	K3 Factor
Beryllium	130.5	1.1	144
Copper	78.8	3	236
Inconel 600	4.2	1.1	5
Niobium	6.4	1.3	8
316 S.S.	8.2	1.1	9
Titanium	10.4	1.1	11
Vanadium	8.8	1.1	10.
Zirconium	3.8	1.5	6

Chapter References

1. Thomas, H. M., "Pipe and Pressure Vessel Failure Probability," Reliability Engineering, 2, 1981, pages 83-124.
2. Von Alven W. H. (editor), Reliability Engineering, Prentice-Hall, NJ, 1964, pages 311-318.
3. Johnson, D. H. and et al., The High Flux Isotope Reactor Probabilistic Risk Assessment, PLG-0604, Pickard, Lowe & Garrick, Inc., Newport Beach, CA, January 1988, Appendix C.

7.0 FAILURE RATES FOR ITER IN-VESSEL TUBING

7.1 Introduction

This chapter uses the material analyses, generic failure rate data, and reliability prediction equation from previous chapters to generate failure rates for ITER in-vessel tubing. Failure rates are generated for each of the proposed tubing materials in combination with the coolant types.

7.2 Failure Rate Tables

Tables 1 through 3 present the generated ITER in-vessel tubing failure rates. The tables identify the tubing material, generated failure rate, and associated error factor.

Table 7-1
Failure Rates for Helium Coolant System

Tubing Material	Leakage (m-h) ⁻¹	Rupture (m-h) ⁻¹	Error Factor
Beryllium	6.5e-04	6.5e-05	10
Copper	7.5e-06	7.5e-07	3
Ferritic Steel	5.4e-09	5.4e-10	10
Ferritic Duplex Steel	9.2e-10	9.2e-11	10
Inconel 600	1.5e-07	4.0e-11	10
Niobium (quasi)	2.2e-08	2.2e-09	10
316 ss	1.4e-08	1.4e-09	3
Titanium	1.0e-09	1.0e-10	10
Vanadium (quasi)	1.2e-09	1.2e-10	10
Zirconium	1.1e-08	1.1e-09	3

Table 7-2
Failure Rates for Liquid Lithium Coolant System

Tubing Material	Leakage (m-h)⁻¹	Rupture (m-h)⁻¹	Error Factor
Beryllium	6.5e-04	6.5e-05	10
Copper	n/a	n/a	n/a
Ferritic Steel	1.9e-09	1.9e-10	10
Ferritic Duplex Steel	9.2e-10	9.2e-11	10
Inconel 600	1.7e-07	4.7e-11	10
Niobium (quasi)	2.2e-08	2.2e-09	10
316 ss	1.4e-08	1.4e-09	3
Titanium	1.0e-09	1.0e-10	10
Vanadium (quasi)	8.9e-10	8.9e-11	10
Zirconium	1.1e-08	1.1e-09	3

Table 7-3
Failure Rates for Liquid Sodium Coolant System

Tubing Material	Leakage (m-h)⁻¹	Rupture (m-h)⁻¹	Error Factor
Beryllium	6.5e-04	6.5e-05	10
Copper	n/a	n/a	n/a
Ferritic Steel	1.9e-09	1.9e-10	10
Ferritic Duplex Steel	9.2e-10	9.2e-11	10
Inconel 600	1.5e-07	4.0e-11	10
Niobium (quasi)	1.9e-08	1.9e-09	10
316 ss	1.4e-08	1.4e-09	3
Titanium	1.0e-09	1.0e-10	10
Vanadium (quasi)	1.0e-09	1.0e-10	10
Zirconium	1.5e-08	1.5e-09	3

Table 7-4
Failure Rates for Water Coolant System

Tubing Material	Leakage (m-h)⁻¹	Rupture (m-h)⁻¹	Error Factor
Beryllium	6.5e-04	6.5e-05	10
Copper	6.4e-06	6.4e-07	3
Ferritic Steel	5.4e-09	5.4e-10	10
Ferritic Duplex Steel	9.2e-10	9.2e-11	10
Inconel 600	1.5e-07	4.0e-11	10
Niobium (quasi)	2.2e-08	2.2e-09	10
316 ss	1.2e-08	1.2e-09	3
Titanium	9.5e-10	9.5e-11	10
Vanadium (quasi)	1.0e-09	1.0e-10	10
Zirconium	1.0e-08	1.0e-09	3

7.3 Failure Rate Graphs

Figures 1 through 12 graphically illustrate the generated ITER in-vessel tubing failure rates. Each coolant type has three graphs that represent the tubing materials' failure rates for that system. The leakage and rupture failure rates are plotted on separate graphs and also on a single graph. The single graph is intended to allow comparison of tubing materials' total reliability for the coolant type.

Figure 7-1
Leakage Failure Rates for Helium Coolant

Figure 7-2
Rupture Failure Rates for Helium Coolant

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

2 of 2

Figure 7-3
Leakage vs. Rupture Rates for Helium Coolant

Figure 7-4
Leakage Failure Rates for Liquid Lithium Coolant

Figure 7-5
Rupture Failure Rates for Liquid Lithium Coolant

Figure 7-6
Leakage vs. Rupture Rates for Liquid Lithium Coolant

Figure 7-7
Leakage Failure Rates for Liquid Sodium Coolant

Figure 7-8
Rupture Failure Rates for Liquid Sodium Coolant

Figure 7-9
Leakage vs. Rupture Rates for Liquid Sodium Coolant

Figure 7-10
Leakage Failure Rates for Water Coolant

Figure 7-11
Rupture Failure Rates for Water Coolant

Figure 7-12
Leakage vs. Rupture Rates for Water Coolant

8.0 CONCLUSIONS

The objective of this report was to identify the material and coolant combination(s) that yielded the lowest failure rate for ITER in-vessel coolant tubing. Since a low tubing failure rate implies improved safety and operational reliability, the analyses are inherently identifying the most reliable tubing material from a leak perspective.

The generated failure rates identify the best tubing performer by its expected reliability or failure rate. With an average leakage failure rate of $3.1\text{e-}10$ (m-hr) and rupture failure rate of $3.1\text{e-}11$ (m-hr) for the coolant types of helium, liquid lithium, liquid lithium, and water, titanium is identified as the most reliable tubing material. Titanium tubing does have several issues of concern, such as oxygen and hydrogen uptake, weldability, and fabricability. Thus, while titanium is selected as the best performer, additional research and testing are necessary to verify its candidacy.

The second leading tubing performers (averaged over all four coolant types for both failure modes) are duplex ferritic steels and vanadium. These two materials had average failure rates that were very similar. It should be noted that the failure rates for the two materials are more approximate than for the other tubing materials. Vanadium's failure rates are calculated using operating experience failure rates for titanium and the duplex ferritic steel uses operating experience failure rates for Type 316 austenitic steel. The cross reference of failure rates was necessary due to the lack of published failure rate data for the two materials.

The two worst tubing performers are beryllium and copper. Beryllium's poor performance is due to its tendency to swell and become embrittled. Copper's poor performance is a result of its immediate dissolution in liquid metal coolants. However, the failure rate data used for copper and beryllium may have biased the tubing materials' performance. Failure rates for beryllium are cross-referenced from PVC pipe and rates for copper are from tubing channels in the TFTR magnets. Future research and testing may yield better results for these two materials.

The other tubing materials, ferritic steel, Inconel 600, Type 316 stainless, and niobium had performances that were greatly dependent upon the coolant type. For these materials, good performance with one coolant type was offset by poor performance in another coolant type. Since the criterion for best performance was based on failure rates averaged over all four coolant types, this gave average to low scores to the aforementioned tubing materials.

Finally, the failure rates generated in this paper are suggested for design trade-off studies and preliminary ITER safety work until further material failure testing provides more accurate data.

APPENDIX A

INVESTIGATION OF SELECTED ACCIDENT FREQUENCIES FOR HELIUM, WATER, AND LIQUID METAL ITER COOLANTS

Introduction

Accident frequencies for various coolant options are compared to provide insight and guidance to the designers in the Engineering Design Activity (EDA) of the International Thermonuclear Experimental Reactor (ITER). The coolant choice will have an impact on ITER safety, so an examination of expected reliabilities and accident frequencies supports the design selection process. This investigation is not for regulatory purposes. The intent is to perform a comparison of safety performance for the various candidate ITER coolants.

Accident frequencies for two classic accident events in divertor cooling systems, the loss-of-coolant accident (LOCA) and loss-of-flow accident (LOFA), are examined for water, helium, and liquid metal coolants. Liquid gallium, lithium, and lead are the current choices for a liquid metal, but since there are no reliability data for these liquid metal flow systems, operating experience data for liquid sodium and sodium-potassium systems are used to represent them. Sodium (or another alkali metal, such as sodium-potassium) cooling systems are also considered here. Component failure rates, used to generate accident frequencies, are based on fission reactor operating experiences and fusion experiences whenever possible. Fission-based failure rates are based on pedigreed, N-stamp (that is, designed using American Society of Mechanical Engineers codes or other similar guidance) components. Figure A-1 gives the specific reliability concerns addressed for each of the accidents.

The accident scenario issues for each coolant will be discussed, and a conclusion on which coolant has the lowest accident frequency, from a LOCA and LOFA perspective, is given.

Depending on the choice of coolant, a loss of coolant in the vacuum vessel can cause several undesirable events besides an experiment shut down for tube

Table A-1. Accident concerns to be examined for the various coolants.

<u>Portion of System Being Examined</u>	<u>LOCA Concern</u>	<u>LOFA Concern</u>
In-Vessel cooling (divertor cooling tube network)	cooling tube breach	plugging a tube or tubes
Ex-vessel cooling (exterior piping and other components in the divertor cooling system)	piping fails in a large breach	pump fails valve closes pipe plugging valve plugging

repairs. First, the inflow of coolant would cause a plasma disruption.^{A-1} Second, the coolant might react with hot plasma facing component (PFC) materials. Third, overpressurization is possible, which might lead to a release of activated dust and possibly tritium from the vacuum vessel. These effects should studiously be avoided for the longevity of a future fusion experiment.

A loss of flow event can quickly lead to tube overheating and subsequent breaching by burnout. Due to the pressure difference between the coolant and the vacuum vessel, coolant can discharge into the vacuum vessel, leading to the same results as a LOCA.

Designs under scrutiny

Divertor coolant accidents are considered because the upper and lower divertors will carry the highest heat loads and are therefore the most sensitive to transients. Traditional divertors have a block of some PFC material (graphite or beryllium, etc.) brazed to a substrate plate. The back of the substrate plate is cooled by transferring heat to coolant inside tubes mounted on, or inside, the plate. The substrate and tubes are usually copper to improve heat transfer and to make braze bonds to the PFC material more feasible. For the International Thermonuclear Experimental Reactor (ITER), a new gas divertor design calls for the divertor arrangement described above to be the sides of a 'trench'. This trench is filled with deuterium or another gas that is being vacuum pumped out of the

bottom of the trench. The diverted plasma radiates energy to the traditional PFC material after colliding with the gas particles moving down through the trench. For this investigation, the gas divertor PFC material could be cooled by any of the mentioned coolants. The gas stream out from the divertor trench must also be cooled before readmission to the trench. Any of the aforementioned coolants could probably also serve as a heat transfer fluid for the divertor gas. The PFC material coolant choices will be discussed here when considering divertor LOCAs and LOFAs.

Preliminary system design information is assumed to allow quantification of the accident frequencies. This information might change during the ITER engineering design activity (EDA), but reflects our best understanding of the design as it exists now. In-vessel material is copper tubing (unless there is a compatibility problem), and ex-vessel material is stainless steel piping. Pertinent information is given below:

<u>Coolant</u>	<u>Pressure (MPa)</u>	<u>Tube diameter (mm)</u>	<u>Total tube length (km)</u>
Water	3	10	13
Helium	5	30	6
Gallium	1.5	20	8
Lead	1.5	25	7
Sodium	1.5	20	8
Lithium	1.5	20	8

These values are based on a total of 200 m² of divertor area, and a 5 mm gap between each of the tubes. The water values reflect the ITER conceptual design activity results, while other values have been found from ITER pre-EDA engineering presentations or approximated using the Dittus-Boelter heat transfer correlation^{A-2} for the various coolants. If the divertor is larger, or there is more tubing, then the resulting frequencies increase. If the tubing amount decreases, then the frequency results also decrease.

For the ex-vessel piping, guidance was taken from pressurized water reactors. A four loop fission reactor has about 800 m of large (1 m) diameter piping in its primary coolant system.^{A-3} Since ITER is roughly equivalent in physical size and heat rejected (2.8 GW compared to fission's 3.1 GW) to a fission reactor, the value is rounded up to an even 1 km of piping outside the vacuum vessel to account for upper and lower ring piping. This 1-km pipe length of ex-vessel piping is used for all cooling system choices in this comparison. If ITER is determined to require more piping, then 1 km serves as an easily scaled length for future piping reliability calculations.

The ITER design is meant to ultimately achieve 25% availability, or about 2200 operating hours per year. The 2200 hours/yr value will be used for determining accident frequency values. Accident frequency results are linearly sensitive to these system design values of pipe length and operating time.

In-vessel LOCA Concern

Water. The piping and tubing material choice may be copper rather than steel, as mentioned above. Even though copper tubing and piping is widely used in a variety of industries, no failure data bank on copper piping or tubing has been found. A source of copper plate operating experience information comes from its use as resistive fusion magnets. From Table 7-4 in the main report, failure rate of $6.4\text{E-}06/\text{m-h}$ with an error factor of 10 is used for copper tubing leakage in the ITER machine. The tubing rupture failure rate is $6.4\text{E-}07/\text{m-h}$ with an error factor of 10. These values are quite large. Copper usage in other industries should be carefully reviewed to verify this failure rate. Other tubing materials would give lower failure rate values.

Using 13,000 m for the tube length and 2200 hours/year for ITER operation gives a tube rupture frequency of $(6.4\text{E-}07/\text{m-h} \times 13,000 \text{ m} \times 2200 \text{ h}) = 18/\text{year}$, with an error factor of 10. This is a very high value. The copper tubing failure rates should be examined in more detail if copper is used for ITER.

Other tubing materials are examined from Table 7-4. These results give a band of frequencies, and these are presented at the end of this section.

Another issue concerning in-vessel water piping reliability is its vulnerability to plasma effects. The most notable effect is runaway electrons generated during plasma disruptions. Water piping seems most susceptible because of the heating effects from electron energy dissipation; helium coolant will not stop the electron, and liquid metals will not heat up appreciably (or boil) when stopping the electron. Unless well shielded, a runaway electron could expend its energy by either melting a hole in a copper tube or by heating the water coolant in the tube, leading to phase change, overpressurization, and LOCA. Runaway electron damage has been noted in existing machines,^{A-4} usually on the outboard midplane of a tokamak, so perhaps first wall cooling systems are more susceptible than the divertor. Quantifying the frequency for runaway electron damage to the cooling systems is highly subjective and requires engineering judgment. The event is definitely credible since it has occurred on existing tokamaks, and not extremely unlikely because runaway electron damage has occurred at several machines. Therefore, as a first approximation the runaway electron damage frequency is in the range of 1E-02 to 1E-04/year, the unlikely events frequency range. To be conservative, the high end of the frequency range gives 1E-02/year for a runaway electron event that causes failure of a first wall water coolant tube (unless the tubes are well protected). This value is of lesser magnitude than other leakage events, but if the tubes are made very reliable, then runaway electrons will be a dominant factor for in-vessel coolant leakage.

Helium. High pressure helium in copper tubes is examined here for comparison to the other tubing design approaches. For copper tubes, Table 7-1 in the main report gives a leakage failure rate of 7.5E-06/m-h with an error factor of 3, and a rupture failure rate of 7.5E-07/m-h, also with an error factor of 3.

Using 6,000 m of helium tubing and 2200 hours/year of ITER operation, the rupture frequency is $(7.5E-07/m-h \times 6,000 \text{ m} \times 2200 \text{ h/yr}) = 9.9/\text{year}$, with an error factor of 3. This result is half the water tubing rupture value.

Other materials are also considered from Table 7-1 in the main body of this report. These other materials give a band of frequency results for helium, and these results are given at the end of this section.

Liquid Metal. Liquid metal piping is usually steel. Normal design practice for fusion reactors is to eliminate ferritic steels anywhere near the magnets, but ferritic steels have a breadth of operating experiences with liquid metal coolants. Using the ferritic steel as a guide, then Tables 7-2 and 7-3 in the main report give a ferritic steel tubing leakage failure rate of $1.9\text{E-}09/\text{m-h}$ with an error factor of 10, and a rupture failure rate of $1.9\text{E-}10/\text{m-h}$, with an error factor of 10. These values apply to all liquid metals, because there are no good failure data sources for liquid gallium, lithium, or lead systems. An assumption that these systems would be designed and operated as well as sodium systems is implied when using these failure rates on fusion liquid metal systems.

Using 8,000 m of tubing length and 2200 hours/year gives a tubing rupture frequency of $(1.9\text{E-}10/\text{m-h} \cdot 8000 \text{ m} \cdot 2200 \text{ h/yr}) = 3\text{E-}03/\text{year}$. This is the best result thus far for tube ruptures.

Other materials for liquid metal (inconel and stainless steel, copper is not compatible with many liquid metals so it is excluded here) tubing are examined here and the results are given below with those for helium and water.

Yearly accident frequency comparison for the in-vessel LOCA

Water

<u>Tubing material</u>	<u>Rupture failure rate (/m-h)</u>	<u>multipliers</u>	<u>Rupture frequency (/y)</u>	<u>Error factor</u>
316 SS	1.2E-09	2200 h/y*13000 m	3.4E-02	3
copper	6.4E-07	2200 h/y*13000 m	1.8E+01	3
inconel	4E-11	2200 h/y*13,000 m	1.1E-03	10
titanium	9.5E-11	2200 h/y*13,000 m	2.7E-03	10

Helium

<u>Tubing material</u>	<u>Rupture failure rate (/m-h)</u>	<u>multipliers</u>	<u>Rupture frequency (/y)</u>	<u>Error factor</u>
316 SS	1.4E-09	2200 h/y*6000 m	1.8E-02	3
copper	7.5E-07	2200 h/y*6000 m	9.9E+00	3
inconel	4E-11	2200 h/y*6000 m	5.4E-04	10
titanium	1E-10	2200 h/y*6000 m	1.3E-03	10

Liquid metal

<u>Tubing material</u>	<u>Rupture failure rate (/m-h)</u>	<u>multipliers</u>	<u>Rupture frequency (/y)</u>	<u>Error factor</u>
316 SS	1.4E-09	2200 h/y*8000 m	2.5E-02	3
inconel	4E-11	2200 h/y*8000 m	7.0E-04	10
titanium	1E-10	2200 h/y*8000 m	1.8E-03	10

Compiling all of the results identified in this in-vessel LOCA section gives the list shown below. These LOCA values are considered to be small LOCAs, based on the rupture of small diameter tubing.

High pressure water	1E-03 to 18/year	error estimate of 10
Helium	5E-04 to 10/year	error estimate of 10
Liquid Metals	7E-04 to 3E-02/year	error estimate of 10

The yearly frequency ranges for water LOCAs in the ITER vacuum vessel is high. Smaller values, but still large - on the order of 2/year for small leakages from coolant tubes - have also been reported from earlier ITER reliability studies.^{A-5} The main reasons for these high spread ranges are the chosen failure rates for the varying materials, and the extra 5 km of water tubing over that used by other coolants. It is reasonable to see a high frequency when dealing with over 10 km of tubing or piping - an entire nuclear fission plant has probably only about 6 or 8 km of large piping, and such plants can experience small leaks many times a year.

In-Vessel LOFA Concern

This event is important for several reasons. First, it is a very real possibility, because the event has occurred once at TORE SUPRA.^{A-6} In the TORE SUPRA event, a copper PFC cooling tube lost flow, exceeded its critical heat flux, then ruptured, sending about 0.5 liter of water into the vacuum vessel. The water caused a plasma disruption. The second reason is that if one of a few tubes plugged with some kind of debris, the water flow would bypass those tubes and flow monitors at the inlet and/or outlet manifolds would not sense any problem until the tube or tubes burst.

Failure rates for tube plugging (debris fouling, not to be confused with workers plugging a leaky steam generator tube) are considered here. Other LOFA causes, such as pump failure, etc., are treated in the ex-vessel LOFA section. This in-vessel accident, while potentially consequential, is still less damaging than the loss of flow from an entire cooling loop.

An important point to note here is that one design alternative under consideration for ITER and future machines is microchannel cooling. Conceptually, microchannels are similar to some fission reactor fuel elements, where water flows in the small rectangular spaces between the thin fuel plates. Such reactors have had past problems with fuel channel blockage^{A-7} from foreign materials in the coolant system (gasket material, crud, wad of tape, lucite shards, ball point pen, etc.); with the microchannel design then flow blockage is a very real possibility. Great care in construction, operation, and maintenance must be taken if microchannels are to be used in future fusion devices.

In Section 5.13.3 of the main report, tube plugging was addressed with the guidance that it is about as frequent as the rupture failure mode. For the in-vessel cooling tubes, the plugging loss of flow failure rate would be the same as the in-vessel LOCA frequency already described. One exception would be helium tubing, where the coolant is not as corrosive (although water vapor could be in the piping, entrained with the helium) as other coolants. No reduction in the helium plugging frequency will be made, since there is the possibility of water vapor entrainment in

the helium flow.

Microchannel cooling systems are quite vulnerable to plugging by corrosion products and foreign material intrusion. The failure rate for microchannel plugging could easily be a factor of 10 higher than for small tubes. Only results for cooling tube designs are presented below:

Yearly accident frequency comparison for the in-vessel LOFA

High pressure water	1E-03 to 18/year	error estimate of 10
Helium	5E-04 to 10/year	error estimate of 10
Liquid Metals	7E-04 to 3E-02/year	error estimate of 10

The in-vessel LOFA accident frequencies vary the same as the small LOCA frequencies, since they are based on the same failure rates.

Ex-Vessel LOCA Concern

The ex-vessel piping for all coolant types will likely be stainless steel, for its strength and corrosion resistance. Unfortunately, each of the coolants behave in different ways, so the failure mechanisms and failure rates will vary. This application is very similar to fission power plant cooling systems, so component failure rates from fission operating experiences will be used for comparisons. Because there are few valves assumed in the external piping, valve body ruptures will not be included. Only pipe breaches will be considered for the ex-vessel LOCA.

Water. In fission power, a large concern for water usage is proper chemistry control. Water is dissociated by radiation, and this process enhances corrosion reactions. The heat transfer surfaces must be kept clean - no fouling or crud buildup - because of the high heat transfer rates in reactor cores. Any air ingress to a coolant system can lead to nitric acid formation with radiolytic decomposition products.^{A-8} High water purity is essential for good operation, because chlorides or other compounds could cause stress corrosion cracking in steels. In practice,

vibration fatigue appears to be the most dominant coolant piping failure concern, with expert opinion giving corrosion as the second failure cause.^{A-3} The failure rate for LOCA-sensitive piping in a pressurized water reactor (PWR, 15 MPa pressure) is $5\text{E-}04/\text{plant-year}$, for all leak sizes, 3 l/s to 30 l/s and higher leak rates. Since there is roughly 3400 m of LOCA sensitive piping in a typical PWR plant, and considering 7000 hours/year (about 80% availability), the LOCA failure rate is about $2\text{E-}11/\text{hour-m}$ (error estimate of 12).

Helium. Helium gas cooled reactors have had concerns with piping embrittlement caused by helium nuclei (alpha particles created from fast neutron reactions in the steel) agglomerating into bubbles at the grain boundaries of the steel,^{A-9} and with helium induced swelling and blister formation.^{A-10} Fusion reactors will suffer from this problem. Fortunately, neither the Fort St. Vrain or Peach Bottom 1 reactors noted any piping degradation of that type.^{A-11,12} However, Fort St. Vrain did suffer from many other problems associated with water ingress from steam generator tube leaks and gas circulator (compressor) bearing water leaks. Volatile chlorides were leached because of the inleakage moisture, causing stress corrosion cracking in a stainless steel control rod cable, in a stainless steel bolt from one of the circulators, in the solid boron/graphite spheres of the reserve shutdown system, and may also have caused the bearing seize up that prevented 6 control rods from scrambling on demand.^{A-11} Moisture ingress was a severe problem. The loss of the 6 control rods in June 1984 was an event that Fort St. Vrain never recovered from - the plant never again operated at any high power level. The plant was decommissioned in August 1989. A way to avoid the Fort St. Vrain moisture ingress problem is to choose another gas circulator cooling and sealing approach, and to design the heat exchanger with the shell side at a slightly lower pressure than the tube side, so any leakage is helium out from the primary coolant loop, rather than water into it. If running the heat exchanger in this manner is not possible, double walled tubes might solve the problem.

The failure rate for LOCAs (disruptive failures) in helium pipes is $4\text{E-}12/\text{hour-m}$ (error estimate of 30).^{A-13} This value was obtained from gas cooled reactor operating experience and test loop operations.

Liquid metal. Chemical compatibility is a large concern for liquid metal pipe reliability. Lithium, sodium, and lead are compatible with stainless steel, but gallium is regarded as aggressive in its chemical attack of steels and other solid metals, and it appears that stainless steels below 200 C are much less vulnerable than inconel alloys.^{A-14} In general, liquid metals can sensitize alloys; that is, they can cause the alloy to precipitate chromium carbides at the grain boundaries, weakening the alloy material.^{A-15}

An estimated liquid metal pipe rupture failure rate, based on fission operating experience data^{A-16} is 1E-09/hour-m (error estimate of 10). While the fission experience is with liquid sodium, it is the best indicator of liquid metal piping performance that is available. It is true that water system component failure rates have been applied to sodium systems,^{A-17} and that sodium coolant system failure rates have been applied to lithium-lead fusion systems,^{A-18} so applying these data to other liquid metals seems reasonable.

The failure rates for various coolants to undergo a large ex-vessel LOCA are given below. The accident frequencies have been calculated with the operating hours and pipe length information given earlier.

Failure rate comparison for ex-vessel LOCA

High pressure water	2E-11/hour-m	error estimate of 12
Helium	4E-12/hour-m	error estimate of 30
Liquid metal	1E-09/hour-m	error estimate of 10

Accident frequency comparisons for ex-vessel LOCA

High pressure water	4E-05/year	error estimate of 12
Helium	9E-06/year	error estimate of 30
Liquid metal	2E-03/year	error estimate of 10

Helium has the lowest accident frequency for ex-vessel LOCA. Still, there are concerns for each coolant. The most well known concerns are for water. Water

chemistry control is an issue for suppressing both neutron activation of impurities and wall corrosion rates. The necessary piping bends, such as ring headers, manifolds, etc., can cause erosion problems for fast flowing (over a few m/s) water.

Ex-Vessel LOFA Concern

Pipe plugging could occur due to the buildup of corrosion products or impurities at flow constrictions or cool areas of the piping, or due to the introduction of foreign materials. Reactor piping systems have had pieces of wood from a board, a paint brush, a work glove, welding slag, various tools, metal parts (wire, ball bearings, etc.), lengths of heavy-duty tape, and other debris left in them after maintenance and/or initial construction activities.^{A-7}

A LOFA can occur for several reasons: loss of pump power (usually considered under the loss of offsite power event, because typically all of the pumps are affected), spurious closure of an isolation valve, plugging of a valve or coolant pipe, or an individual pump failure. The latter three events are considered here for each coolant. For this calculation, only one module of divertor cooling is considered, not the entire system. Losing the flow to only one divertor module would be consequential enough.

Water. The isolation valves between each major component are typically provided to allow maintenance on each component. Just one major isolation valve failing shut will completely shut down the coolant flow loop. Considering one valve between the pump and the heat source, one valve between the heat source and the heat exchanger, and one valve between the heat exchanger and the pump, there are three valves in a system. For motor operated valves, a spurious closure failure rate^{A-16} is $5\text{E-}08/\text{hour}$ (error estimate of 10). For 3 valves, the rate is about $2\text{E-}07/\text{hour}$. The valve plugging failure rate^{A-16} is $5\text{E-}09/\text{hour}$ (error estimate of 10), so for the 3 valves that rate is $2\text{E-}08/\text{hour}$. Pipe plugging is a very low probability event for large pipe diameters. Some risk assessments have assumed that plugging is incredible for pipe diameters greater than 305 mm. For piping larger than 75 mm, the plugging failure rate is $2\text{E-}10/\text{hour-m}$ (error estimate of 30).^{A-16}

The failure-to-run failure rate for a large centrifugal pump varies in the reliability literature, but failure-to-run values of $1\text{E-}05/\text{hour}$ (error estimate of about 3)^{A-19} to $1\text{E-}04/\text{hour}$ (no error estimate reported)^{A-20} for different fission plant operating experiences have been reported. The value of $1\text{E-}05/\text{hour}$ is used in this analysis. Possible pump problems could be a short circuit of the motor windings, pump shaft shear, imbalance that trips the unit off line, pump seizure from improper lubrication or impeller binding, or pump trip due to depressurization (gross seal leakage fault) or vapor lock in the pump casing.

Helium. Helium isolation valves have a spurious closure failure rate of $3\text{E-}06/\text{hour}$ (error estimate of 10).^{A-13} Again considering 3 valves in the system gives a failure rate of about $1\text{E-}05/\text{hour}$ for a given coolant loop. Pipe plugging failure rates are considered negligible because the helium flow rate is high (an order of magnitude higher than that of water) and the helium is inert, so no corrosion processes should occur in the piping if impurities are controlled.

Electric motor driven helium circulators (compressors) can experience seal failures (about half of all outages), motor failures, and imbalance trips. For faults severe enough to require a parameter safeguard, the circulator failure rate is $0.06/\text{operating year}$ (with an error estimate of less than 6).^{A-21} Considering 80% availability, or about 7000 hours, this failure rate is $9\text{E-}06/\text{hour}$.

Liquid metal. Liquid metals can experience corrosion-induced plugging, just as other fluids can. They are susceptible to having the MHD coating disintegrate and begin to plug up elbows, bends, etc. Also, while liquid metal coolants can be 'freeze plugged', there should still be isolation valves in the coolant system. A liquid metal system's motor operated valve spurious closure failure rate^{A-16} value is $5\text{E-}07/\text{hour}$ (error estimate of 10), and valve plugging has a failure rate value of $5\text{E-}08/\text{hour}$ (error estimate of 10).^{A-16} These failure rate values may be high, according to a comparison of water and liquid metal valve failure rates,^{A-22} but they will be used until further information is available. For 3 valves, the failure rates are about $2\text{E-}06/\text{hour}$ and $2\text{E-}07/\text{hour}$, respectively. For pipes with diameters between 25 and 100 mm, plugging has a failure rate of $3\text{E-}09/\text{hour-m}$ (error estimate of 10).^{A-20} This failure rate does not account for the MHD coating in the

pipe, but the effect of the coating cannot be quantified at this time. Plugging is not considered credible for larger pipes, even though the metal could freeze. Freezing is very unlikely in such a large pipe because of the heat contained in the fluid.

Liquid metal electromagnetic induction pumps are cited as having a fail-to-run failure rate of $1\text{E-}05/\text{hour}$ (error estimate of 10).^{A-16} The Experimental Breeder Reactor II has a large sodium electromagnetic pump that has operated flawlessly for 28 years at about 80% availability.^{A-23} This performance gives a conservative, overall failure rate of $1/(28 \text{ years})(8760 \text{ hours/year})(0.80) = 5\text{E-}06/\text{hour}$. Since that experience might be better than average, the $1\text{E-}05/\text{hour}$ value is sufficient for the purpose here.

The failure rates for the various coolant components are compiled here for comparison. Comparative accident frequencies are calculated, based on the pipe length and run time information given earlier.

Failure rate comparison for ex-vessel LOFA

High pressure water	2E-07 valve closure/hour	error estimate of 10
	2E-08 valve plug/hour	error estimate of 10
	2E-10 pipe plug/hour-m	error estimate of 30
	1E-05 pump fail/hour	error estimate of 3
Helium	1E-05 valve closure/hour	error estimate of 10
	negligible valve plugging	
	negligible pipe plugging	
	9E-06 pump fail/hour	error estimate of 6
Liquid metal	2E-06 valve closure/hour	error estimate of 10
	2E-07 valve plug/hour	error estimate of 10
	3E-09 pipe plug/hour-m	error estimate of 10
	1E-05 pump fail/hour	error estimate of 10

Accident frequency comparison for ex-vessel LOFA

High pressure water	2E-01/year	error estimate of 30
Helium	4E-02/year	error estimate of 10
Liquid Gallium	3E-02/year	error estimate of 10
Liquid Lead	3E-02/year	error estimate of 10
Liquid Sodium	3E-02/year	error estimate of 10

Summary

The results for this accident frequency investigation are given in Table A-2, ranked by the point estimate values. The uncertainties are largely a factor of 10 and are not considered in this ranking, where #1 denotes the best choice from an accident safety perspective. In some cases, the choice between 1, 2, or 3 is close, and depends greatly on the tubing material.

Table A-2. Results of the coolant system component accident frequency investigation.

<u>Portion of System Being Examined</u>	<u>LOCA Concern</u>	<u>LOFA Concern</u>
In-vessel cooling (divertor cooling tube network)	1) helium 2) liquid metal 3) water	1) helium 2) liquid metal 3) water
Ex-vessel cooling (exterior piping and other components in the divertor cooling system)	1) helium 2) water 3) liquid metal	1) liquid metal 2) helium 3) water

The results show that helium is a good choice, only when the proper tubing material (titanium or inconel) is used. Helium was also a reasonable choice for reducing ex-vessel LOFA effects. Helium has the advantage of no PFC chemical reactions (it would cause a plasma disruption, but no tile burning), and helium systems do not plug up as readily as other coolant systems, because there are reduced corrosion effects - as long as the helium is kept dry. Water is often considered because it has several advantages, such as a well established set of design guidelines from fission, much industrial experience with its use, better heat transfer ability and much lesser expense than helium.

Helium coolant is the best choice from a safety perspective, if the operational problem of its low convective heat transfer can be resolved.

Appendix A References

- A-1. B. J. Merrill and S. C. Jardin, "Coolant Ingress Induced Disruption Calculations for ITER," Fusion Technology, 19, 1991, pages 1278-83.
- A-2. M. M. El-Wakil, Nuclear Energy Conversion, American Nuclear Society publishers, 1978. chapter 7.
- A-3. R. E. Wright et al., Pipe Break Frequency Estimation for Nuclear Power Plants, NUREG/CR-4407, EGG-2421, May 1987, appendix B.
- A-4. H. Hoven et al., "Material Damage in Graphite by Run-away Electrons," Journal of Nuclear Materials, 162-164, 1989, pages 970-975.
- A-5. A. Pashkov, Estimation of LOCA Occurrence Rate for ITER First Wall and Divertor Cooling Systems, ITER-IL-SA-4-0-49, ITER Safety and Environment Homework Task 1, July 1990.
- A-6. K. J. Dietz et al., "Experience with high heat flux components in large tokamaks," Fusion Engineering and Design, 16, 1991, pages 229-251.
- A-7. L. C. Cadwallader, memo to D. M. Henry, "Transmittal of the Historical Review of Fission Reactor Fuel Blockage Events," letter LCC-46-90, EG&G Idaho, Inc., October 31, 1990.
- A-8. P. Cohen, Water Coolant Technology of Power Reactors, American Nuclear Society, Chicago, 1980.
- A-9. A. A. Sagues et al., "The influence of helium on the high temperature mechanical properties of an austenitic stainless steel," Journal of Nuclear Materials, 78, 1978, pages 289-298.
- A-10. H. Yamada, "Neutron-induced Helium Implantation in Helium Coolant Pipes for Fusion Reactors," Journal of Nuclear Materials, 103&104, 1981, pages 615-618.
- A-11. D. L. Moses and W. D. Lanning, "Analysis and Evaluation of Recent Operational Experience from the Fort St. Vrain HTGR," Nuclear Safety, 27, January-March 1986, pages 58-69.
- A-12. W. C. Birely, "Operating Experience of the Peach Bottom Atomic Power Station," Nuclear Engineering and Design, 26, 1974, pages 9-15.
- A-13. L. C. Cadwallader and S. J. Piet, 1989 Failure Rate Screening Data for Fusion Reliability and Risk Analysis, EGG-FSP-8709, EG&G Idaho, Inc., September 1989.

- A-14. Liquid-Metals Handbook, NAVEXOS-P-733, revision 1, Atomic Energy Commission and the Department of the Navy, June 1952, pp. 170-171.
- A-15. D. J. Wulpi, Understanding How Components Fail, American Society for Metals, Metals Park, Ohio, 1985, pages 240-241.
- A-16. S. A. Eide, S. V. Chmielewski, and T. D. Swantz, Generic Component Failure Rate Data Base for Light Water and Liquid Sodium Reactor PRAs, EGG-SSRE-8875, EG&G Idaho, Inc., February 1990.
- A-17. H. L. Thaggert and I. M. Jacobs, "Component Failure Rates Applicable to LMFBRs as Derived from LER Data," Nuclear Safety, 24, November-December 1983, pages 801-809.
- A-18. H. John et al., "Reliability Investigations and Improvements of the Cooling System of a Self-Cooled Liquid Metal Breeder Blanket," Fusion Technology 1992. Proceedings of the 17th Symposium on Fusion Technology, September 14-18, 1992, Rome, Italy, pages 1399-1403.
- A-19. Z. Milivojevic and J. R. Riznic, "The Empirical Failure Rate and Repair Rate of PWR Primary Coolant Pumps," Reliability Engineering and System Safety, 24, 1989, pages 267-273.
- A-20. Component Reliability Data for Use in Probabilistic Risk Assessment, IAEA-TECDOC-478, International Atomic Energy Agency, Vienna, 1989.
- A-21. L. Cave and R. S. Gow, "Experience of the Reliability of Steam and Electrically Driven Circulators for GCRs," Nuclear Engineering International, 22, August 1977, pages 57-60.
- A-22. S. L. Painter et al., "CREDO Data Analysis: A Comparison of Liquid Metal and Water Valve Reliabilities," Proceedings of the International Conference on Nuclear Power Plant Aging. Availability Factor and Reliability Analysis, San Diego, CA, 8-12 July, 1985, American Society for Metals, pages 523-526.
- A-23. H. W. Buschman, National Laboratory-West, private conversation, August 5, 1992.

**DATE
FILMED**

8 / 12 / 94

END