NUREG-0358
August 1, 1978

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

FOR

U.S. DEPARTMENT OF ENERGY

FAST FLUX TEST FACILITY

PROJECT NO. 448

This document is
PUBLICLY RELEASABLE
B Steele
Authorizing Official
Date: 5-9-06

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

TABLE OF CONTENTS

		PAGE
1.0	INTRODUCTION AND GENERAL DISCUSSION	
1.1	Introduction	1-1
1.2	General Plant Description	1-5
1.3	Interactions at the Site	1-8
1.4	Experience with Similar Facilities	1-8
1.5	Identification of Agents and Contractors	1-13
1.6	Summary of Principal Review Matters	1-14
1.7	Facility Modifications and/or Changes During the Review	1-18
1.8	Requirements for Future Technical Information	1-21
1.9	Summary of Outstanding Issues	1-23
2.0	SITE CHARACTERISTICS	
2.1	Geography and Demography	2-1
2.2	Meteorology	2-1
2.2.1	Regional Climatology	2-5
2.2.2	Local Meteorology	2-8
2.2.3	Onsite Meteorological Measurements Program	2-8
2.2.4	Short-Term (Accident) Diffusion Conditions	2-10
2.2.5	Long-Term (Routine) Diffusion Estimates	2-13
2.2.6	Conclusions	2-13
2.3	Hydrology	2-14
2.4	Geology/Seismology	2-14
3.0	DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS	
3.1	Conformance with General Design Criteria	3-1
3.2	Classification of Structures, Components, and Systems	3-2
3.2.1	Seismic Classification	3-2
3.2.2	System Quality Group Classification	3-4
3.3	Wind and Tornado Criteria	3-5
3.3.1	Wind Design Criteria	3-5
3.3.2	Tornado Design Criteria	3-6

TABLE OF CONTENTS (Cont'd)

		PAGE
3.4	Water Level Flood Design	3-6
3.4.1	Flood Protection	3-6
3.5	Missile Protection Criteria	3-7
3.5.1	Missiles Generated Outside of Containment	3-7
3.5.2	Internally Generated Missiles Inside Containment	3-9
3.6	Protection Against Postulated Piping Failures	3-10
3.6.1	Piping Failures Outside of Containment	3-10
3.6.2	Piping Failures Inside of Containment	3-11
3.7	Seismic Design	3-11
3.7.1	Seismic Input	3-11
3.7.2	Seismic System Analysis	3-12
3.7.3	Seismic Subsystem Analysis	3-13
3.7.4	Seismic Instrumentation Program	3-14
3.8	Design of Seismic Category I Structures	3-14
3.8.1	Steel Containment	3-14
3.8.2	Containment Interior Structures	3-15
3.8.3	Design of Other Seismic Category I Structures	3-17
3.8.4	Foundations	3-18
3.9	Mechanical Systems and Components	3-18
3.9.1	Special Topics for Mechanical Components	3-18
3.9.2	Dynamic Testing and Analysis of Mechanical System Components and Systems	3-19
3.9.2.1	Piping Preoperational Test Program	3-19
3.9.2.2	Seismic Qualification of Mechanical Equipment	3-19
3.9.2.3	Preoperational Flow-Induced Vibration Testing of Reactor Internals	3-20

TABLE OF CONTENTS (Cont'd)

	PAGE
3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports, and Code Support Structure	3-20
3.9.3.1 Design Load Combination and Stress Limits	3-20
3.9.3.2 Operability of Active Valves	3-21
3.9.4 Control Rod Drive Systems	3-21
3.9.5 Pressure Vessel Internals	3-22
3.9.6 Inservice Testing of Pumps and Valves	3-22
3.10 Seismic Qualification of Category I Instrumentation and Control Equipment	3-22
3.11 Environmental Qualification of Safety-Related Instrumentation and Control Equipment	3-24
4.0 REACTOR	
4.1 Description	4-1
4.2 Fuel System Design	4-4
4.2.1 Description	4-4
4.2.2 Design Bases and Limits	4-5
4.2.3 Design Evaluation	4-9
4.2.4 Fuel System Design Evaluation Summary and Conclusions	4-17
4.3 Nuclear Design	4-18
4.3.1 General	4-18
4.3.2 Power Distributions	4-18
4.3.3 Reactivity Coefficients	4-18
4.3.4 Control Rod Worth	4-19
4.3.5 Stability	4-20
4.3.6 Analytical Methods	4-20
4.4 Thermal-Hydraulic Design	4-20
4.4.1 Design Basis	4-20
4.4.2 Thermal-Hydraulic Limits	4-21
4.4.3 Flow Distribution	4-22
4.4.4 Thermal Power Distributions	4-24

TABLE OF CONTENTS (Cont'd)

		PAGE
4.4.5	Upper Plenum	4-24
4.4.6	Design Analysis Methods	4-26
4.4.7	Thermal-Hydraulic Uncertainty Analysis	4-29
4.4.8	Hydraulic Stability	4-29
4.4.9	Loose Parts Monitoring System	4-31
4.4.10	Evaluation and Findings	4-31
4.5	Mechanical Design	4-32
4.5.1	Control Rod System Structural Materials	4-32
4.5.1.1	Control Rod Drive Mechanism and Drive Line	4-32
4.5.1.2	Control Rod Absorber Assembly	4-33
4.5.1.3	Control Rod System Materials Evaluation	4-34
4.5.2	Reactor Internals	4-34
4.5.2.1	Core Support Structure	4-35
4.5.2.2	Radial Shield	4-36
4.5.2.3	Instrument Trees	4-36
4.5.2.4	In-Vessel Handling Machine	4-37
4.5.2.5	Reactor Internals Materials Evaluation	4-37
4.5.2.6	Reflectors	4-38
4.6	Functional Design of Reactivity Control System	4-38
5.0	REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS	
5.1	Reactor Vessel System	5-1
5.1.1	Reactor Vessel	5-1
5.1.2	Reactor Head Assembly	5-2
5.1.3	Reactor Guard Vessel	5-3
5.1.4	Reactor Vessel System Materials Evaluation	5-3
5.2	Heat Transport System	5-5
5.2.1	Primary Sodium Loop	5-5
5.2.2	Guard Vessels	5-6
5.2.3	Secondary Sodium Loop	5-6

TABLE OF CONTENTS (Cont'd)

	PAGE
5.2.4 Insulation	5-7
5.2.5 Verification and Acceptance Test Programs	5-8
5.2.6 Leak Detection	5-8
5.2.7 Heat Transport System Materials Evaluation	5-10
5.3 Overpressure Protection	5-11
5.4 Surveillance and Inservice Inspection	5-12
5.4.1 Design Verification Test Program	5-13
5.4.2 Operation and Maintenance Test Program	5-13
5.4.2.1 Visual Inspection	5-13
5.4.2.2 Leak Detection	5-14
5.4.2.3 Materials Surveillance	5-14
5.4.3 Evaluation of Materials Surveillance and Inservice Inspection	5-15
5.4.4 Recommendations	5-16
5.4.4.1 Leak Detection	5-16
5.4.4.2 Material Surveillance	5-16
5.4.4.3 Preservice Inspection	5-17
5.5 Decay Heat Removal System	5-17
6.0 ENGINEERED SAFETY FEATURES	
6.1 Summary Description	6-1
6.2 Containment System	6-1
6.2.1 Containment Functional Design	6-1
6.2.2 Ex-Containment Secondary Sodium System Fire Consequences	6-5
6.2.3 Containment Isolation	6-6
6.2.4 Containment Leakage Testing	6-8
6.3 Interim Decay Storage Emergency Cooling System	6-9
6.4 Control Room Habitability	6-9
6.4.1 General Description	6-10
6.4.2 Radiation Protection Provisions	6-11
6.4.3 Toxic Vapor Protection	6-11

TABLE OF CONTENTS (Cont'd)

		PAGE
7.0	INSTRUMENTATION AND CONTROLS	
7.1	Summary Description	7-1
7.1.1	Evaluation	7-1
7.2	Reactor Shutdown Systems	7-3
7.3	Engineered Safety Feature Systems	7-8
7.4	Systems Required for Safe Shutdown	7-11
7.5	Safety-Related Display Instrumentation	7-12
7.6	All Other Instrumentation Systems Required for Safety	7-14
7.7	Control Systems Not Required for Safety	7-17
8.0	ELECTRIC POWER	
8.1	General	8-1
8.2	Offsite Power Systems	8-1
8.3	Onsite Power Systems	8-2
8.3.1	Emergency Power System	8-2
8.3.2	Safety Power System	8-3
8.3.3	Physical Independence of Electrical Systems	8-4
8.3.4	Environmental Qualification of Class IE Equipment	8-5
8.3.5	Power Lockout to Manually-Controlled Electrically Operated Valves	8-5
8.3.6	Electrical Connectors and Terminal Blocks	8-5
8.3.7	Electrical Penetrations	8-5
9.0	AUXILIARY SYSTEMS	
9.1	Summary Description of Fuel Storage and Handling	9-1
9.2	Facility Systems	9-5
9.2.1	Component and Compartment Cooling System	9-5
9.2.2	Inert Gas System	9-6
9.2.3	Failed Element Detection System	9-7
9.3	Process Auxiliaries	9-8

TABLE OF CONTENTS (Cont'd)

	PAGE
9.3.1 Compressed Air System	9-8
9.3.2 Process Sampling System	9-9
9.3.3 Liquid Metal Receiving Storage and Processing System	9-10
9.4 Heating and Ventilating Systems	9-11
9.4.1 Control Room Area Ventilation System	9-11
9.4.2 Auxiliary Equipment Area Ventilation System	9-12
9.5 Service Systems	9-12
9.5.1 Fire Protection System	9-12
9.5.2 Communication Systems	9-13
9.5.3 Lighting System	9-14
9.5.4 Emergency Diesel Generator Auxiliary Systems	9-15
9.5.5 Emergency Diesel Engine Fuel Oil Storage and Transfer System	9-16
9.5.6 Emergency Diesel Engine Cooling Water System	9-16
9.5.7 Emergency Diesel Engine Starting System	9-17
9.5.8 Emergency Diesel Engine Lubrication System	9-17
9.5.9 Emergency Diesel Engine Combustion Air Intake and Exhaust System	9-18
10.0 CLOSED LOOP SYSTEM	
11.0 RADIOACTIVE WASTE MANAGEMENT SYSTEMS	
11.1 Summary Description	11-1
11.2 Liquid Waste System	11-2
11.3 Gaseous Waste System	11-3
11.4 Solid Waste System	11-5
11.5 Process and Effluent Radiological Monitoring System	11-6
11.6 Evaluation Findings	11-7
12.0 RADIATION PROTECTION	
12.1 General	12-1
12.2 Radiation Sources	12-2
12.3 Radiation Protection Design Features	12-3
12.4 Dose Assessment	12-6
12.5 Health Physics Program	12-7

TABLE OF CONTENTS (Cont'd)

PAGE

13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Qualifications..... 13-1
13.2 Emergency Planning..... 13-2
13.3 Review and Audit..... 13-2

14.0 INITIAL TEST PROGRAMS

14.1 Summary..... 14-1
14.2 Natural Circulation Verification Test Plan..... 14-2

15.0 ACCIDENT ANALYSIS

15.1 General..... 15-1

15.1.1 Reactivity Insertion Events..... 15-2
15.1.2 Heat Removal Reduction Events..... 15-4
15.1.3 Local Fuel Failure Events..... 15-8
15.1.4 Closed Loop System Safety Analysis..... 15-9

15.2 Ex-Reactor Safety Analysis..... 15-10

15.2.1 General..... 15-10
15.2.2 Ex-Reactor Fuel Handling..... 15-10
15.2.3 Ex-Reactor Fuel Storage..... 15-11
15.2.4 Radioactive Waste Processing..... 15-11
15.2.5 Sodium Fires..... 15-12

15.3 Analysis and Evaluation of Core Meltdown and Disassembly Accidents..... 15-12

15.3.1 Transient Overpower Assessment..... 15-12
15.3.2 Loss-of-Flow Accident Assessment..... 15-22
15.3.3 Recriticality..... 15-25
15.3.4 Mechanical Disruption/Structural Response..... 15-35
15.3.5 In-Vessel Post-Accident Heat Removal..... 15-39
15.3.6 Containment Response to a Core Meltdown Event..... 15-40
15.3.7 Radiological Consequences of Core Disruptive Accidents..... 15-58

16.0 TECHNICAL SPECIFICATIONS
17.0 QUALITY ASSURANCE

TABLE OF CONTENTS (Cont'd)

	PAGE
17.1 General	17-1
17.2 Organization	17-1
17.3 Quality Assurance Program	17-5
17.4 Conclusions	17-6
18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS	
19.0 CONCLUSIONS AND ADVICE	
19.1 Conclusions	19-1
19.2 Staff Review	19-1
19.3 Advice and Recommendations	19-2
19.4 Miscellaneous Unresolved Issues	19-8

LIST OF FIGURES

		PAGE
2.1-1	Location of Hanford Reservation	2-2
2.1-2	Cumulutive Population Distribution About FFTF (1970 Census)	2-3
2.2-1	Directional Frequency of Wind - WPPSS Site	2-9
4.1-1	Reactor Elevation	4-2
4.1-2	FFTF Core Map	4-3
4.4-1	Fraction of Flow Through Fuel Assemblies	4-25
4.4-2	Reactor Vessel Outlet Temperature Drop (Normal Scram)	4-27
7.2-1	Primary, Secondary RSS and HTS Shutdown System Block Diagram	7-4
15.3-1	Overall Transient Response of 20-Channel BOC-4 Base Case	15-16
15.3-2	Pressure Vs. Volume Disassembly Calculations	15-36
15.3-3	Sectional View of Reactor Vessel and Cavity Showing PAHR Melt-Through Level of Sodium Pool	15-45
17.2-1	Hanford Engineering Development Laboratory Organization	17-3
17.2-2	Hanford Engineering Development Laboratory Quality Assurance Program	17-4

LIST OF TABLES

1.4-1	Comparison of Liquid Metal Fast Breeder Reactors in the United States	1-10
1.4-2	Foreign Reactor Comparison to FFTF	1-12
2.1-1	Projected Population Within 10 Miles of FFTF Site	2-4
2.1-2	Population Projections	2-4
2.2-1	Design Basis Tornado Characteristics for the FFTF Site	2-6
2.2-2	WPPSS 1, 2, and 4 Meteorological Tower Instrumentation	2-10
2.2-3	Short-Term Relative Concentration by Downwind Direction - FFTF Site	2-11
2.2-4	Short-Term Relative Concentration Values Used for Accident Analyses - FFTF Site	2-11
2.2-5	Short-Term Relative Concentration Values Presented by the FFTF Project	2-12
2.2-6	Long-Term Relative Concentration Estimates by Downwind Direction - FFTF Site	2-13
4.2-1	Fuel Assembly Materials Listing	4-5
4.2-2	Cladding Strain and Temperature Limits	4-8
4.4-1	Thermal-Hydraulic Characteristics of the FFTF	4-21
4.4-2	Reactor Flow and Temperature Distribution	4-23
4.4-3	Hot Channel Factors at Initial Operating Conditions	4-30
6.2-1	Valves To Be Leak Tested	6-9
7.2-1	Plant Protection System Functions	7-5
9.3-1	Process Sample System Locations	9-9

LIST OF TABLES (Cont'd)

		PAGE
11.2-1	Design Parameters of Principal Components Considered in Evaluation of the Liquid and Gaseous Radioactive Waste Treatment Systems	11-3
11.5-1	Process and Effluent Monitoring Locations	11-6
15.1-1	Maximum Hot Channel Cladding Temperature for Each Flow Reduction Incident	15-6
15.3-1	FFTF VENUS-II Disassembly Calculations	15-38
15.3-2	Summary of In-Vessel Cases	15-47
15.3-3	Summary of Ex-Vessel Coolable Debris Bed Cases	15-49
15.3-4	Summary of the Effect of Liner Failure for Ex-Vessel Coolable Debris Bed Cases	15-51
15.3-5	Comparison of INTER and GROWS Code Predictions for Molten Pool Penetration into Reactor Cavity Concrete Floor	15-53
15.3-6	Summary of Ex-Vessel Cases	15-56
15.3-7	FFTF Containment Margins Summary	15-58
15.3-8	FFTF Meltdown Accident Doses, Rem	15-65
17.1-1	Regulatory Guidance for Quality Assurance	17-2

APPENDICES

APPENDIX	A	Chronology
	B	Reference Letters
	C	Sodium-Concrete Interactions
	D	Molten Fuel/Steel-Concrete Interactions
	E	Piping Integrity
	F	Staff Analysis of FFTF Containment Margins Ex-Vessel Scenario with Coolable Debris Bed Assumption
	G	Bibliography

1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Fast Flux Test Facility (hereinafter referred to as the FFTF) is a sodium-cooled fast spectrum experimental reactor owned by the Federal Government. The Department of Energy has authority for the operation of the FFTF and the Department's FFTF Project Office in Richland, Washington is responsible for its safe operation. The Nuclear Regulatory Commission (hereinafter referred to as the Commission), upon request of the Department of Energy, provides advice regarding safety issues and the adequacy of the FFTF design. This is in accordance with provisions of the Energy Research and Development Administration-Commission interagency agreement for the performance of safety reviews for reactors exempt from licensing.

Advice regarding the FFTF was requested by the Energy Research and Development Administration, now the Department of Energy, through the FFTF Project Office in a letter dated November 13, 1975, and further clarification was supplied by a second letter dated August 20, 1976. Both letters are included in Appendix B to this safety evaluation report.

The nature of the FFTF safety review by the Commission is advisory only and does not involve the issuance of an operating license. The review is directed to evaluating the adequacy of the design to ensure safe operation of the plant. The Department of Energy has a discretionary option of following the Commission's advice and has the final responsibility for the design and safe operation of the FFTF. The Commission has not provided inspection-enforcement support during the construction of the FFTF nor will it be involved in its operation under the existing interagency agreement.

The Commission however does expect to be provided certain acceptance test information reports in several areas identified in this safety evaluation report following startup of the facility since its advice is predicated on satisfactory completion of certain aspects of acceptance testing. The FFTF Project Office has onsite responsibility for the startup acceptance testing activities of the FFTF and all other activities relating to safe plant operation as well as supplying adequate security and safeguards during plant operation.

Our safety review of the FFTF has been rather complex in that it involved not only specific information relating to the FFTF but included substantial portions of the liquid metal fast breeder reactor base technology programs funded by the Department of Energy and performed by national laboratories and other organizations under

contract to the Department of Energy. Discrimination between the liquid metal fast breeder reactor base research and development program and those directly related to the FFTF has not always been clearly defined. Major milestones of interest during the construction stage of the FFTF safety review are listed below:

(1)	Site Safety Analysis Report	April 1970
(2)	Preliminary Safety Analysis Report Received by Staff	September 1970
(3)	Review Meetings with Staff (15)	November 1970 through July 1971
(4)	Interim Construction Agreement	July 1971
(5)	Design Basis Tornado Decision	November 1971
(6)	Review Meetings with Staff (4)	September 1971 through January 1972
(7)	Continuation of Construction Agreement	February 1972
(8)	FFTF Environmental Statement	May 1972
(9)	Design Safety Assessment Report	July 1972
(10)	Review Meetings with Staff (4)	September 1972 through January 1973
(11)	Safety Evaluation Report	October 1972
(12)	Limited Construction Agreement	May 1973
(13)	Supplement 1 to Safety Evaluation Report	December 1974
(14)	Supplement 2 to Safety Evaluation Report	March 1975
(15)	Commission Letter on Operating Basis Earthquake	April 1975
(16)	Advisory Committee on Reactor Safeguards Letter Regarding Balance of Construction	July 1975

During the construction review stage there were twenty-one Advisory Committee on Reactor Safeguards meetings and approximately twenty-one Commission staff meetings with the FFTF Project Office and their contractors to discuss liquid metal fast breeder reactor technology and the FFTF safety review.

During the current Final Safety Analysis Report review we have held many additional technical review meetings with the FFTF Project contractors and have conducted several site visits to observe as-built systems for implementation of design criteria. In addition, we have reviewed several hundred references submitted by the Project. Additional supplementary references are identified in the Final Safety Analysis Report, not all of which were reviewed by the staff. Our review of any of these references does not imply concurrence or approval of them for future liquid metal fast breeder reactors, in a generic sense, but only a recognition of the existence of some useful relevant information contributing to the safety review of the FFTF, in the same context as any other textual reference. In particular we did not

consider these documents or the reference System Design Descriptions, which are used by the FFTF Project as control documents, because they are subject to revision without the changes being correspondingly amended in the Final Safety Analysis Report at the same time. The amendments and supplements submitted by the FFTF Project have been considered as formal changes and form part of the Final Safety Analysis Report and have provided the basis of our findings. In addition to the staff review, the Advisory Committee on Reactor Safeguards is reviewing the FFTF and will meet with the FFTF Project, their contractors, and the Commission staff to discuss the final design and startup plans for the facility. A copy of the Advisory Committee on Reactor Safeguards' Report to the Commission will be provided to the Department of Energy FFTF Project Office.

The FFTF Environmental Statement, WASH-1510, dated May 1972, prepared by the Atomic Energy Commission (now the Department of Energy) was not updated for the Final Safety Analysis Report.

Our review objective has been to provide an in-depth technical review of the design of the FFTF comparable to that of a licensed plant. Site-related matters were considered to have been adequately reviewed during the construction phase and were not re-reviewed. An in-depth review was not requested but comments were solicited in the following areas:

(1) Operations
(2) Startup testing
(3) Quality assurance
(4) Technical Specifications

The scope of the review requested was defined by letter from R. L. Ferguson to R. P. Denise dated August 20, 1976. The scope of the review was later modified by letter of July 14, 1977, from R. L. Ferguson to R. P. Denise, advising us that it was not necessary for us to review the FFTF safeguards and security provisions. We responded in a January 17, 1978 letter, R. P. Denise to R. L. Ferguson, confirming that the staff will proceed with the evaluation of the FFTF without including this matter in its review. Therefore since we have not included the FFTF safeguards and security provisions in our review, we are unable to reach any conclusions regarding their effectiveness compared to those of licensed facilities, nor are we able to reach any conclusion on the completeness or adequacy of the evaluations performed by the Department of Energy and its contractors. Both the Project's and the staff's letters are included in Appendix B to this report. In June 1978, it was agreed that we defer the review of the closed loops since their design was incomplete. We will address this subject in a future supplement.

Our review reassessed those problem areas identified during the initial review in the Safety Evaluation Report dated October 31, 1972, Supplement 1 dated December 13, 1974, and Supplement 2 dated March 7, 1975.

FFTF DESIGN CHARACTERISTICS

HEAT TRANSPORT SYSTEM (3 PRIMARY LOOPS)	INITIAL OPERATION	RATED CONDITIONS
PLANT POWER	400 MWt	400 MWt
REACTOR OUTLET TEMP.	858°F	1050°F
REACTOR INLET TEMP.	600°F	792°F
FUEL SUBASSEMBLY OUTLET TEMP.	900°F	1100°F
ΔT-CORE	300°F	308°F
ΔT-REACTOR	258°F	258°F
IHX-LMTD	85°F	85°F
DHX MODULES	12 @ 33 MW	12 @ 33 MW
TOTAL COOLANT FLOW	41,000 GPM	42,730 GPM
SODIUM SYSTEMS COVER GAS	ARGON	

STEEL CONTAINMENT BUILDING	
SHELL MATERIAL, SHAPE	SA-516, GRADE 60 STEEL CYLINDER, ELLIPSOIDAL HEAD
SHELL SIZE	135 FT. DIAMETER x 187 FT. HIGH
CORE ARRANGEMENT	VERTICAL, 91 LATTICE POSITIONS WITH 73 DRIVER FUEL SUBASSEMBLIES IN HEXAGONAL ARRAY
SUBASSEMBLY LENGTH	12 FT. OVERALL, 3 FT. FUEL
FUEL PIN LENGTH	95 INCHES (FUEL & AXIAL blanket)
FUEL COMPOSITION	22-26 VOLUME % PuO_2 - 72-78 VOLUME % UO_2
FUEL TARGET BURNUP	45,000 MWD/TONNE AVERAGE
CAPABILITY FOR INSTRUMENTED IN-CORE OPEN TEST POSITIONS	8
INITIAL FLUX	7×10^{15} n/cm^2/sec
CLOSED LOOPS	
NUMBER	2 GENERAL PURPOSE - 2 MWt (each)

The more conventional portions of the facility have been reviewed as well as the reactor and heat transport system and are covered in this report except for the fire hazard evaluation, documentation for which was submitted late and which will be addressed in a supplement. Sodium fires are addressed in this report.

The organization of this report follows the pattern established by the Standard Format and Standard Review Plan insofar as practical rather than the Final Safety Analysis Report submitted by the Project.

1.2 General Plant Description

The major facility structures in the FFTF are free standing and constructed of concrete and steel. The containment structure is a steel shell insulated above ground and embedded in concrete below ground. Adjacent concrete structures house the control room, auxiliary services, diesel generators and the radioactive waste system. The heat rejection systems are primarily constructed of steel with concrete substructures.

The Fast Flux Test Facility is designed for the purpose of testing fuels and materials in a fast neutron flux environment of 7×10^{15} n/cm^2/sec. An additional stated function of the FFTF is to provide base technology for advanced reactor concepts. The reactor system design includes provision for two in-core closed loops, with space for two more, and includes four additional open test positions within the core matrix. The design reference core has 73 driver fuel subassemblies. The plant power is 400 MWt including pump power. The driver fuel is composed of mixed plutonium and uranium oxides. The reactor power is regulated through the use of three primary and six secondary control rods of B_4C. Either group of control rods can initiate reactor shutdown. The primary shutdown system will perform its functions even with the highest worth rod failing to insert. Variable speed pumps adjust flow to match power and maintain an essentially constant core ΔT with power variations. The two reactivity shutdown systems are mechanically and electrically separated to provide independent and redundant shutdown systems. However, the mechanical drives of all control rods are identical so that the goal of true diversity is not quite achieved. The common mode failure potential has been minimized by many life cycles of testing of control rod drives and components in a simulated sodium environment at operating temperature to simulate actual in reactor operation.

The reactor coolant is liquid sodium circulated in three parallel pipe loops common to the reactor vessel. The reactor system rated temperatures are 792°F inlet and 1050°F outlet. Heat energy is transferred from the primary sodium coolant system to a secondary sodium system. This system is also a piped loop system with three parallel loops that reject their heat to the atmosphere via the dump heat exchangers. The air atmosphere is the ultimate heat sink for the FFTF reactor and commercial electric power is not produced by the facility. The two closed test loops to be installed initially will operate in parallel with the reactor heat transfer system

out the sodium within the two systems is not intermixed. All sodium heat transfer loops removing energy from the reactor are seismically designed Category 1 for the safe shutdown earthquake, but only the east secondary sodium main heat transfer system loop is tornado hardened against missiles. The other two secondary sodium loops are at a low elevation and are sheltered by pipe tunnels mostly below ground. This physical arrangement reduces the exposure to tornado missiles. The likelihood of sodium pipe penetration by missiles is thereby reduced. Electric power is fed to the site from two offsite transmission lines.

In the event of failure of the offsite power systems, diesel generators are provided on site to carry only those loads necessary to minimize economic damage to the facility. The diesel generators normally start sequentially depending on the availability of offsite power. The diesel generators are intended only to augment or back up the offsite power sources and, although called emergency diesel generators, are not qualified as such. They are not regarded as an engineered safety feature by the Project and are not tornado hardened. Normally, with loss of both sources of offsite power, the emergency diesel generators would operate pony motors to provide an adequate level of forced convection sodium flow in both the primary and secondary coolant systems to remove decay heat.

The Project has deemed the concurrent loss of all AC power for an indefinite period to be extremely unlikely, and an acceptable risk. In the event of loss of all AC power, the safety power system (battery/DC invertor) supplies initial instrument and control function power for fifteen minutes, unloading pneumatic valve actuators and motor operated valves for containment isolation and a controlled shutdown, and then provides essential plant monitoring functions for a period of one week. Natural circulation of sodium, monitored by instruments powered by the DC IE electric system, would be relied upon for ultimate decay heat removal in the event of loss of all offsite and onsite AC electrical power. Tests to demonstrate the adequacy of natural circulation for ultimate decay heat removal have been proposed by the Project and are discussed in Section 14.2 of this report. The removal of decay heat by natural circulation would prevent loss of coolable core geometry due to accumulated heat. Without requiring pony motor pump power public safety would therefore not be jeopardized by a release of radioactivity from such a sequence of events. The one-week period during which the plant can be monitored by instruments powered by the DC IE electric system is intended to provide time to re-establish an AC power system, using temporary lines or mobile units if necessary.

The reactor vessel and primary heat transport loops are located in the containment building in independent, concrete, biologically-shielded, steel-lined cells, with a nitrogen (<2% oxygen) atmosphere to inhibit sodium fires. The reactor containment concept is a low pressure open type of steel shell containment. That is, heated air is exhausted by the heating and ventilating system through the containment building to the outside air atmosphere along with very low levels of gaseous radioactivity from the containment building that has escaped through equipment seals

during normal reactor operation. The containment building is isolated in accident situations by high radiation detection signals adjacent to the ducts of the containment building atmosphere ventilation system. Isolation of containment involves double valve closure in the heating and ventilating exhaust duct and is accomplished before the radioactive cloud in the duct reaches the containment isolation valves.

Normal driver fuel handling is provided by three in-vessel fuel handling machines, located within the reactor vessel and below the reactor vessel head. The fuel is fully immersed in sodium during the fuel transfer operation from a core position to an in-vessel storage position where it is inserted into component storage pots immersed in sodium. From there, by means of a closed loop ex-vessel machine the component storage pots containing the fuel elements are transferred to an interim decay storage vessel still in the pot full of sodium. Closed loop and instrumented open test assemblies are transferred directly to and from core positions by the closed loop ex-vessel handling machine. A bottom loading tranfer cask is used to remove the decayed fuel elements from the interim decay storage vessel and transfer them outside of the containment for examination or to a cask for shipment.

The FFTF reactor operates with an argon cover gas over the liquid sodium surface at a pressure of less than one psig. The argon gas communicates to the space above the liquid surface in the primary sodium pumps as well as in the reactor vessel through interconnecting piping. The gas spaces are all interconnected to balance the pressure between the multiple sodium surfaces. Sodium volume control within the reactor coolant system is maintained by means of an overflow tank and two electromagnetic pumps. A constant level is maintained in the reactor vessel by means of this overflow pump system compensating for sodium volume changes due to temperature change.

Oxygen control of the sodium in the reactor coolant system is maintained by means of cold traps. The cold traps maintain a low oxygen concentration in the liquid sodium to minimize corrosion of steel surfaces at elevated temperatures.

The argon cover gas purity is maintained by the radioactive argon processing system. Reprocessed argon cover gas is also used as a buffer gas for the seals in components in the reactor vessel head. Sensible heat and gamma heating generated in equipment spaces is removed and transferred from these spaces by circulating nitrogen to a water-glycol cooling system which ultimately rejects the heat to air.

The reactor service building, outside of the containment, houses most of the auxiliary services for the facility such as the radioactive gas reprocessing and storage system and temporary liquid waste storage systems. The diesel generators are adjacent to this structure. The control room is part of the substructure and is tornado-missile protected by its location. Radioactive liquid wastes are generally stored and then transported offsite at a convenient time and low level gaseous wastes are held up for decay and subsequently released to the atmosphere. The design of the FFTF is based on a 20-year life expectancy.

1.3 Interactions at the Site

Facilities of interest in proximity to the FFTF are the three Washington Public Power Supply System (WPPSS) Nuclear Project Plants to be placed in operation in the early 1980s. These plants are as follows: WPPSS No. 1, a 1218 MWe pressurized water reactor, Docket No. 50-460; WPPSS No. 2, a 1100 MWe boiling water reactor, Docket No. 50-379; and WPPSS No. 4, a 1218 MWe pressurized water reactor, Docket No. 50-513. These three plants are approximately three miles north-northeast of the FFTF, with WPPSS No. 2 being the closest.

The Department of Energy is proposing also to construct a high performance fuel laboratory approximately 1/4 mile west of the FFTF. The draft environmental statement for this fuel facility is Report Number ERDA-1550-D, dated September 1976. The stated objective for this facility is to fabricate test fuel assemblies for the FFTF from powdered fuel. It is our understanding that fuel reprocessing is not contemplated at this facility. The facility is in the planning stage, and not currently under construction.

The three WPPSS power plants will meet Commission requirements for licensed light water power reactors. Agreements exist between WPPSS and the Department of Energy regarding plant operations. WPPSS No. 2 has specified an exclusion radius of 1.2 miles and a low population zone of 3 miles. In the event of a loss-of-coolant accident, doses would not exceed the values of 10 CFR Part 100. At the boundary of the low population zone doses were calculated for a 30-day period to be 160 rem to the thyroid and 10 rem to the whole body. For WPPSS Nos. 1 and 4 similarly, the doses were calculated to be 82 rem thyroid and 9.2 rem whole body. It is believed that radiological releases from the FFTF, in the event of a postulated core disruptive accident, would not likely affect public safety or compromise the safety of the three WPPSS power plants. Only a combined sequence of low probability events leading to loss of post accident heat removal or core meltdown, with reactor cavity steel liner failure, appears to be a potential challenge to the FFTF containment. This subject is more fully discussed in Section 15.3.6 of this report.

1.4 Experience with Similar Facilities

In the following discussion we will compare some FFTF design parameters with corresponding parameters of similar reactors in order to gain an appreciation of its similarity to existing liquid metal facilities. In addition we will examine the general features of the FFTF safety requirements.

The purpose for the development and the construction of the FFTF is the testing of nuclear fuels and materials in a fast flux environment and the development of base liquid metal fast breeder reactor technology. As such, the neutron flux and its spectrum are the primary parameters of the reactor design coupled, of course, with safety of operation. For purposes of material and fuel testing and experimentation

the reactor system includes two closed loops and four open test positions within the core matrix. Accessibility and inspectability of the instrument stalks on the closed loops and other external equipment for fuel handling, experimentation and supporting activities are essential to collect experimental data.

Because the requirements of the neutron flux, the neutron spectrum and the space for irradiation and testing dictated the core design parameters, it is of interest to assess the resultant design in comparison to the bounds of existing technology. Table 1.4-1 compares some of the FFTF design parameters with liquid metal fast breeder reactors built and operated in the United States. The choice of the PuO_2-UO_2 mixture for the fuel pellet material is the technology where the bulk of the experience with fuel material properties and fabrication exists. The maximum fuel temperature, the average power density, the coolant mean outlet temperature, etc. are within existing design technology limits. In addition to the domestic experience there exists significant experience from foreign reactors. Table 1.4-2 compares some design parameters of existing and planned foreign liquid metal fast breeder reactors to those of the FFTF. An overview of Tables 1.4-1 and 1.4-2 suggests two major variants: (1) core size as it relates to safety and (2) the loop vs pool choice of reactor coolant system arrangement. The first is intimately related to the core safety characteristics and is examined in detail elsewhere in this report. We have assessed the merits of a loop type fast reactor especially in view of the opinion of part of the technical community that a pool type arrangement has some safety advantages over the loop type. Briefly, in the pool arrangement the fast reactor core and the primary coolant loop equipment (pumps and intermediate heat exchangers) are contained within a large tank which is filled with sodium. In the loop system the reactor core is contained within a relatively small reactor vessel while primary coolant loop equipment is located external to the vessel. The vessel contains relatively small amounts of sodium. The pool type reactor has a greater ability in the primary system to absorb temperature transients and decay heat due to the large heat capacity. Because large amounts of sodium surround the core container with all components located in one vessel it is virtually impossible to lose the core coolant. A leak in the primary piping system introduces sodium to the reservoir, but does not uncover the core. However, the loop type offers greater accessibility for maintenance of external equipment and for fuel handling and experimental features such as closed loops. Ease and facility of experimentation are among the primary objectives of the FFTF.

Our review of the relative merits of the pool vs the loop as far as safety is concerned for the Clinch River Breeder Reactor failed to indicate a clear advantage for either type of concept. In addition, the domestic pool type experience is limited to EBR-II which is only 62.5 MWt with metallic fuel pins. On the other hand there exists wider experience in the United States regarding loop type reactors.

TABLE 1.4-1

COMPARISON OF LIQUID METAL FAST BREEDER REACTORS IN THE UNITED STATES

	FFTF	Fermi I	SEFOR	EBR-II
Design Thermal Power, MWt	400	200[3]	20	62.5
Primary System Type	Loop	Loop	Loop	Pool
Fuel Type	Oxide	Metallic	Oxide	Metallic
Equivalent Core Diameter, cm	120	78	83	48
Active Core Height, cm	92	78	86	36
Kilograms of Heavy Metal	2,520	1,940	2,120	385
Plutonium Mass, kg	600	----	395	----
Average Power Density, MWt/liter	0.37	0.46	0.034	0.735
Average Linear Power, KWt/ft	7.35	5.35	12.3	12.3
Peak Linear Power, KWt/ft	14.1	9.05	20.8	18.6
Core Bulk Inlet Temperature, °F	830*	550	700	700
Core Bulk Outlet Temperature, °F	1,100*	822	820	900

TABLE 1.4-1 (Continued)

	FFTF	Fermi I	SEFOR	EBR-II
Maximum Clad Surface Temp., °F(1)	1,315	1,139	1,000	1,086
Maximum Fuel Temperature, °F(1)	4,650	1,209	4,550	1,271
Safety Rod Worth, $	20(2)	8.2	----	2
Total Movable Rod Worth, $	43(2)	9.1	11	7.5
Maximum Positive Na Void Reactivity, $	3.1	----	.28	----

(1) Includes Hot Spot and Hot Channel Factors
(2) Based on Fig. 3-10, FFTF PSAR, $b = .003$
(3) Core-A limitation; Ultimate Design Capability is 430 MWt.
(*) Initial operation revised 3-21-77 to 680°F inlet - 930°F mix mean outlet (design conditions shown in Table).

TABLE 1.4-2

FOREIGN REACTOR COMPARISON TO FFTF

	FFTF	PHENIX	SUPER PHENIX	PFR	SNR 300	BN 350
Design Thermal Power, MWt	400	567	2900	600	736	1000
Primary System Type	Loop	Pool	Pool	Pool	Loop	Loop
Fuel	PuO_2-UO_2	PuO_2-UO_2	PuO_2-UO_2	PuO_2-UO_2	PuO_2-UO_2	PuO_2-UO_2
Equivalent Core Diameter, in (Fuel/Blanket)	47.2/Ref	54.9/74	138/744	-	-	5/18/104
Core Height, in (Fuel/Assy)	36.0/	33.5/169	39.4/216	36/	37.4/	41.8/138
Power Density Average, KWt/liter	390	406	275	-	-	480
Linear Power KWt/ft						
Average	7.35	7.9	13.7	8.3	11.3	-
Maximum	14.1	13.1	16.4	14.0	14.0	13.4
Fuel Pin OD, in	.23	.26	.341	.23	.236	.24
Reactor Mix Mean Outlet Temperature, °F	1050	1026	1004	1044	1015	972
Fuel Pin Material	316SS	316SS	-	-	-	16 CR - 15 Ni 3 Mo - .8 N6

In summary: (1) the FFTF reactor is a type for which design and operation experience exists in the United States and (2) the choice of the loop type of arrangement seems more reasonable for a test reactor like the FFTF because, while the pool type might enhance safety, it would not allow flexibility for experimentation and accessibility which are essential test reactor objectives.

1.5 Identification of Agents and Contractors

The FFTF is a United States Government owned reactor falling under the administration of the Department of Energy, formerly the U.S. Energy Research and Development Administration. The FFTF Project Office within the Department of Energy administers the FFTF contract with the Hanford Engineering Development Laboratory operated by the Westinghouse Hanford Company, a subsidiary of the Westinghouse Electric Corporation, which by contract has overall design and operation responsibility for the FFTF. Contracts exist with the Department of Energy and various national laboratories (Argonne National Laboratory, Oak Ridge National Laboratory, Los Alamos Scientific Laboratory) to provide liquid metal fast breeder reactor base technology, and with private industry such as the Bechtel Corporation, Atomics International and Aerojet General to provide specific FFTF design engineering. The contractual agreements with the national laboratories are in general administered by the Department of Energy Headquarters, or its appropriate Field Offices, and those relating to the FFTF site construction (Bechtel) or FFTF research and development to be performed by the Hanford Engineering Development Laboratory may be administered by the Hanford Engineering Development Laboratory or by the FFTF Project.

This general arrangement has existed since the Energy Reorganization Act. Prior to that time most FFTF-related activities were directed from the Atomic Energy Commission Headquarters by the Division of Research and Reactor Development. The Westinghouse Advanced Reactors Division at Waltz Mills, Pa. initiated the procurement of the FFTF reactor vessel and head, the reactor cooling system, the in-vessel fuel handling equipment and the control and protective systems. The present and follow-on activities for the FFTF have been transferred to the FFTF Project and the Hanford Engineering Development Laboratory for the foreseeable future.

The interfacing between participants and contractors is complex but it should be emphasized that the FFTF Project Office has the delegated onsite responsibility for the safety, management, construction and operation of the FFTF under authorization by the Federal Government regardless of present and future contracts involving modifications and core changes during operation of the FFTF. The Commission is limited to an advisory role in the final disposition of safety matters regarding the FFTF.

1.6 Summary of Principal Review Matters

Our review and evaluation of the Final Safety Analysis Report and other information submitted by the Project has considered matters relevant to the safety of the FFTF design and we are providing advice to the Project in accordance with the existing interagency agreement.

Pursuant to prior agreement (reference letters R. Boyd to R. L. Ferguson dated August 13, 1976 and R. L. Ferguson to R. Denise dated August 20, 1976) the FFTF site seismology, geology and hydrology were not subject to current review. It was mutually agreed by both agencies that the physical site characteristics of the Hanford Reservation, more specifically the FFTF site, had been sufficiently reviewed during the construction review stage. It was considered unlikely that the Commission staff could make further constructive advisory contributions in this area related to the FFTF considering the stages of the review and the status of the FFTF construction.

The most recent reactors under construction in the vicinity of FFTF are WPPSS Nuclear Generating Plants Nos. 1, 2 and 4, for which site-related information and documentation exist in the public domain. Specific site documentation for the FFTF also is in the public domain. Those aspects of the FFTF site meteorology that were re-examined during the current review suggested the use of WPPSS 1 and 4 meteorology for X/Q values rather than the methods described in NRC TID 14844 and used by the Hanford Engineering Development Laboratory for the evaluation of radiological releases of the FFTF at the site boundary. TID 14844 is usually applied only at the construction permit stage of review when more detailed meteorological information is not available.

We are also cognizant of the site specific prevailing weather patterns in the vicinity of the FFTF site and of the Atomic Energy Commission 1971 ruling regarding the design basis tornado for construction of the FFTF containment structure. Additional recent studies by Dr. Fujita appear on a preliminary basis not to indicate substantial differences from this early judgment by the Atomic Energy Commission Commissioners although the amount of data added is not large. There are some other subjective positive factors in considering the case for the FFTF such as its location, relatively small size and short design life which differentiate it from a large commercial power reactor. Collectively these factors make the probability of exceeding the dose limitations of 10 CFR Part 100 for an energetic tornado at the FFTF unlikely. There is however, in our opinion, a small residual risk in the design of the FFTF in protecting only one secondary loop against missiles. It is our view that the Project position regarding General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and General Design Criterion 34, "Residual Heat Removal," represents a departure from the spirit and intent behind the guidance provided by the General Design Criteria and the defense-in-depth safety philosophy. This should not be construed to mean an unsafe situation exists.

The Project did request a variance in designing for the operating basis earthquake requirements. The Project judged it would be acceptable to the Project and to the Department of Energy to shut down the FFTF reactor for all seismic events greater than 0.05g rather than to design for the normal operating basis earthquake value of one-half the safe shutdown earthquake. We do not have a technical objection to the Project's position on this item since the FFTF is not a commercial electric power generating facility and consequently the Project's decision results only in an extended outage to the test facility and does not affect public safety. The facility is still designed for the safe shut down earthquake of 0.25g.

It is recognized that specific general design criteria and regulatory guides do not exist for liquid metal fast breeder reactors. Nevertheless sufficient guidance can be found in the existing General Design Criteria and Regulatory Guides for light water reactors to make it abundantly clear that the role of safety is intended to be dominant. We believe that the as-built FFTF design is generally above normal in the quality of components but has accepted marginal risks which do not necessarily appear unacceptable but, on a comparability basis with light water reactors, would not be thought of as in compliance with the intent of 10 CFR Part 50 with respect to tornado missile protection and emergency power.

We have reviewed the results of the Hanford Engineering Development Laboratory's reference reports over a range of transient conditions in the emergency and faulted range of operation. Independent analysis has been performed by the staff for some of the more severe unlikely postulated events such as pipe rupture, loss of heat sink, sodium fires and core disruptive accidents, all of which can be potential challenges to the containment. Secondary sodium fires, a loss-of-coolant accident at WPPSS 1, 2 or 4, or a chlorine spill on the road to the facility have the potential of affecting the habitability of the FFTF control room.

Those system transient events, within the design basis of the FFTF, appear to be adequately protected by component design or protected by a dual shutdown system actuating either or both of two groups of B_4C rods. Normal decay heat removal is by pony motor-driven pumps providing forced circulation which, in the event of loss of all AC power, reverts to natural circulation to an atmospheric heat sink. Defense-in-depth to control activity releases is provided by compartmented steel-lined cells surrounded by a low pressure steel containment system. For those most severe accident events, which are thought to be extremely unlikely but are not dismissible, sufficient containment margin capability is thought to exist to cause the impact of those events to be near the limits of acceptability to be delayed or controlled for a reasonable period of time during which emergency plans could be put into effect.

We have evaluated the Project's plans for the conduct of facility operations, including the organizational arrangements with the Hanford Engineering Development Laboratory and the qualifications for operating and technical support personnel. Although the FFTF will not be operated by licensed operators, equivalent levels of

education, training, and experience are being provided through EBR-II operator training experience, simulator training at the Hanford Engineering Development Laboratory and in-plant training provided by the Hanford Engineering Development Laboratory at the FFTF. Emergency plans have been prepared for the FFTF to be implemented in the unlikely event of an accident that might affect plant personnel or the general public. These plans are integrated into the overall planning associated with portions of the Hanford Reservation and adjacent areas.

We have evaluated the Project and the Hanford Engineering Development Laboratory quality assurance plans for the conduct of operations, training plans and emergency plans and found them to be generally acceptable except for qualifications regarding startup testing for natural circulation.

We have reviewed the design of the reactor and heat transport system, results of safety-related research and development, the applicability of ASME codes and code cases, and high temperature design criteria to assure that major components important to safety in the liquid metal systems will remain within their design envelope throughout their operating life, including the design margin capability of accommodating a postulated core disruptive event and some sodium release in containment excluding gross component failure. With respect to those portions of the facility within the design basis we reviewed those systems important to safety in accordance with existing General Design Criteria and Regulatory Guides. Departure from these criteria, guides, codes and standards has been identified. Where the Project has taken exception to use of these criteria, the Project position has been evaluated and discussed.

We have evaluated the expected response of the facility to a spectrum of postulated accidents and transient conditions, and although many unlikely design basis accidents can be accommodated, there are several unlikely events which could potentially challenge the FFTF containment system structures. These are a large continuous sodium spray release in closed equipment spaces and the unlikely loss of coolable core geometry with subsequent vessel melt-through. These postulated accidents are discussed in Section 15.0 of this report.

Sodium fires have been classified separately from the ordinary fires resulting from combustible materials found in a power plant such as organic materials, diesel and lubricating oils. The sodium fires are addressed in this report but the more conventional fire hazards analyses will be presented in a supplement to this report.

We evaluated systems provided to control the release of radioactive effluents generated by the FFTF during its normal operation. The basis of our review did not include failed fuel. It is the Project's stated intent not to initially operate the FFTF with failed fuel. However, a gaseous collection, storage and processing system is provided to process the argon reactor cover gas in any event

so that activity released offsite from this source would be minimal. Liquid and solid wastes generated by the FFTF will ultimately be transported elsewhere on the Hanford Reservation for disposal. We believe that the above proposed practices and objectives comply with the intent of 10 CFR Part 50 and the intent of Appendix I of 10 CFR Part 50 to minimize the liquid and gaseous releases from the FFTF to as low as is reasonably achievable.

The Project contractor's analysis was reviewed and independent studies of the functional behavior of the reactor for sodium spills in the reactor cavity, intermediate heat exchanger cells and sodium overflow tank were performed. The staff analysis postulated piping failures with steel liners separating sodium and concrete and evaluated the cell pressures and the effect on the reactor. The integrity of the reactor coolant system, including the three primary piping loops, has been evaluated considering its design, fabrication, quality assurance and need for inservice inspection. Specific advice to the Project related to piping integrity is provided in Sections 4.0, 5.0 and 19.0 of this report.

A major matter under staff review has been the proposed reliance on natural circulation for decay heat removal in the event of a total loss of power. Flow in each of the primary or secondary sodium loops is driven normally by large sodium pumps. There are also pony motors on each of the primary and secondary sodium loop pumps for use during low coolant flow conditions. The electrical power supply to the pony motors is independent of the power supply to the large pump motors. However, neither power supply is designed to withstand the associated effects of a tornado. Thus, as a redundant mechanism for removing decay heat following a loss of power event, the Project has proposed to rely on natural circulation, and the heat transport system has been designed to make use of the thermal effects (i.e., density changes) in driving the natural circulation flows. Section 5.5 of this report discusses our evaluation of the decay heat removal system more fully.

The subject of fuel failure propagation potential has been assessed also and is more fully addressed in Section 15.0 of this report. Fuel handling, electrical power systems, heating, ventilation and miscellaneous auxiliary service and gas systems have also been evaluated and discussed in their respective sections of this report.

Staff emphasis has been placed on very low probability accident situations having the highest potential for a serious radiological release. We conducted a significant independent review and analysis in the area of initiation of core disruptive accidents and the more likely path of such events, and have performed a parametric analysis of the consequences of such events should they not terminate in the manner hypothesized by the Project. The objective of these parametric evaluations is to assure that under unfavorable circumstances the existing containment, with ancillary features if necessary, will provide sufficient time to implement emergency plans

should it become necessary to minimize the risk to the public. These evaluations are discussed in Section 15.0 of this report.

1.7 Facility Modifications and/or Changes During the Review

Several construction changes were identified in the initial phases of our previous review in Supplements 1 and 2 to the Safety Evaluation Report. In the current phase of the review our technical assessment has been directed primarily to providing the Project with advice derived from our experience with varied types of commercial and test reactors. Significant FFTF modifications, completed, planned or contemplated, are as follows:

(1) The Project has initiated inservice inspection development work for high temperature ultrasonic transducer heads with silicon oil used as a couplant for use at a 400°F reactor shutdown temperature. Removable piping insulation sections were to be provided by the Project at selected welds in the hot crossover piping of the reactor coolant system for periodic inspection. It is not clear that this prior commitment is still valid. We now understand that the Project proposes to inspect secondary sodium system welds instead. The staff does not agree that this alternative proposal is acceptable.

(2) We have recommended that at least one sodium leak detector system, preferably a sodium ionization aerosol type, be seismically qualified and furnished with Class IE power. This is consistent with Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection System." It appears to the staff that the sodium ionization detector offers the best chance of meeting leak detection sensitivity and response time requirements. The Project is willing to provide the leak detector but not Class IE power. This issue is unresolved.

(3) Modifications were made to the protective system control panels in which instrument control electrical separation criteria had not been fully satisfied. We find the modifications acceptable.

(4) Containment vessel penetrations which had not met the required normal, seismic and accident environmental qualifications were modified. The method of protection of electrical power penetration is still under review.

(5) Reactor cavity vents were added to vent steam and vapor behind steel liner plates in the upper portion of the reactor cavity and prevent liner plate failure due to steam back-pressure from a sodium release into the reactor cavity. This change was recommended in Supplement 2 to our Safety Evaluation Report.

(6) As part of our evaluation at the construction review stage we advised the Project against filling the reactor subcavity with ordinary concrete since

this would foreclose all other options for using this space to improve safety margins if the need became apparent at a later date. This space still exists except that the 36 inch thick concrete hole has been filled with concrete and covered with a steel liner.

(7) We recommended at the construction review stage that anti-splash features be provided in the reactor head access area to minimize the amount of sodium spray ejected into the upper reactor containment building following an energetic core disruptive accident. The objective was to confine the released sodium to a pool fire in the head access area. The recommended devices were incorporated by the Project.

(8) The quality assurance program was modified to require more formalization of quality assurance polices and procedures and to require more participation and review by the Hanford Engineering Development Laboratory management. The Hanford Engineering Development Laboratory management has agreed to participate more extensively in the planning of experimental tests and their safety review.

(9) The Project has agreed to provide safety grade, post-accident radiation monitors outside containment.

(10) By mutual agreement with the Project we have temporarily discontinued our review of the interaction of the core with an experiment in a closed loop and of the capability of the closed loop plant protection system to protect the reactor. This decision is based partially on our review of HEDL TC-1059, "Evaluation of the CLIRA Outlet Temperature PPS Function for Design Loss of Flow Events." Potential problems appear to exist in the existing design.

(11) The staff has recommended installation of a loose parts monitoring system when developed. The Project has indicated its intention to do so as soon as a system is developed and is practical to install. An experimental program is underway at Argonne National Laboratory.

(12) Due to certain problems existing with tornado missile protection, the staff has recommended shutdown of the facility when potential tornado conditions are detected in the immediate vicinity of FFTF. The Project has committed to do this. The commitment alleviates a number of staff concerns regarding the use of non-Class IE power for safety-related functions.

(13) The staff had recommended the installation of radiation, sodium aerosol, and chlorine detectors, powered by Class IE power and missile protected, ahead of the air intake to the control room, greater separation between air intakes, and inclusion of charcoal absorbers in the control room heating and ventilation system.

The Project has agreed to install the radiation and sodium aerosol detectors on Class 1E power, but the chlorine detector will be placed on control power. The Project's rationale is that the chlorine does not originate at FFTF and no credible mechanisms can be postulated for a loss of control power concurrent with an offsite chlorine release. The existing distance between air intakes, which are missile protected, is being changed to ~320 feet. The Project sees no need for a charcoal filter since the iodine potential at the control room is small, originating from offsite sources, and the control room will be isolated. Tentatively, this appears to be acceptable to the staff.

(14) The staff recommended the use of a fuel channel exit temperature of 1670°F for safety analysis rather than the proposed 1705°F. The staff concern is related to the fact that a significant fraction of the pressure drop across the fuel subassembly is along the long fission gas chamber in the upper half of the fuel pin. The fuel channel exit temperature is thus not a good indicator of the fuel and clad temperatures in the power production region. We believe it to be more conservative to use the lower value until better data is available. The Project has agreed to this recommendation.

Plant modifications initiated by the Project are identified in Table 1.3-1 of the Final Safety Analysis Report. Twenty-seven changes were listed by the Project contractors along with the reasons for the changes. These changes were reviewed during the normal course of the safety review. We believe the reasons given in some instances deserve some further comments:

(1) Hardening of the diesel generators against tornados was eliminated. We view this decision to have been a departure from the intent of 10 CFR Part 50 and an acceptance of an increase in risk, not necessarily unsafe but certainly not improving the safety of the FFTF. The acceptance of this design decision is coupled to the demonstration and verification of natural circulation, which at this time remains an open safety problem until satisfactory acceptance tests have been completed. The Project has been made aware of our concerns.

(2) Tornado missile protection was provided for one secondary loop only. We believe it would have been prudent to have added the same protection to a second loop, although this was not done. We do not view one missile-hardened loop as meeting the intent of the design criteria in 10 CFR Part 50. We recommend that operation of the facility not be continued with any perturbation of the flow in the one hardened loop, such as erratic behavior of the primary check valve. Two loop operation is outside the scope of the staff's review, but we have considered the safety consequences of accidental loss of one or two loops.

1.8 Requirements for Future Technical Information

(1) Natural Circulation - We have conditionally accepted the premise that there is reasonable assurance that natural circulation will be demonstrated as a viable method of removing decay heat. The normal decay heat removal method in the FFTF is by forced circulation through the use of pony motors driven by either normal AC power or, in the event of loss of both sources of offsite power, by the diesel generators. The two offsite sources of power and the onsite diesel generators are the sources of AC power available to drive the pony motors. Although the principle of natural circulation is not in dispute by the staff, the uncertainty regarding natural circulation relates to transient thermal behavior and also involves uncertainties in low flow behavior in the core during the transition to stable natural circulation, hot channel factor values, and the fuel cladding temperature and its time history in the core during this whole process. We also have some concerns regarding re-establishment of natural circulation, if once lost, in a timely manner, since multiple interconnected loops are involved and the thermal inertia may cause sluggish delayed response in overall system heat transport. For these reasons, which are safety-related, we have requested that the Project furnish us with the test specifications to be implemented during the startup phase of testing. The objectives of these tests are to validate natural circulation as an adequate, safe and reliable method of decay heat removal without resulting in core voiding (boiling) or fuel cladding failure and to validate the models used in the thermal-hydraulic analysis. It is our opinion that lack of analytical predictability by the models would leave doubt regarding the adequacy of the tests. In support of this validation effort we have requested substantive verification of the mathematical models of the IANUS and FLODISC codes. This includes pretest prediction of the core behavior under a range of test conditions such that the computer models can be used to predict the core thermal-hydraulic behavior during flow coastdown, through flow transition, and into the stable natural circulation flow regime. Effort should be made to pretest system behavior and collect experimental data over as wide a range of plant conditions as possible in order to verify the applicability of these models to flow transients not associated with natural circulation (e.g., accident analyses).

We have recommended that the Project submit for staff review the results of these pretest computer code analyses predicting the best estimate FFTF system and component transients for the various natural circulation tests to be performed. These results should be submitted to us prior to the actual performance of the tests. The reduced test data should also be submitted promptly for our review, along with the results of the Project's review and analysis of these data.

(2) Containment Adequacy for Core Melt Events - Postulating various core meltdown scenarios, we have evaluated the FFTF containment margins utilizing state-of-the-art experimental information from the Department of Energy and Commission confirmatory research programs, as well as other applicable information in this area. The FFTF contractors' computational models were also reviewed and used by the staff. The amount of data regarding core debris (sodium, fuel, steel) interacting with containment materials (steel liners, concrete) is sparse. We request that work be continued in this area to improve the confidence for predicting the behavior of such material interactions in the containment system. We further request to be kept informed of all Project efforts in this regard.

(3) Leak Detection - We have requested demonstration of the viability of large scale gaseous (or aerosol) sodium leak detection system during the preliminary startup phase of the FFTF. More specifically, we request a demonstration of system function and scale up of ionization detectors and differential pressure detectors in the as-built plant. Presumably this can be done with tracer gases in the large intermediate heat exchanger cells or reactor cavity volumes and compared to the small-scale laboratory tests performed at the Liquid Metal Engineering Center. We also request demonstration of on-line in-plant testability for the above leak detectors. We have requested that we be provided with the results of this simulated demonstration of sodium leak detector function.

(4) Inservice Inspection - We have recommended that the Project perform preservice baseline inspections of, and make provisions for inservice examination of, selected welds in the high stressed elevated temperature reactor coolant system crossover piping. This will require removing segments of insulation, installing tracks on the hot leg crossover piping for power-driven ultrasonic detectors, and performing a reference ultrasonic examination at 400°F. The response by the Hanford Engineering Development Laboratory to this request is that the necessary equipment will not be sufficiently developed by plant startup but that they will continue to develop pitch-catch ultrasonic test heads, couplants, computer indexing and data resolution for a defect image such that it may be located and compared to previous images to determine changes. The staff requests to be kept informed of the progress of this work and recommends that this effort continue to be supported and implemented prior to power operation or as soon as possible thereafter.

(5) Cell Liner Test Plan - We have considered the steel liners for the reactor cavity, the primary system pipe tunnels, the primary heat transport system cells and the primary sodium system overflow tank cell to be important to plant safety. In the event of a large hot sodium leak, a potential challenge to containment integrity will exist if gross failure of the steel liners occurs and results in a large sodium-concrete reaction. The quantities of

sodium involved can range from relatively small amounts to 100,000 pounds. We believe that some form of a periodic statistical sampling method can be devised by the FFTF designers to demonstrate the continued functional capability of these steel liners during the 20-year life of the facility. As the reactor cavity is inaccessible after startup of the reactor while the other cells are not, different liner functional integrity verification methods may be considered for different liners where access is limited. We are not yet convinced that tracer gases, oxygen diffusion or pressure decay testing should not be considered meaningful for the reactor cavity. We recognize that it is unlikely that tests could be devised to locate small leaks in the reactor cavity, and it is unlikely that such leaks could be repaired. In the event of a sodium spill, it may become evident that only large liner leaks are important if small cracks are found to be self-plugging as recent tests have indicated. In any event we request to be kept informed of the Project's plans and tests to demonstrate the continued functional capability of the above liners for the life of the facility.

(6) Loose Parts Monitoring - We have recommended that the Project install a loose parts monitoring system. We understand it is currently beyond the state-of-the-art but we encourage its development and installation in the FFTF. We request to be informed of progress at Argonne National Laboratory with regards to accoustical monitoring devices in sodium and of the planned incorporation of such a device in FFTF.

(7) Containment Adequacy for Core Melts - We request to be kept informed of the Project's plans for minimizing radioactive releases in the event of a low probability core melt event particularly in regard to the potential hydrogen generation problem.

1.9 Summary of Outstanding Issues

The major outstanding issues on the FFTF can be grouped under 11 major headings even though each heading has multi-complex subissues within it. The identity of these issues is described below. Following meetings between the Project and the staff on July 25 and 26, 1978, it is probable that resolution will be forthcoming, pending adequate documentation for items 1, 4, 5, 6, 7, and 8.

(1) Natural Circulation - We will review and evaluate the adequacy of information obtained from the natural circulation test program regarding both natural circulation and computer model verification and the Project's analysis of these data in support of the position that natural circulation is a viable option for decay heat removal. We will advise the Project of our findings (see Section 14.2 of this report for a more detailed discussion).

(2) Containment Adequacy for Core Melt Events - In Supplement 2 to the Safety Evaluation Report, issued in March 1975, we evaluated the capability of the reactor coolant system boundary to contain an in-vessel core meltdown accident. In the event of failure of in-vessel post accident heat removal we also expressed our concern regarding the containment margins for a postulated core melt-through accident. We did agree that this was a low likelihood event but that the residual risk was not dismissible. In Item 4, Conclusions and Advice, of Supplement 2 it was recommended that the Project contractors evaluate the best method for reducing the consequences of early containment failure times and develop emergency plans. We suggested that scrubbers, filters and iodine traps in the building exhaust system be examined by the Project contractors to reduce the radiological consequences. It was also suggested that vents for the reactor cavity liner be installed and an emergency plan for FFTF be prepared. These items were subsequently addressed by the Project. The Project contractors have argued that the staff was treating core meltdown as a design basis accident and in a pessimistic and very conservative manner rather than as a mechanistic or more likely course of events. The principal differences at this point in time between the Project's evaluation and ours is the treatment of the sodium/concrete and fuel-steel/concrete attack rates and the manner in which radioactivity is released. We advocate consideration of a controlled release rate and discuss this approach further in Section 15.3.7. We also urge further consideration of the problem of hydrogen accumulation, particularly with regard to any information that may be developed as a result of large-scale tests.

(3) Piping Integrity - With respect to primary coolant system piping integrity, the staff in its letter of June 11, 1976 to Mr. R. L. Ferguson conditionally accepted the Project's position for the cold leg piping integrity in Appendix B of the Final Safety Analysis Report with several qualifications. The following qualifications were stated in our recommendation:

(a) Perform preservice ultrasonic base line inspection of the primary heat transport system wherever possible, considering the status of construction.

(b) Provide access for volumetric inservice inspection of selected highly stressed welds.

(c) Model a full size mock-up of the reactor and guard vessel to demonstrate the sensitivity and reliability of the sodium leak detection systems in an operating situation and provide a method of periodic verification of system operability.

(d) Provide periodic reactor coolant systems integrity verification capability by using material surveillance specimens or by removing a highly stressed pipe section or fitting for metallurgical examination.

The Project's response to our recommendations was provided in Supplement 2 to the Final Safety Analysis Report. The response in substance was:

(a) Preservice inspection would be performed prior to reactor startup if the ongoing ultrasonic test development program is successful. This response was later modified to state that the necessary equipment will not be developed before plant startup.

(b) Access to six high stress welds in the hot crossover piping is being provided with removable insulation. This response was later modified to state that this will not be done.

(c) It is not feasible to mock up the reactor and guard vessel full size. The Hanford Engineering Development Laboratory will continue development of aerosol leak detector testing to demonstrate its adequacy. Prior to startup an installation-as-built continuity test will be performed with helium to demonstrate functional continuity. Backup leak detection is also provided by the H_2 and Na^{24} detection systems. The staff finds this acceptable.

(d) Instead of cutting into the heat transport system, alternate material samples will be taken from in-reactor components periodically removed for metallurgical testing. We recommend that, in the event any component is removed from the heat transport system, it be subjected to metallurgical examination.

Our recommendation to the Project reflects the concern that confidence in the overall heat transport system piping integrity is predicated on the assumption that little degradation of the reactor coolant system will occur during the life of the facility. Assurance is to be provided by periodic inspection and early detection of small leaks. We still are of the opinion that this rationale is valid and further urge that at least one aerosol leak detection system, preferably the sodium ionization detectors, be seismically qualified and powered by Class 1E power. The guidance provided in our letter of June 11, 1976, evaluating Appendix B of the Final Safety Analysis Report, remains unchanged in principle. We urge that conventional preservice inspection be performed and that provisions be made for later inservice inspection, with implementation of advanced techniques as soon as practical.

(4) Cell Liner Integrity - In our letter of August 9, 1977, we requested additional information and stated that our preliminary evaluation of major sodium spills indicated that there existed a potential challenge to containment if liner failure either existed before or occurred during a major sodium spill and significant quantities of sodium and concrete surface areas come in contact with each other. Independent tests conducted at Sandia Laboratory

have indicated sodium concrete reaction rates considerably greater than those being used by the Hanford Engineering Development Laboratory. We, therefore, requested that plans be proposed by the Hanford Engineering Development Laboratory to demonstrate the functional capability of the steel liners throughout the life of the facility for both the reactor cavity and the primary heat transport system cells.

The Project responded in Supplement 17 to the Final Safety Analysis Report indicating that cavity and leak rate tests would be performed prior to sodium fill, that a test would be performed at 0.25 psig and that the permissible leak rate would be 1 volume percent per day. No tests other than this are being planned. The Project's justification for this position is the apparent success of the LT-1 test. We acknowledge that the LT-1 test was a successful, but limited, demonstration. We are concerned about assuring that the functional capability of the steel liner is maintained over the life of the facility; therefore, we still recommend that the Project include in the technical specifications a periodic verification test plan. We consider that it is the Project's responsibility to propose an adequate program to assure the continuing functional capability to place reliance on cell liners and welds during the life of the facility for the unlikely event of a major sodium release accident. We were advised on July 25, 1978 that such plans are in preparation.

(5) Loose Parts Monitoring - We believe the FFTF should have a method of detecting loose parts in the reactor coolant system and recommend the addition of such a system to the FFTF. We understand that the Department of Energy is developing such a device at Argonne National Laboratory and we encourage continuation of its development. The Project has stated its intention to install such a device when it is developed.

(6) Maximum Fuel Channel Exit Temperature - We based our evaluation on the presumption that no boiling will be allowed in the core. Considering the uncertainties in hot channel factors, flow under natural circulation and the scarcity of experimental data, we recommend that the maximum fuel channel exit temperature be 1670°F rather than 1705°F. The Project has indicated acceptance of our recommendation.

(7) Tornado Conditions - Our view that the implementation of the modified design to satisfy tornado criteria (Table 1.3-1, Final Safety Analysis Report) represents a departure from the spirit and intent behind the guidance provided by the General Design Criteria and the defense-in-depth safety philosophy. We view that recovering from a planned shutdown is a more orderly process than recovering from an unplanned shutdown in which debris may obstruct the access of plant operators and prevent certain operations from being carried out in a timely manner. We therefore recommend shutdown of the facility when severe

weather conditions exist which are conducive to the formation of a tornado. We have been advised that the Project is in accord with this principle.

(8) Containment Isolation Valve Position Indicators - It was the staff's opinion that the containment isolation valve position indicator transmitters should be protected, including the wiring, and powered by Class IE power, and that the valve positions should be displayed in the control room for operator information at all times. Since, as mentioned earlier, the Project intends to bring the plant to an orderly shutdown in the event of conditions conducive to tornado formation, we have modified our position and find the proposed system design, including the use of control power for position indication, acceptable pending demonstration that the use of control power will not cause false signals to occur.

(9) Control Room Isolation - We believe that the control room air intakes should be protected against tornado missiles, including safety-related instrumentation and wiring. Class IE power should be provided. Safety-related instruments are considered to be the radiation, sodium oxide and chlorine detectors. Recently we have been advised that the above, except for the chlorine detector, are on Class IE power. We will address this in a supplement to this report.

(10) Fire Hazards Analysis - The fire hazards analysis was not submitted initially with the Final Safety Analysis Report. Our review of this subject has been delayed and will be addressed in a supplement to this report.

(11) Closed Loops - The closed loop review has been deferred until the design is completed and will be addressed in a supplement to this report.

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

The Fast Flux Test Facility is located on the Department of Energy's Hanford site which is situated in the south-central portion of the State of Washington. The distance to the nearest boundary of the Hanford site is 4-1/2 miles east of the FFTF. Figure 2.1-1 illustrates the location of the Hanford site in the state and with respect to nearby centers of population.

The Department of Energy has established a 1-1/2 mile control radius surrounding the plant and states that at this time there are no activities within that area which are unrelated to operation of the FFTF. Highway RT-4, which is owned by the Department of Energy, traverses the control area about one mile northeast of the reactor.

Population surrounding the area is sparse, and the Project indicates no residential population within 5 miles. Figure 2.1-2 illustrates the cumulative 1970 population within 50 miles and Table 2.1-1 shows the 1970 and projected population to the year 2000 within 10 miles of the site as provided by the Project. The nearest centers of population are the communities of Kennewick, Pasco, Richland, and West Richland which lie south and east of the site. The nearest residential area of Richland begins about 8 miles south-southeast of the site. Table 2.1-2 lists the 1970 population of these communities and their projected growth to the year 2000.

Based on our review of the location of the FFTF within the Hanford Reservation, the controls exercised over access to the reservation as well as the site, and the location and distribution of population around the site, we conclude that with respect to these characteristics the site is well suited for an experimental reactor such as the FFTF.

2.2 Meteorology

To ensure that safety-related plant design and operating bases are within Commission guidelines requires an evaluation of regional and local climatological information, including occurrences of extremes of climate and severe weather which may affect safe design and siting of a nuclear power plant. To determine that postulated accidental and routine operational releases are within these guidelines requires an evaluation of the atmospheric diffusion characteristics in the vicinity of the proposed site. Our evaluation and description of the meteorological

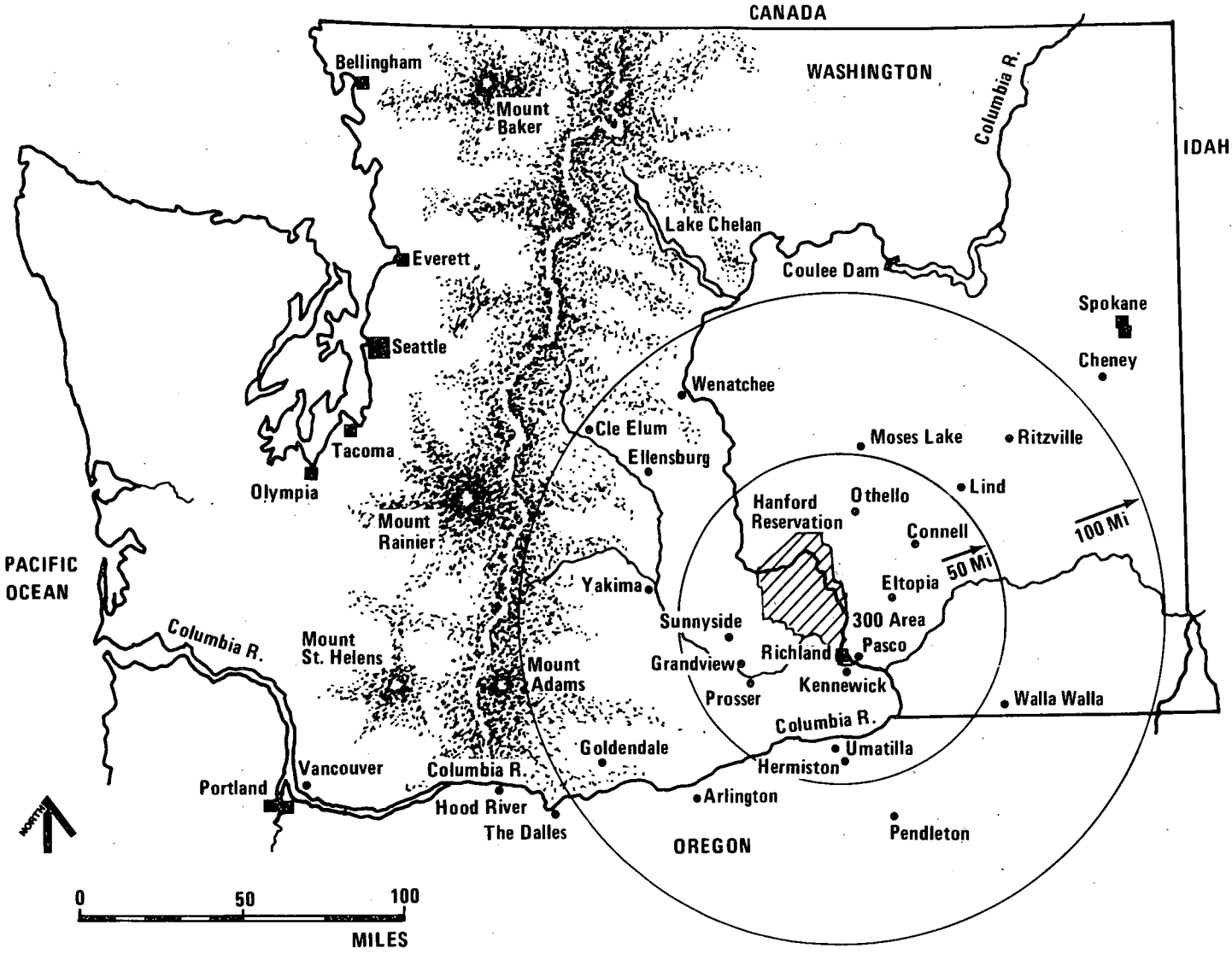


Figure 2.1-1 Location of Hanford Reservation

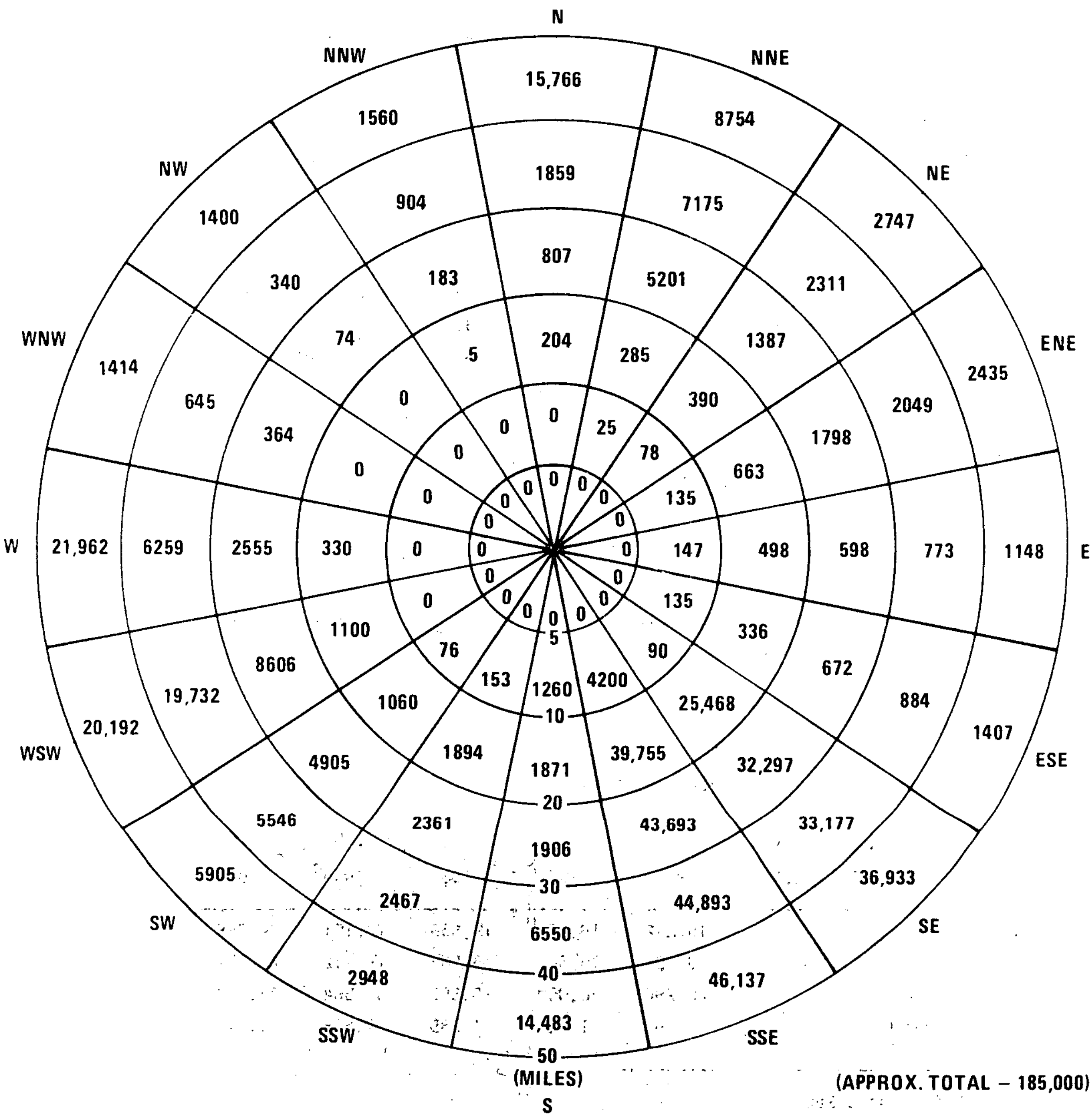


Figure 2.1-2 Cumulative Population Distribution About FFTF (1970 Census)

TABLE 2.1-1

PROJECTED POPULATION WITHIN 10 MILES OF FFTF SITE[a]

DIRECTION	1970	1980	1990	2000
N	0	0	0	0
NNE	25	27	30	33
NE	78	84	95	104
ENE	135	145	164	180
E	147	158	178	195
ESE	135	145	184	180
SE	90	97	109	119
SSE	4200	4523	5106	5591
S	1260	1357	1532	1717
SSW	153	165	186	204
SW	76	83	92	101
WSW	0	0	0	0
W	0	0	0	0
WNW	0	0	0	0
NW	0	0	0	0
NNW	0	0	0	0

[a]Projection based on percent (%) forecast established by State of Washington for population growth rate through the decade of 1990's.

TABLE 2.1-2

POPULATION PROJECTIONS

TOWN (INCORPORATED CITY LIMITS)	1970[a]	1974[b]	1980[c]	1990[c]	2000[c]
Kennewick	15,212	18,234[d]	19,638	22,191	24,299
Pasco	13,920	16,565	16,565	19,712	23,457
Richland	26,290	28,000	28,393	30,664	33,117
West Richland	1,107	1,247	1,389	1,556	1,745

[a]1970 U.S. Census Report

[b]"State of Washington Population Trends 1974," Office of Program Planning and Fiscal Management, Olympia, Washington, August, 1974.

[c]State Highway Department Projections Based on Benton-Franklin Governmental Conference.

[d]"1975 Special Census for the City of Kennewick," City of Kennewick Planning Department, Kennewick, Washington.

characteristics of this site followed the procedures outlined in Sections 2.3.1 through 2.3.5 of the Standard Review Plan.

2.2.1 Regional Climatology

The Columbia Basin of western Washington has a mild, dry climate as a result of frequent incursions of maritime polar air from the Northern Pacific Ocean into a basically continental steppe type climatic area. Although maritime polar air masses are predominant over the basin, the air usually loses most of its moisture in ascending the western slopes of the Cascades and is warmed in descending the eastern slopes. Occasionally continental air penetrates into the basin from the interior of the continent. These continental air masses are responsible for the large annual range of temperatures in the region and also can cause large diurnal temperature ranges. Temperatures of 90°F or higher may be reached on about 50 days annually over the southeastern Columbia Basin, while temperatures below 0°F may be expected on four days annually. Temperatures will be below 32°F on about 120 days each year, and on about 20 of these days the temperature will not rise above the freezing mark.

Since the basin is in the "rain shadow" of the Cascades, rainfall is sparse in the region, averaging only six to eight inches annually. The annual snowfall averages from about 10 to 20 inches over the basin. The FFTF is designed for a snow load of 20 pounds per square foot. Based on the regional historical data, we consider that this value is reasonable for the FFTF site. Hail may occur one day each year, while freezing rain may occur seven days annually.

Although infrequent, thunderstorms, dust storms, and tornados can affect the site area. Thunderstorms occur about 10 days annually, predominantly between May and August. Between 1940 and 1970, dust or blowing dust and sand reduced visibility to under seven miles about 20 hours annually.

Table 2.2-1 lists the design basis tornado parameters for which the FFTF is designed and the tornado parameters we estimated by our site-specific analysis based on the methods outlined by Markee, et al (1974). For comparison, this table also lists the design basis tornado parameters for Regulatory Guide 1.76 Tornado Intensity Region III (in which the plant is located). These values are generally considered when no site-specific analysis has been done.

During the construction review for the FFTF, the Atomic Energy Commission Regulatory Staff and the Advisory Committee on Reactor Safeguards concluded that tornado criteria then established for the eastern United States should also be applied to the western part of the United States. These criteria included a tornado with a 300 mph tangential speed plus a 60 mph translational speed. The FFTF Project presented information supporting a view that tornados were less severe in the Hanford area and recommended that the tornado should be 175 mph (150 mph tangential

TABLE 2.2-1

DESIGN BASIS TORNADO CHARACTERISTICS FOR THE FFTF SITE

The values listed are those estimated for the FFTF site using different analytical methods. The column labeled "FFTF Design" are those for which the plant is designed and supported by Fujita (1976). "NRC Site-Specific" values are our estimates using the methods of Markee, et al (1974). For comparison "R.G. 1.76 Region III" values are included; we consider that these latter values are acceptable for a site in this region for which no site-specific analysis has been done.

Tornado Parameter	FFTF Design	NRC Site Specific	RG. 1.76 Region III
Maximum Speed (miles/hour)	175	215	240
Rotational Speed (miles/hour)	150	170	190
Max. Translational Speed (miles/hour)	25	45	50
Total Pressure Drop (pounds/inch^2)	0.75	1.0	1.5
Rate of Pressure Drop (pounds/inch^2/ second)	0.25	0.4	0.6

speed plus 25 mph translational speed). Disagreements between the two staffs centered on the applicability of eastern United States data at Hanford in view of its specific geographic and meteorological setting and the reliability of the data for western tornados. In addition, the FFTF Project qualitatively acknowledged the capability of proposed facilities to survive more severe tornados than the design bases they proposed. Based on these discussions the Atomic Energy Commissioners on November 19, 1971, concluded that the Project arguments had merit and that the proposed 175 mph criteria was acceptable.

Following the Commissioners' decision, Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," was issued in April 1974. This guide regionalized tornado design parameters for the United States. The guide also allowed a site-specific analysis if sufficient local information were available.

In support of their Final Safety Analysis Report, the Project supplemented their tornado analysis for the construction review with a report by Fujita (1976) which updated his original review of the site (Fujita 1970). Fujita (1976) added six years of data and assessed tornado occurrences between 1950 and 1975 within the region. He used two methods to estimate tornadic winds as a function of probability of occurrence in the region. His conclusions from the updated study supported the design basis tornado parameters which the Atomic Energy Commissioners approved and for which the FFTF is designed.

Our review of the Fujita analysis still identified several concerns:

(1) It uses the "DAPPLE" method for computing tornado probabilities. This method is based on data from a tornado outbreak of April 1974 in the eastern United States. This method has not been submitted for generic review, nor has it been adopted for tornado intensity determinations.

(2) Use of a more "refined" method for estimating tornado probabilities (such as the "DAPPLE" method) may not be justifiable considerng the present uncertainties of tornado reporting. Such uncertainties include whether most of the tornados which have occurred in the site vicinity have been observed, whether their sizes have been estimated correctly, and whether their intensities have been properly classified.

Because the Project supplied regional tornado information, we calculated estimates of the tornado parameters specifically for the FFTF site. Due to our concerns about the Project's analytical method, we have kept our analysis method consistent with that outlined by Markee, et al. (1974). We used the tornado data in the Fujita (1976) report. We consider that the data base outlined by Fujita is the best available for the region and is consistent with our analysis method.

Our analysis considered only the region which we concluded to be meteorologically and topographically homogeneous within a 150-mile radius of the site (generally the flatter terrain east of the Cascades). Twenty-two tornados occurred in this region over the 26-year period of record. The expected path area of a tornado in this region was calculated to be 0.5 square miles. Thus the recurrence interval for a tornado at the plant site is calculated to be 120,000 years.

Table 2.2-1 lists our tornado parameter estimates, labeled "NRC Site-Specific." A tornado of the severity of our design basis tornado has a probability of occurrence of approximately 10^{-7} per year. Using our estimate as a base, we conclude that a tornado the size of the design basis tornado for which the FFTF is designed has a probability of occurrence of about 4×10^{-7} per year.

Based on regional data, we consider that the operating basis wind speed (defined as the "fastest mile" wind speed at a height of 30 feet with a return period of 100 years) of 100 miles per hour, for which the FFTF is designed, to be reasonable for this site. The highest wind speed recorded at elevations representative of the plant site on the Hanford Reservation meteorology tower has been a gust to 80 miles per hour (January 1972).

We conclude that the FFTF Final Safety Analysis Report has sufficiently described the regional climatology and severe weather phenomena which are important to the design of the facility.

2.2.2 Local Meteorology

To assess the local meteorological characteristics of the FFTF site, we evaluated climatological data that have been recorded in the Hanford Reservation area since 1912. Data similar to those recorded at National Weather Service stations have been observed since 1945, with the primary observation station located about 16 miles northwest of the FFTF site. Data are also available from the WPPSS nuclear plant site located about 2-1/2 miles north-northeast of the site (WPPSS, 1974). These data are reasonably representative of conditions expected at the site.

In the site area, average daily maximum and minimum temperatures range between 90° and 61°F in July, the warmest month, and between 37° and 22°F in January, the coolest month. The extreme maximum temperature recorded was 115°F (July 27, 1939); the extreme minimum temperature has been -27°F (December 12, 1919).

Hanford receives about 6.3 inches of rain annually. November, December, and January are the "wettest" months, totalling an average of 2.6 inches. For July, August, and September, the combined average is less than 0.7 inches. The maximum 24-hour rainfall recorded has been 1.9 inches (October 1957). Snowfall averages 13 inches per year, with 7.1 inches being the greatest snowfall over a 24-hour period (January 1954). Heavy fogs (visibility of 0.4 kilometer (1/4 mile) or less) occur 24 days annually, with 80 percent of the daily occurrences recorded between November and January.

For the two-year period of April 1974 through March 1976, the windflow over the site, as measured at the 10-meter (33-foot) level of the WPPSS meteorological tower, occurred about 60 percent of the time from the south- and north-westward directions. Figure 2.2-1 shows the directional frequency of the site vicinity winds. Winds were calm, that is windspeeds less than 0.22 meters per second (0.5 mile per hour) less than 0.1 percent of the time at the 10-meter (33-foot) level.

We conclude that sufficient information exists to describe conditions which are important to the safe design of the facility.

2.2.3 Onsite Meteorological Measurements Program

The onsite meteorological measurements program for the FFTF site consists of measurements of wind speed and direction at 13 feet above ground level on a mast 1000 feet east-northeast of the containment building. The data from this system are not sufficient in themselves to use in estimating atmospheric diffusion for the site; however, other measurement programs exist on the Hanford Reservation which can be used to estimate atmospheric diffusion for the FFTF site.

In our analysis we have used data from the meteorological measurements program of the WPPSS 1, 2 and 4 site. This program consists of a 245-foot high tower located

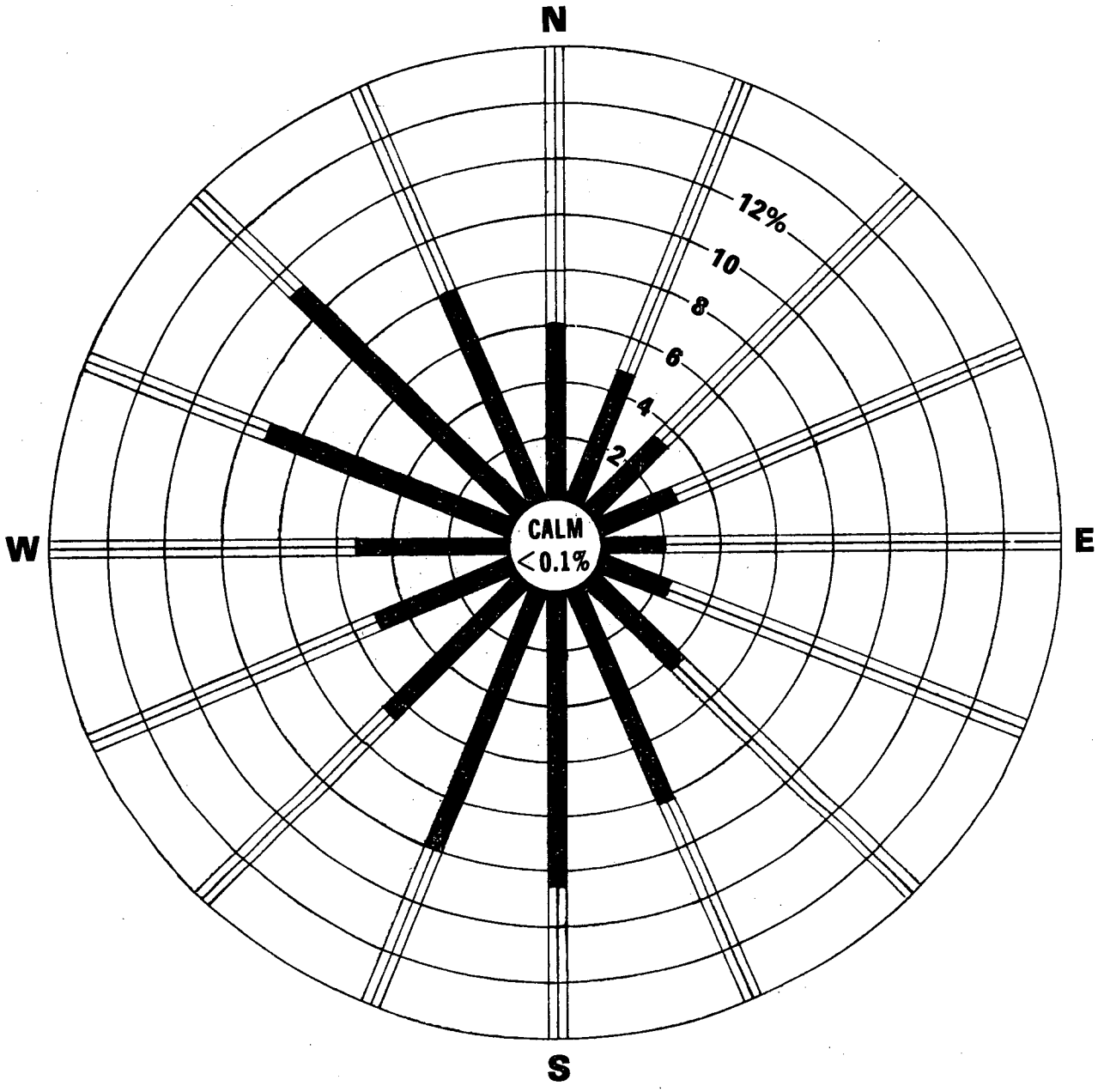


Figure 2.2-1 Directional Frequency of Wind – WPPSS Site Data at 33 feet above ground level, April 1974 through March 1976. Bars show the direction from which the wind blows. Calms are those winds with hourly average speeds less than 0.22 meters per second (0.5 miles per hour). This site is located 2½ miles north-northeast of the FFTF facility.

about 2-1/2 miles north-northeast of the FFTF. Table 2.2-2 list the parameters measured and the measurement levels on the tower (WPPSS 1974). This program meets the regulatory position contained in Regulatory Guide 1.23, "Onsite Meteorological Programs."

To calculate our atmospheric diffusion estimates, we used the joint frequency distributions of wind speed and direction by atmospheric stability class, based on the vertical temperature gradient, collected at the WPPSS site during the period April 1974 to March 1975. The distributions were for wind speed and direction measured at the 33-foot level with the vertical temperature difference between the 245- and 33-foot levels. The joint data recovery data for stability and wind was 94 percent.

Additional meteorological measurements are available from the Hanford main meteorological tower (located about 16 miles northwest of the site) and from a network of remote stations around the Hanford Reservation. We consider that use of the measurement programs available on the reservation can describe the meteorological conditions necessary to estimate diffusion conditions at the FFTF site.

TABLE 2.2-2

WPPSS 1, 2, AND 4 METEOROLOGICAL TOWER INSTRUMENTATION

(Data from this program were used to evaluate atmospheric dispersion for the FFTF site)

Measured Parameter	Elevation Above Ground, Feet
Wind Direction and Speed	33,245
Dry-bulb Temperature	33,245
Dry-bulb Temperature Gradient	245-33
Precipitation	3

2.2.4 Short-Term (Accident) Diffusion Conditions

We estimated short-term relative concentration (X/Q) values for accidental releases from plant buildings and vents. These values are considered for various time periods following a release and are for the exclusion area boundary, a 1.5 mile radius from the containment building and the minimum distance to the reservation boundary, a 4.5-mile radius from the facility. We used the WPPSS meteorological data for the one year (April 1974 - March 1975) of data collection with wind direction and speed measured at the 33-foot level. We also used a calculational model which considered the variability of meteorological conditions by direction and which was more suitable for a site in a desert location.

Table 2.2-3 shows the 0-2 hour X/Q values which we estimate will be exceeded no more than 27 hours per year on the average at the exclusion area boundary for each of 16 downwind directions. The southwest downwind sector had the highest X/Q value for both the exclusion area boundary and reservation boundary distance. The values from this sector were used in our evaluation of short-term, accidental releases and are listed in Table 2.2-4. This maximum sector 0-2 hour X/Q value of 1.4×10^{-4} seconds per cubic meter will occur less than five percent of the time (438 hours per year) around the exclusion boundary.

TABLE 2.2-3

SHORT-TERM RELATIVE CONCENTRATION BY DOWNWIND DIRECTION - FFTF SITE

The values are the 0-2 hour relative concentrations (X/Q) which we estimate will be exceeded no more than 27 hours per year at the exclusion boundary (a 1.5-mile radius from the containment buildings) in the downwind direction indicated.

Downwind Sector	X/Q (seconds/meter^2)	Downwind Sector	X/Q (seconds/meter^3)
N	9×10^{-5}	S	13×10^{-5}
NNE	8×10^{-5}	SSW	13×10^{-5}
NE	10×10^{-5}	SW	14×10^{-5}
ENE	9×10^{-5}	WSW	12×10^{-5}
E	7×10^{-5}	W	12×10^{-5}
ESE	10×10^{-5}	WNW	11×10^{-5}
SE	11×10^{-5}	NW	7×10^{-5}
SSE	12×10^{-5}	NNW	8×10^{-5}

TABLE 2.2-4

SHORT TERM RELATIVE CONCENTRATION VALUES USED FOR ACCIDENT ANALYSES - FFTF SITE

The values are estimates of short-term relative concentrations (X/Q) used to evaluate accidental releases from plant buildings and vents. The values are for specific time periods following a release at the exclusion area boundary (ER 1.5 mi) and the reservation boundary distance (RB 4.5 mi).

Time Period	Location	X/Q (sec/meter^3)
0-2 hours	ER	1.4×10^{-4}
0-8 hours	RB	2.7×10^{-5}
8-24 hours	RB	1.9×10^{-5}
1-4 days	RB	8.9×10^{-6}
4-30 days	RB	3.0×10^{-6}

The Project based its short-term relative concentration values on the curves provided in Regulatory Guide 1.4, Revision 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors." These values are listed in Table 2.2-5. Regulatory Guide 1.4 states

that the atmospheric conditions assumed in the guide should be used only until adequate site meteorological data are obtained. As discussed in Section 2.2.3, we consider that the meteorological measurement programs available on the Hanford Reservation can better describe the meteorological conditions necessary to estimate the diffusion conditions at the FFTF site; therefore, we concluded that these available data should be used in lieu of Regulatory Guide 1.4 assumptions. We also concluded that our evaluation procedures for this site should (1) include recognition of experimental data generic to a site in a desert location and (2) provide estimates of the variations in atmospheric dispersion that occur as a function of wind direction and distance from the source to receptor.

TABLE 2.2-5

SHORT-TERM RELATIVE CONCENTRATION VALUES
PRESENTED BY THE FFTF PROJECT

The values are the short-term relative concentrations (X/Q) presented by the FFTF Project in FSAR Table 2.3-12, based on Regulatory Guide 1.4. The values are for specific time periods following a release and are for the exclusion area boundary (ER, 1.5 miles) and the reservation boundary distance (RB, 4.5 miles).

Time Period	Location	X/Q (sec/meter^3)
0-2 hours	ER	1.7×10^{-4}
0-8 hours	RB	3.8×10^{-5}
8-24 hours	RB	8.0×10^{-6}
1-4 days	RB	2.8×10^{-6}
4-30 days	RB	5.8×10^{-7}

Our Standard Review Plan procedures involve the use of onsite meteorological data in a direction independent model to estimate atmospheric diffusion conditions which occur no more than five percent of the time (438 hours per year) around the site at a distance equal to the minimum exclusion boundary distance. An interim Branch Technical Position, approved by the Regulatory Requirements Review Committee at their May 2, 1978 meeting, allows the use of either this direction independent approach or the direction dependent approach as outlined in the draft Regulatory Guide 1.XXX, "Atmospheric Dispersion Models for Potential Accidental Consequences Assessments at Nuclear Power Plants." This direction dependent approach considers horizontal plume meander, variation of meteorological conditions by direction, and variable exclusion area boundaries. In our evaluation we modified the calculational procedures described in Standard Review Plan 2.3.4 by using the approach outlined in the draft guide. We assumed a ground level release with a building wake factor of 550 square meters.

The Standard Review Plan also requires that we determine whether the lateral and vertical plume spread parameters used in our calculational model are conservative for estimating relative concentrations. Based on atmospheric diffusion field test data developed at two desert locations (the National Reactor Resting Station, Idaho, and the Hanford Area, Washington), we concluded that the overall plume

dispersion for stable atmospheric conditions with low wind speeds (periods of poorest dispersion) can be less for a desert climate than the dispersion assumed in the draft Regulatory Guide. Thus, in lieu of the plume spread and plume meander factors described in the draft guide, we used the lateral and plume spread factors based on the desert data.

2.2.5 Long-Term (Routine) Diffusion Estimates

Using the 33-foot level wind data and the vertical temperature difference data from the WPPSS site, we made reasonable estimates of average atmospheric dispersion conditions for the FFTF site using our atmospheric dispersion model for long-term releases. This model is based on the "Straight-Line Trajectory Model" described in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors." As suggested in Regulatory Guide 1.111 and as discussed in Section 2.2.4, we used the vertical dispersion parameters developed from atmospheric diffusion data from the desert field tests. We assumed a ground level release only and included an estimate of maximum increase in calculated relative concentration (X/Q) due to the spatial and temporal variation of the airflow not considered in the straight-line trajectory model, as discussed in Regulatory Guide 1.111.

Table 2.2-6 lists the X/Q values we estimated for a distance of 4.5 miles from the facility.

TABLE 2.2-6

LONG-TERM RELATIVE CONCENTRATION ESTIMATES
BY DOWNWIND DIRECTION - FFTF SITE

The values are our estimates of annual average relative concentrations (X/Q for the reservation boundary distance (a 4.5 mile radius) in the downwind direction indicated.

Downwind Sector	X/Q ($sec/meter^3$)	Downwind Sector	X/Q ($sec/meter^3$)
N	6×10^{-7}	S	8×10^{-7}
NNE	4×10^{-7}	SSW	6×10^{-7}
NE	4×10^{-7}	SW	7×10^{-7}
ENE	4×10^{-7}	WSW	5×10^{-7}
E	3×10^{-7}	W	4×10^{-7}
ESE	5×10^{-7}	WNW	4×10^{-7}
SE	7×10^{-7}	NW	3×10^{-7}
SSE	8×10^{-7}	NNW	5×10^{-7}

2.2.6 Conclusions

The FFTF Final Safety Analysis Report has provided sufficient information for us to evaluate the regional and local meteorological conditions of importance to the

design of the FFTF. The meteorological data available on the Hanford Reservation has provided a meteorological description of the site and vicinity, and were used as bases for calculation of relative concentration values of post-accident and annual average atmospheric diffusion conditions. Desert conditions at the site required special consideration of diffusion parameters to provide more suitable estimates of relative concentration values.

2.3 Hydrology

The following discussion is abstracted from our Safety Evaluation Report for the construction stage review; there is no new information which would change the findings made at that time.

The Ben Franklin Dam Pool would be at elevation 400 feet above mean sea level. This is still well below the reactor operating floor at elevation 550 feet. The maximum probable flood would be at elevation 436 feet, 114 feet below the operating floor of the reactor.

Ground water is about 170 to 180 feet below grade and is hydraulically connected to the Columbia River. The construction of the Ben Franklin Dam would raise the level 20 to 50 feet but would not be expected to change the direction of movement of the ground water.

The water wells in proximity to the site are monitored periodically for contamination. It is unlikely that a radiation waste effluent spill at the FFTF site would go undetected and contaminate the potable ground water.

Our conclusions reached in 1972 that the relatively high site elavation is well above the potential flood level from the Columbia River and that there is little likelihood of undetected ground water contamination originating from the FFTF remain unchanged.

2.4 Geology/Seismology

The geology/seismology findings for the FFTF remain unchanged from the findings made in 1972. The only perturbation to the seismology in the State of Washington relates to the Skagit facility west of the Cascades and north of Hanford. This concern has since been resolved in the course of the Skagit review and is not considered to be related to the Hanford site.

In addition to our review in 1972, the U.S. Geological Survey and the National Oceanic and Atmospheric Administration prepared reports and findings which were included as Appendices A and B in our 1972 construction stage review Safety Evaluation Report.

We and our consultants concluded at that time that the seismic design for the design basis earthquake, now referred to as the safe shutdown earthquake, value of 0.25g is adequate for the FFTF. We have no basis to change this conclusion.

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 Conformance with General Design Criteria

The Fast Flux Test Facility was designed and is being constructed on the basis of the "Principal Design Criteria" which were contained in Appendix F of the FFTF Preliminary Safety Analysis Report. These criteria were adapted by the FFTF Project from the proposed AEC General Design Criteria, which were published July 11, 1967, and modified to relate specifically to FFTF systems. On July 11, 1971, the Atomic Energy Commission published the General Design Criteria as Appendix A to 10 CFR Part 50.

The light water reactors which we have reviewed for operating licenses in recent years were designed and constructed on the basis of the proposed 1967 General Design Criteria. To the extent practical, these applicants have attempted to comply with the newer criteria of July 1971. Recognizing work already accomplished and design commitments made, the Final Safety Analysis Reports for these light water reactors have discussed the plant design with respect to the July 1971 criteria.

In keeping with our licensing policy for light water reactors, our technical review has attempted to assess the FFTF design against the General Design Criteria now in effect. However, we found that the FFTF Final Safety Analysis Report did not discuss the FFTF design in relation to the July 1971 criteria. In a letter to the Project dated April 26, 1978, we requested that the FFTF Project provide such a discussion in the Final Safety Analysis Report, recognizing the state of completion of the facility and the fact that the FFTF is a liquid metal-cooled reactor.

In its May 31, 1978 response, the FFTF Project stated its position that the 1971 General Design Criteria were applicable for light water reactors only, and did not apply specifically to the FFTF. For this reason, the Project claimed that a comparison against the 1971 criteria would not contribute to overall FFTF safety and therefore was neither necessary nor warranted.

We do not agree with the Project's position. Although the 1971 General Design Criteria (and the 1967 proposed General Design Criteria for that matter) are specifically oriented toward light water reactors, the introduction to Appendix A to 10 CFR Part 50 states that these criteria "are intended to provide guidance in establishing the principal design criteria for such other (types) units." This was indeed how the Project established its Principal Design Criteria published in the Preliminary Safety Analysis Report. In its letter of August 20, 1976 (see

Appendix B to this report), the Project specifically requested the review of the design to be comparable in technical matters to a licensed plant review.

As stated in Section 1.6 of this report, we recognize that specific general design criteria and regulatory guides for liquid metal fast breeder reactors do not exist. However, we believe that the guidance found in the existing light water reactor General Design Criteria and Regulatory Guides is enough to indicate the dominant role that safety must have in nuclear plant design. Basic to the defense-in-depth concept is the principle of designing a nuclear plant to accommodate a single failure by means of redundant, diverse, and independent systems. In our opinion, although the as-built FFTF design is generally above average in the quality of components, the Project has not implemented this defense-in-depth concept to the extent normally expected of a licensed plant.

Consequently, on a comparability basis with light water reactors the FFTF design would not be judged as having met the spirit and intent of certain portions of the existing General Design Criteria. Although this has not necessarily led to an unacceptable design, it does represent a change in risk.

3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

As in the case of light water reactors, we have reviewed the seismic classification methodology used for the FFTF against the appropriate requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the recommendations of Regulatory Guide 1.29, "Seismic Design Classification."

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100. Regulatory Guide 1.29 describes an acceptable method of identifying and classifying those features that should be designed to withstand the effects of the safe shutdown earthquake.

In Section 3.2.1.1 of the Final Safety Analysis Report, the FFTF Project has identified a number of differences (denoted as Items A through K) between the FFTF design and the recommendations of Regulatory Guide 1.29. Considering that the FFTF is a sodium-cooled fast reactor, we have reviewed these differences. Based on our review, with the exception of Items A, C, D and K, we find that the differences are acceptable. Our evaluation of Items A, C, D and K is given below.

Item A states that the FFTF does not have a separate emergency core cooling system for removing core decay heat. Instead, this function is performed by the seismic Category I reactor heat transport system. It is further stated that the emergency core cooling system function can be performed by convective flow without electric power being supplied to the heat transport system. Based on our review of the FFTF natural circulation capability, we find that Item A is acceptable provided that the ability of the FFTF design to remove decay heat safely and reliably under all natural circulation conditions is adequately verified.

According to Items C and D, the current FFTF design has no provisions for a containment heat removal system or a containment atmospheric cleanup system. It is the position of the FFTF Project that neither system is required to mitigate the consequences of an accident. Our evaluation of the containment radiological release is contained in Section 15.0 of this report.

Item K states that the lube oil systems for the primary and secondary heat transport system pumps are not qualified to seismic Category I requirements since these pumps would not be required to operate following an earthquake. In our January 24, 1978 letter to the FFTF Project, we expressed concern that the failure of the lube oil seals following a safe shutdown earthquake might reduce the level of safety of the reactor coolant system. Therefore, we requested that the Project assess the effects of the failure of these oil seals, including a lube oil fire, on the safety of the plant. In Supplement No. 24 to the Final Safety Analysis Report, the Project provided the requested assessment. Based on our review of the Project's response to our concern about pump lube oil system failure, we agree with the Project's conclusions that, if the heat transport system pumps are not required to operate following an earthquake, loss of the lube oil cooling system will not lead to unacceptable consequences. However, as with Item A, our acceptance of Item K is conditioned upon adequate verification of the natural circulation capability of the FFTF design (see Sections 5.3 and 14.2 of this report).

Table 3.2.1 of the FFTF Final Safety Analysis Report identified the structures, systems, and components important to safety that are designed to withstand the effects of a safe shutdown earthquake and remain functional. From our review of Table 3.2.1 and Figures 3.2-1 through 3.2-19, we have determined that, for piping systems which are partially seismic Category I, the Category I portion of the system extends up to and includes the isolation valves that isolate the part which is Category I from the nonseismic portions of the system. In the case of light water reactors, the staff's licensing position, as stated in Regulatory Guide 1.29, is that the seismic Category I portion of such systems should extend to the first seismic restraint beyond the isolation valves that isolate the Category I portion from the nonseismic portion of the system. Our concern is that the failure of a nonseismic Category I component (piping and piping restraint) could impair the

safety function of a seismic Category I component (isolation valve). Through discussions with the Project this matter has been clarified and found to be in compliance with Regulatory Guide 1.29 and General Design Criterion 2.

On several occasions during our review (e.g., our letters of August 13, 1976 and January 24, 1978), we expressed concern that none of the sodium leak detection systems were qualified to remain functional following a safe shutdown earthquake and thus did not meet the intent of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." As a result of our review, we recommend that at least one sodium leakage detection system be seismically qualified to function following a safe shutdown earthquake. Further discussion of our evaluation of this matter may be found in Section 5.2 of this report.

3.2.2 System Quality Group Classification

General Design Criterion 1, "Quality Standards and Records", requires that nuclear power plant structures, systems, and components important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," established quality assurance requirements for the design, construction, and operation of those structures, systems, and components of nuclear power plants important to safety.

Section 50.55a of 10 CFR Part 50 identifies those ASME Section III, code Class 1 components of light water reactors important to safety which are part of the reactor coolant pressure boundary. These components are designated as Quality Group A in Regulatory Guide 1.26, "System Quality Group Classification and Standards." This Regulatory Guide describes an acceptable method for determining quality standards for Quality Group B, C, and D, water- and steam-containing components important to safety for light water reactors.

Taking into account the difference between the FFTF and light water reactors, we have reviewed the FFTF system quality group classification in a manner consistent with our review of light water reactors. Our evaluation of the FFTF Quality Assurance Program is contained in Section 17.0 of this report.

The minimum code requirements applicable to each quality group classification have been identified in Table 3.2-2 of the Final Safety Analysis Report. These requirements correspond to those specified in Section 50.55a of 10 CFR Part 50 and Regulatory Guide 1.26. Table 3.2-1 of the Final Safety Anaysis Report identifies the quality group classification for the safety-related systems or portions of systems. These quality group classifications have also been denoted on system piping and instrumentation diagrams.

During our review of this material, we determined that the dump heat exchanger standby compressed air cylinders and the associated piping and valves have been designed to seismic Category I and Quality Group D requirements. Since the standby compressed air systems provide the emergency air supplies to establish control for decay heat removal, they have a safety-related function. Because of their safety significance, these systems should be classified Quality Group C to conform with Regulatory Guide 1.26 recommendations. Therefore, we recommend that the quality group classification of the sump heat exchanger standby compressed air systems be upgraded from Quality Group D to Quality Group C.

In Section 3.2.2 of the Final Safety Analysis Report, the FFTF Project has identified differences (designated items A through E) between the FFTF quality group classification scheme and the recommendations of Regulatory Guide 1.26. In these instances, the FFTF Project has established an equivalent level of quality by upgrading a lower quality level through the imposition of appropriate Hanford Works Standards, Reactor Development Technology Standards and additional industry codes and standards. On this basis, we find items A through E to be acceptable.

Therefore, based on our review, with the above exception, we conclude that fluid system pressure-retaining components important to safety will be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Fluid system pressure-retaining components which are part of the reactor coolant pressure boundary and other fluid systems important to safety, where reliance is placed on these systems (1) to prevent or mitigate the consequences of accidents and malfunctions originating with the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintenance in the safe shutdown conditions, and (3) to contain radioactive material, have been classified in an acceptable manner in Tables 3.2-1 and 3.2-2 of the Final Safety Analysis Report and on system piping and instrumentation diagrams.

3.3 Wind and Tornado Criteria

3.3.1 Wind Design Criteria

All seismic Category I structures exposed to wind forces are designed to withstand the effects of the design basis wind. The design basis wind specified has a velocity of 100 miles per hour with a recurrence interval of 1000 years.

The procedures that are used to transform the wind velocity into pressure loadings on structures and to establish the associated vertical profile of wind pressures and the gust factors are in accordance with ASCE paper No. 3269 entitled "Wind Forces on Structures." This paper was written by a task committee consisting of a panel of experts in the areas of meteorology, hydrodynamics and structural dynamics. The recommendations in this report have been used in the design of conventional structures against winds and also as a basis in the establishment of building codes and standards. Buildings and structures which are designed in accordance with

these recommendations have been found to possess adequate resistance against the specified wind forces.

On the basis of the above considerations, we conclude that the procedures used to establish the wind forces on seismic Category I structures provide reasonable assurance that they are conservatively estimated. The use of these procedures provides reasonable assurance that the structural integrity of seismic Category I structures in the FFTF will be maintained in the event of design basis winds. Consequently, seismic Category I systems and components located within these structures are adequately protected and will perform their intended safety functions if needed. Conformance with these procedures is an acceptable basis for satisfying the requirements of General Design Criterion 2.

3.3.2 Tornado Design Criteria

All seismic Category I structures exposed to tornados and needed for the safe shutdown of the plant in the event of a tornado are designed to resist a tornado of a 150-miles per hour tangential wind velocity and a 25-miles per hour translational wind velocity with a simultaneous atmospheric pressure drop of 0.75 psi in 3 seconds. The acceptability of these parameters is discussed in Section 2.2 of this report. The procedures which consist of transforming the tornado wind velocity into pressure loadings are similar to those used for the design basis wind as discussed in Section 3.3.1 with no variation on pressure in the vertical profile and no consideration of the gust effect. The tornado missile effects are considered by using the procedures as presented in Section 3.5 of this report. Structures which are not designed for the tornado were designed by the FFTF Project contractors for potential missile generation and for possible collapse to ensure that the integrity of seismic Category I structures would not be jeopardized.

The procedures utilized to determine the loadings on structures induced by the design basis tornado specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that, in the event of a design basis tornado, the structural integrity of the plant structures that need to be designed for tornados will not be impaired and, in consequence, safety-related systems and components located within these structures will be adequately protected and may be expected to perform their necessary safety functions as required. Conformance with these procedures is an acceptable basis for satisfying the requirements of General Design Criterion 2.

3.4 Water Level Flood Design

3.4.1 Flood Protection

The FFTF site was evaluated during the construction agreement stage and, since the ultimate heat sink is air, it was not necessary to locate the FFTF near water. The

facility is 114 feet above the maximum probable flood elevation of 436 feet at the operating floor and the bottom of the FFTF foundation mat is approximately 30 feet above the maximum probable flood. No special flood provisions are provided for the seismic Category I structures.

3.5 Missile Protection Criteria

3.5.1 Missiles Generated Outside of Containment

We reviewed the basis for establishing tornado requirements for the FFTF in 1971 and conclude that a tangential wind speed of 150 miles per hour and a translational speed of 25 miles per hour was an appropriately prudent estimate for the FFTF.

The tornado criteria described in "Review of Tornado Consideration for FFTF" HEDL-TME-71-35 (basis for the Commission decision) are consistent with those presented in Section 3.5-5 of the Final Safety Analysis Report. Potential damage from a spectrum of postulated missiles is taken into consideration in the design of the FFTF so as to assure no loss of function or damage to safety-related equipment essential for the plant safe shutdown as a result of missile impact. The spectrum of postulated missiles consists of missiles generated inside and outside the plant facility. The external missiles which are taken into consideration as tornado-generated missiles consist of:

(1) a 2-inch x 12-inch x 12-foot plank traveling at 100 miles per hour

(2) a 4-foot x 8-foot x 3/4-inch plywood sheet traveling at 150 miles per hour

(3) a 26-inch x 20-foot sheet of 26 gauge corrugated steel siding traveling at 150 miles per hour.

We have arbitrarily included in our evaluation a 3-inch schedule 40 pipe 10 feet long in the list of selected external missiles. The inclusion of the pipe and its velocity are arbitrary since it is unlikely it would be picked up by the 175-mile per hour tornado. A missile of this mass and shape travelling at 100 miles per hour would not penetrate the steel containment shell or concrete walls of the auxiliary service building (2-foot walls and 3-foot roof). This portion of the structure houses the secondary sodium pumps, portions of the secondary sodium piping, cable spreading room and control room. The most vulnerable portion of the facility to external missiles is the remaining secondary heat transport system piping above grade in the south and west loops in the cruciform region between the air dump heat exchangers which are not tornado missile hardened. This portion of the secondary sodium piping is outside of the hardened auxiliary service building. The east secondary sodium loop is fully tornado-hardened with 1/2-inch-thick steel plates providing missile barriers for piping and air dump heat exchangers.

The emergency diesel generators are not a Class IE system and function in the FFTF as an onsite auxiliary power system. The Class IE emergency power system is a dual set of batteries in the substructure of the auxiliary services building and by location is tornado hardened.

We consider the design decisions applied to the FFTF with regard to implementation of General Design Criteria 2 and 4 to result in a somewhat less than desirable situation in providing adequate missile protection from natural phenomena. It would have been prudent, in our view, to have tornado-hardened a second secondary sodium loop. It also would be desirable to provide missile protection to certain containment isolation valve operators and upgrade the non-Class IE power supplied to them outside of containment to achieve comparability with light water reactor practice even though the loss of power to the containment isolation valves will presumably leave them in their proper closed position for containment isolation. In our view the knowledge of valve position indication would be important and desirable to a plant operator in all containment isolation situations.

The closure of certain dump heat exchanger isolation valves and air louvers is required following scram to prevent sodium from freezing in the dump heat exchanger in the event of loss of all AC power. One could postulate that the pilot valve operators important to safety could also malfunction and that the motor actuated control of important safety functions in the facility following a tornado would not occur. One then would rely on remote manual regulation of the dump heat exchanger and local manual operation of dump heat exchanger isolation valves outside of the containment. In our view it would not be prudent to operate the FFTF when a severe tornado weather condition exists. The effect of this decision would be to reduce the risk of loss of heat sink due to overcooling from not isolating the dump heat exchanger, reduce the consequences of a secondary sodium fire due to secondary sodium pump pressure reduction, improve control or reduce risk of loss of containment isolation and provide a modified, controlled safe plant shutdown as opposed to an emergency shutdown of unknown severity following a tornado.

The early shutdown of the reactor is further justified on the basis of achieving a modified controlled shutdown using plant power and valve operators as opposed to an unplanned indeterminate shutdown. The time of shutdown (minutes) has little impact on the stored energy or the decay heat removal process other than starting from an initial controlled situation rather than starting with a possible unknown or uncontrolled loss of power scenario.

Missile protection is provided to ensure safe shutdown capability of the reactor facility. There are essentially no high pressure fluid systems in the facility; therefore, no internal missiles resulting from such systems are identified as a design basis. Rotating machinery with high kinetic energy has been analyzed as a source of potential internal missiles. Protection against externally generated

missiles outside containment is achieved by proper orientation of components and systems, by use of missile barriers and by physically separating redundant safety-related systems or components from each other and from non-safety-related systems.

As a result of our review, we conclude that the design incorporating the above recommendations satisfies General Design Criterion 4, "Environmental and Missile Design Bases" as it relates to structures housing essential systems and to safety-related systems being protected against externally generated missiles and, therefore, is acceptable.

3.5.2 Internally Generated Missiles Inside Containment

The reactor coolant systems and rotating machinery were examined within the facility to determine their potential to affect the function of safety-related equipment. The liquid metal systems and the gas systems are not high energy systems.

The rotating machinery reviewed included the primary sodium pump impellers and the primary closed loop motor generator flywheels. The primary sodium pump impellers are 41-1/2-inches in diameter, weigh 1500 pounds and rotate at 1200 revolutions per minute. The pump casing radial wall section consists of a 1-inch turning vane, 2-to 4-inch thick casing and a 1-3/4-inch thick pump tank. The motor-generator set flywheels are 58 inches in diameter and weigh 7750 pounds. The barriers are inherent in the substructure by placement and orientation. Where this is inadequate a 24-inch thick concrete wall has been installed.

The staff review consisted of examining the Project's analysis and evaluating the potential for penetration of pump casings and adjacent walls, and the proximity of safety-related equipment in accordance with acceptable methods for evaluating the effect of impact energy on shells and structures.

The procedures that were used in the design of structures, shields, and barriers to resist the effect of missiles are those specified in Bechtel Topical Report BC-TOP-9A, Revision 2, "Design of Structures For Missile Impact." The staff has reviewed this report and approved its use in nuclear power plant design in a letter dated November 25, 1974. The procedures as described in this Bechtel report and utilized in the FFTF to determine the effects and loadings on structures and missile shields and barriers due to design basis missiles provide a conservative basis for engineering design. Missile impact loads have been combined with other loads as described in Section 3.8 of this report.

The use of the procedures mentioned above provides assurance that in the event of design missiles, the structural integrity of seismic Category I structures or missile shields and barriers in the FFTF will not be impaired or degraded. Seismic Category I systems and components protected by these structures, shields, or barriers are, therefore, adequately protected against the effects of missiles and

will perform their intended safety function, if needed. Conformance with the procedures of Bechtel Topical Report BC-TOP-9A is an acceptable basis for satisfying the requirements of General Design Criteria 2 and 4.

The existing FFTF design does not have a high potential for internally generated missiles which would impair the safety-related functions of safe shutdown of the reactor and its decay heat removal.

The potential for missile generation resulting from the core disruptive accident was also examined and found not to be a challenge to the containment vessel.

3.6 Protection Against Postulated Piping Failures

3.6.1 Piping Failures Outside of Containment

Plant design criteria are applied to the facility to protect against the effects of postulated pipe breaks and cracks, including pipe whip, jet effect, and environmental effects. The means used to protect safety-related systems and components include physical separation, enclosure within suitable designed structures, pipe whip restraints, and equipment shields. Protection against pipe failure outside containment is in accordance with A. Giambusso's letter dated December 12, 1972, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment." The Project has analyzed high energy piping systems for the effects of pipe whip, jet impingement, and environmental effect on safety-related systems and equipment. We conclude that the plant design criteria discussed above are acceptable.

High energy systems that were evaluated for the above effects include the compressed air system, reactor heat transport system, closed loop system, auxiliary liquid metal system, inert gas receiving and processing system, and impurity monitoring and analysis system. Areas evaluated to determine that there is adequate protection against the effects of pipe whip, jet impingement and environmental effects on safety-related systems and components include the control room, safety-related electrical equipment and instrumentation, containment penetration area, secondary heat transport system loops, and the emergency dump heat exchanger controls.

Except for the compressed air and inert gas receiving and processing systems, the process fluid contained in the high energy piping outside containment is high temperature (approx. 950°F) liquid sodium at relatively low pressures (less than 185 psig). The sodium temperature is below its atmospheric pressure boiling point and, hence, there will be no flashing in the event of pipe rupture. The pressure and energy available for pipe whip during pipe break will be considerably less than that for the high temperature, high pressure water piping that is found in other reactor facilities. We are in agreement with this approach.

The plant design basis includes the ability to sustain a high energy pipe break accident coincident with a single active failure and retain the capability for safe shutdown.

The Project has also presented an analysis of the effect of the moderate energy line failures outside containment on safety-related systems. The moderate energy systems are designed to meet the criteria set forth in Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment." The moderate energy systems that were analyzed include the service piping systems, radioactive waste system, heating and ventilation system, fire protection system, portions of the inert gas receiving and processing systems, and portions of the impurity monitoring and analysis system. We have evaluated the analysis and conclude that a postulated pipe crack in a moderate energy line will not cause loss of function of any safety-related system.

Based on our review, we conclude that the Project has adequately designed and protected areas and systems required for safe plant shutdown following postulated events, including the combination of high energy pipe failure and single active failure. The plant design meets the criteria set forth in A. Giambusso's letter dated December 12, 1972, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment." The plant design also meets Branch Technical Position APCSB 3-1 in regard to protection of safety-related systems and components from a postulated moderate energy line failure and is, therefore, acceptable.

3.6.2 Piping Failures Inside of Containment

The piping inside of the containment is located in shielded cells in the containment substructure except for some bottles for inert gas systems. Those liquid metal sodium and NaK systems in the containment substructure are moderate energy systems and subcooled such that there is no stored energy which would flash and cause pipe whip. The inert gas lines are too small and at too low a pressure to have a high structural damage potential; nor are they located adjacent to safety systems. On the basis of the information provided in the FFTF Final Safety Analysis Report, we conclude that the intent of General Design Criteria 1, 2, 4, 14 and 15 have been complied with.

3.7 Seismic Design

3.7.1 Seismic Input

An operating basis earthquake of 0.125g and a safe shutdown earthquake of 0.25g were originally specified for use in the design of the FFTF structures, systems and components. However, during the design of the FFTF it was determined by the FFTF Project, and approved by the staff, that analysis for the 0.125g operating basis earthquake was not necessary since the Project has stipulated that the

reactor be shut down in the event of an earthquake with peak ground acceleration exceeding 0.05g.

The free field response spectra utilized for the seismic design of the FFTF structures, as described in "Seismic Design Criteria for the Fast Flux Test Facility," JABE-WADCO-02, were approved by the staff before the issuance of Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," and are, therefore, different from those in Regulatory Guide 1.60. The FFTF basic free field response spectra are less "broad-banded" than the Regulatory Guide 1.60 spectra. Despite the apparent difference in the response spectra, the FFTF Project has demonstrated through a series of analyses that the seismic design of FFTF structures is as conservative as if the current design criteria were used. This was possible because of the conservative seismic design practices originally followed by the FFTF Project. These practices consisted of: (1) use of a subsurface control motion instead of one at finished grade level as permitted by current criteria, (2) use of two additional control motions for buildings other than containment, resulting in a fairly "broad band" design spectra, and (3) use of conservative damping values, that is, damping values smaller than currently allowed.

The acceleration time histories used for the design of components and equipment were adjusted in amplitude and frequency to envelop the response spectra specified for the site.

On the basis of our review of the information related to seismic input, we conclude that the seismic input criteria used by the Project provide an acceptable basis for seismic design.

3.7.2 Seismic System Analysis

Most of the seismic Category I structures in the FFTF are embedded in soil. The embedment ranges from 16 feet to 82 feet. For the containment structure, in order to take the effects of embedment into consideration, the FFTF Project used a 2-D finite element analysis. In the analysis, the structures are idealized and coupled together with the soil media to form a plane strain finite element model. The mesh size of the finite element model is capable of transmitting shear waves up to a frequency of 20 Hertz. The base of the model is placed at the bedrock 200 feet below the ground level and is assumed to move with the bedrock. The FFTF Project utilized the SHAKE computer program to derive the rock motion from the control motion which is placed approximately at the base level of the structure in the free field. The 2-D finite element analysis was performed through the use of the LUSH and the FLUSH computer programs. The two-dimensional finite element analysis gives the global responses of the structure which consist of the displacement and acceleration time histories and response spectra at the base and floor levels of the structure.

For the containment building, global vertical and horizontal response spectra are derived directly from the two-dimensional finite element analysis, considering the building floors to be rigid. For structures, systems and components in buildings outside the containment where seismic response amplification in the vertical direction is considered to be significant, a seismic system analysis in the vertical direction was made through the use of a multi-degree-of-freedom lumped mass system to develop vertical local response spectra.

For the reactor containment building, the fundamental frequency of torsional vibration was estimated through the use of a multi-degree-of-freedom system and soil springs. In order to calculate the response spectra due to torsional effects, spectral amplification factors for torsional motion were assumed to be equal to those for horizontal translational motion. These factors were used to establish the peaks of the response spectra for torsional and maximum acceleration due to torsional response.

For all other structures an incidental eccentricity of 5 percent in the long dimension of the structures was assumed for torsional design considerations. The ordinates of the response spectrum curves were increased by at least 10 percent from the calculated values of all frequencies. For torsionally sensitive structures the FFTF Project used a three-dimensional space frame response analysis to study the torsional effects. The stability of seismic Category I structures against overturning under horizontal and vertical earthquakes was checked by the FFTF Project assuming no passive soil pressure. The minimum factor of safety against overturning was found to be 1.8.

In order to take into account variations in soil characteristics and structural characteristics, different damping and shear modulus values were used by the FFTF Project. In addition the peaks of the design response curves were broadened to allow for shifting in peak frequency.

The seismic system analyses as summarized above are described in great detail in report No. HEDL-BR-5853-C-2, Rev. 2, "Seismic Analysis Report For Building Structures" Vols. 1 to 4, submitted by the FFTF Project as supplementary information to the Final Safety Analysis Report. We have reviewed the report and Section 3.7.2 of the Final Safety Analysis Report and found that the seismic system analysis methods and procedures used by the FFTF Project provide an acceptable basis for the seismic design.

3.7.3 Seismic Subsystem Analysis

For the subsystem analysis, the global responses obtained from the system analysis are considered in determining the input excitation at the point where the subsystem is attached to the structure. Model response spectrum and time history methods for multi-degree-of-freedom systems form the basis for the analysis of the subsystems.

Governing response parameters are combined by the square root of the sum of the squares to obtain the maximum when the modal response spectrum method is used. The absolute sum of modal responses is used for modes with closely spaced frequencies. Only a two-component earthquake, one horizontal and one vertical, is considered. The effects of responses in horizontal and vertical direction are added directly.

We conclude that the above methods and procedures used by the FFTF Project provide an acceptable basis for seismic design of subsystems.

3.7.4 Seismic Instrumentation Program

A seismic monitoring system has been provided to monitor continuously the seismic activity. The system consists of three triaxial strong motion accelerometers, two omnidirectional triggers, an analog tape recorder, strip chart recorder, power supply and peak recording accelerograph.

The accelerometers are located on the containment foundation structure at the 480-foot elevation, on the containment structure at ground level, at the 550-foot elevation, and at ground level 150 feet from the containment building. A peak recording accelerograph is mounted on the containment structure at the 625-foot elevation, approximately 75 feet above the operating floor.

One omnidirectional trigger is mounted on the containment foundation structure at the 480-foot elevation and the other at the opposite side of the building.

The FFTF Project has taken exception to Regulatory Guide 1.12, "Instrumentation for Earthquakes," by not providing response spectrum recorders or any seismic instrumentation to piping and seismic Category I equipment. The justification given by the FFTF Project for taking such exception is that the plant will be shut down in the event that a seismic event producing a ground acceleration of 0.05g or more occurs which is only one-fifth of the safe shutdown earthquake of 0.25g used for the FFTF seismic design.

Whenever a seismic event producing a ground acceleration of 0.05g or more occurs, seismic Category I systems will be requalified and a detailed analysis of the seismic data will be conducted before the plant is restarted.

We have reviewed the Project's seismic instrumentation program and find it to be acceptable, since the precautionary measures adopted by the FFTF Project will ensure the safety of the public.

3.8 Design of Seismic Category I Structures

3.8.1 Steel Containment

The reactor coolant system is housed within a cylindrical steel vessel with hemi-ellipsoidal top and bottom heads. The diameter of the cylinder portion is 135 feet

and its height is 102 feet. The portion of the vessel below grade is sandwiched in concrete, while above grade the vessel exterior surface is insulated. The containment vessel is designed for an internal design pressure of 10 psig.

The steel containment, including all its penetrations, is designed, analyzed, fabricated, constructed, inspected and tested to the requirements of Subsection B of the ASME Boiler and Pressure Code, Section III with addenda through summer 1970. The containment is designed for all load combinations that are considered credible, including appropriate combinations of design basis accident, design basis earthquake and other loads. The materials used in the construction of the containment which consist of SA-516 grade 60 for steel shell plate and SA-299, SA-350 and SA-333 for penetration components meet the requirements of Subsection B of the ASME Section III Code.

After the completion of construction and prior to operation, the containment will be subjected to a structural proof test at a testing pressure of 12.5 psi.

We have concluded that the criteria used in the analysis, design, construction, and testing of the steel containment structure to account for the loadings and conditions that are anticipated to be experienced by the structure during its service lifetime are in conformance with established criteria as contained in codes, standards, and specifications acceptable to the staff.

The use of these criteria as defined by applicable codes, standards, and specifications; the load and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within and outside the containment, the containment structure will withstand the specified conditions without impairment of structural integrity or safety function. Conformance with these criteria constitutes an acceptable basis for satisfying the requirements of General Design Criteria 2, 4, 16, and 50.

3.8.2 Containment Interior Structures

The containment interior structures, mostly located below the operating deck, consist mainly of cells and pipeways built of reinforced concrete and lined with steel plates. Cells and pipeways which house piping and/or equipment containing primary sodium are to be inerted with a nitrogen atmosphere. The cells are separated physically from each other. The reactor vessel is located in the reactor cavity and is supported by a steel ring forging attached to the concrete ledge formed at the top of the reactor cavity wall. The head compartment, which is located above the reactor head and forms a service area for the reactor, has reinforced concrete walls and a 14-inch deep steel triskelion plate supported on a grid of steel beams to form the operating deck above the reactor. Other major steel structures within the containment are: the polar gantry crane support structure which is a space

frame structure consisting of a curved box girder resting on 18 steel columns, steel frame supports for the primary sodium pump guard vessel and the intermediate heat exchange guard vessel, and reactor cavity shield supports.

The reinforced concrete walls and slabs which form the cells, pipeways and head compartment are designed in accordance with the ACI-318-71 Code. For the design of steel structures, the AISC 1970 specifications are used. Modifications to the loads and load combinations peculiar to nuclear plants are considered.

For reinforced concrete cells, the FFTF Project performed a three-dimensional analysis through the use of the STARDYNE computer code. For steel frames three-dimensional space frame analyses were made through the use of ANSYS and MRI/STARDYNE computer programs. Two types of cell liners are used: the cold liner and the hot liner. Two basic cold liner designs are used. One design consists of carbon steel panels welded to I-beam sections embedded in the backup concrete. The second cold liner design makes use of EFCO steel modular concrete forms. These forms were set and welded in place to serve as the cell inside concrete formwork. The concrete was then placed. The hot liner system is designed, in general, to (1) expand thermally as a result of a hot sodium spill without rupture of the liner, (2) separate the sodium from the concrete during a sodium spill, and (3) protect the integrity of the concrete structure by the use of insulating material between the liner and the backup concrete. To maximize free movement of the liner, the plate is anchored only at the top edge, where it joins the cold liner. For walls of cells where large spills of high temperature sodium have been assumed for design purposes, the steel liners are designed to provide an essentially leak-tight barrier to accommodate the cell inert atmosphere and to partially protect the concrete cell structures from sodium attack.

One unique design feature of the liner is the consideration of venting for the steam and other gases generated from the heating of concrete in case of sodium spill. If the steam and gases are not vented, pressure will build up behind the liner and could result in a possible failure of the liner. Two reports on the study of the steel liners were submitted by the FFTF Project: "Analysis of FFTF Reactor Cavity Liners Under Ex-Vessel PAHR Conditions," HEDL-TC-572, and "Cell Liner Integrity Evaluation for the Fast Flux Test Facility," HEDL-TC-823. In the report on the cell liners two sets of assumptions were made to parametrically study the variations in the problem postulating liner failure. Under one set of assumptions, weld cracking was postulated allowing steam to mix with the sodium without any sodium-concrete reaction. In an alternate set of assumptions sodium was allowed to fill the space between the liner and concrete at the wall perimeter where it reacted. Both situations indicate substantial pressure and H_2 generation. It was indicated that of the twelve cells evaluated, ten cells could potentially experience loss of liner integrity by the pressure build-up behind the liner. However, the FFTF Project concluded that the pressure in the cells beyond the design pressure of 10 psig would be relieved to the lower cooler area and the

hydrogen produced from the sodium-concrete reaction would not raise the hydrogen concentration in the reactor containment building above 5 percent. An assessment of sodium spills is made in Section 15.3 of this report.

In the reactor cavity liner report the FFTF Project established a maximum allowable plastic strain of 5 percent as a failure criterion. This maximum plastic strain of 5 percent is established by the FFTF Project on the basis of data on uniaxial elongation of steel which amounts to 40 percent before failure. In order to take factors such as multiaxiality of stress and strain, variation in material properties, etc. into consideration, the FFTF Project adopted 5 percent as a failure criterion, and made a failure analysis for the cavity liner. On the basis of the analysis results the FFTF Project concluded that there is no danger of a failure of the liner due to a sodium spill. The use of a maximum plastic strain of 5 percent appears to be reasonable and the liner designed on the basis of this amount of plastic strain would be expected to perform its function of minimizing the effects from the unlikely event of a major sodium spill. The venting system described and analyzed in the report is conceptual. A scale test of the venting system design was performed by the Project.

We have reviewed the information provided by the FFTF Project on the design and testing of the venting system and we conclude that the venting system as designed is expected to perform its function in the unlikely event of a sodium spill.

The criteria used by the FFTF Project in the design, analysis, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime are in conformance with established criteria defined by codes, standards and specifications acceptable to the staff. We conclude that conformance with these criteria constitutes an acceptable basis for satisfying the intent of the requirements of General Design Criteria 2 and 4.

3.8.3 Design of Other Seismic Category I Structures

The support buildings which surround the reactor containment building consist of the reactor service building, the control building, the auxiliary equipment building, the heat transport system service buildings and heat pump structures. All these buildings have concrete substructures and structural steel frame superstructures.

The structural components consist of slabs, walls, beams and columns. The design method for concrete follows that specified in the ACI-318 Code, and for steel it follows the AISC specifications with appropriate modifications required by the staff to account for loading conditions unique to nuclear power plants. We have concluded that the criteria used in the analysis, design and construction of all seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime are

in conformance with criteria as defined in codes, standards, and specifications acceptable to the staff.

The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures, the structural acceptance criteria, the materials quality control, and established construction technique provide reasonable assurance that, in the event of winds, tornados, earthquakes, and various postulated accidents within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.8.4 Foundations

Foundations of seismic Category I structures are described in Sections 6.1 and 6.2 of the Final Safety Analysis Report. Primarily these foundations are reinforced concrete mats. The major code used in the design of these concrete mat foundations is the ACI 318-71 Code.

The design and analysis procedures that are used for these seismic Category I foundations are in accordance with procedures delineated in the ACI-318 Code. The various seismic Category I foundations are designed and proportioned to remain within limits of the ACI-318 Code for various combinations of dead, live, wind or tornado, pressure and thermal loads.

We have reviewed the criteria used in the analysis, design and construction of seismic Category I foundations. Based on our review, we conclude that the criteria used in the analysis, design, and construction of all Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the staff.

Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

The Project has provided information on computer programs, an experimental stress analysis program, and inelastic analysis methods used for the design of safety-related components. The information provided is adequate and is consistent with Standard Review Plan 3.9.1, "Dynamic Testing and Analysis of Systems, Components, and Equipment." The use of these criteria provide assurance that the stresses,

strains, and displacements calculated are as accurate as the current state-of-the-art permits and are adequate for design of the FFTF mechanical components.

3.9.2 Dynamic Testing and Analysis of Mechanical System Components and Systems

3.9.2.1 Piping Preoperational Test Program

The preoperational test program which will be conducted during startup and initial operation on all safety-related piping, components and supports classified as ASME Class 1, 2, and 3 is an acceptable program. These tests are divided into five phases:

(1) Phase 1 consists of preoperational tests to check the operation of individual components and the calibration of instruments.

(2) Phase 2 are startup tests without liquid metal coolant and fuel. These tests are to demonstrate that the systems and instrumentation are ready for further acceptance testing.

(3) Phase 3 tests are conducted with sodium coolant to verify the overall operational performance of the systems to the extent possible without the nuclear fuel source. Simulated core assemblies with actual control rods and the control rod drive mechanisms are used. Tests are conducted to demonstrate that the thermal/hydraulic and vibration characteristics of the reactor control rods, control rods drive mechanisms, and heat transport system are acceptable at this stage of testing.

(4) Phases 4 and 5 are nuclear startup and power ascension tests to be sure the plant systems, including the control rod drive mechanisms and heat transport system, can withstand power ascensions and subsequent reactor shutdown. Included are low power vibration tests of selected components.

The tests provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design basis operational transients. Compliance with the test program constitutes an acceptable basis for fulfilling, in part, the requirements of General Design Criterion 15, "Reactor Coolant System Design."

3.9.2.2 Seismic Qualification of Mechanical Equipment

The capability of safety-related mechanical equipment to perform necessary protective actions in the event of a design basis earthquake is essential for plant safety. The qualification testing program which is being implemented for seismic Category I mechanical equipment provides adequate assurance that such equipment will function properly under the loads from vibratory forces imposed by the design basis

earthquake and under the conditions of post-earthquake operation. This program is consistent with the applicable paragraphs of Standard Review Plan 3.9.2, "Dynamic Testing and Analysis of Mechanical Systems and Components."

3.9.2.3 Preoperational Flow-Induced Vibration Testing of Reactor Internals

The dynamic response of reactor internals under operational flow transients and steady state conditions were investigated experimentally and analytically in the hydraulic core mockup flow tests. These tests indicated that no adverse flow-induced vibrations developed at either overflow or underflow rates. To verify that these analyses and test results are applicable to the FFTF reactor, a vibration open test assembly will be installed in the reactor to monitor in-vessel vibrations during the acceptance test program. The planned preoperational vibration program provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those experienced during operation. The combination of tests, predictive analysis, and post-test inspection provides adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations of reactor operation without loss of structural integrity. The conduct of the preoperational vibration tests are in conformance with the applicable portion of Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," and constitutes an acceptable basis for demonstrating design adequacy of the reactor internals.

3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports, and Code Support Structure

3.9.3.1 Design Load Combinations and Stress Limits

The specific design basis combinations of loadings as applied to safety-related ASME Code Class 1, 2 and 3 pressure-retaining components and their supports in systems designed to meet seismic Category I standards provide assurance that, in the event of an upset, emergency, or faulted plant condition occurring during plant operation, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. The seismic standards include the requirement that seismic Category I components and supports will continue to perform their safety function after three low level earthquakes (0.05g) and a design basis earthquake (0.25g). The design basis earthquake is identical to the safe shutdown earthquake. The plant will be shut down for inspection if the vibratory acceleration from an earthquake exceeds the 0.05g low level earthquake. The heat transport system has been evaluated for the hypothetical core disruptive accident to ensure its capability of performing post-accident decay heat removal. However, hypothetical core disruptive accident loads are not design basis loads.

Limiting the stresses appropriate to the defined loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

The design load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components are consistent with the recommendations of Regulatory Guide 1.48, "Design Limits and Loading Combinations of Category I Fluid System Components," and RDT F9-IT Standard, "Requirements for Nuclear Components at Elevated Temperatures (Supplement to ASME Elevated Temperature Code Case 1331)." We have concluded that this constitutes an acceptable basis for the design of ASME Code Class 1, 2, and 3 components.

3.9.3.2 Operability of Active Valves

The sodium heat transport system is designed so that the decay heat developed after a reactor scram following a design basis earthquake can be removed by natural circulation with one coolant loop operative. The faulted condition for the FFTF consists of only the design basis earthquake, dead weight, pressure differential, and thermal expansion load combinations. The loss-of-coolant accident in the cold leg reactor downcomer is not a design basis event for the FFTF. Therefore, the operability of active components need be demonstrated only for the loadings associated with normal operation plus the design basis earthquake.

The operability of the primary and dump heat exchanger valves was demonstrated by a series of proof-of-principle tests. The parameters associated with these tests included ranges in operating pressure and temperatures and appropriate thermal transients using full flow water or gaseous nitrogen as the coolant fluid. The dump heat exchanger isolation valve was also evaluated in sodium flow. These tests indicate the valves will function free of vibrations for their operational life.

The FFTF Project's program coupled with the analytical program discussed in Section 3.9.3.1 of this report provides assurance that these active valves can withstand postulated seismic loads in combination with operating loads without loss of structural integrity or unacceptable distortion. These programs demonstrate that the active valves will function when a safe plant shutdown is to be affected. The Project's component operability assurance program constitutes an acceptable basis for meeting the applicable requirements of Standard Review Plan 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

3.9.4 Control Rod Drive Systems

The design criteria and the testing program conducted in verification of the mechanical operability and life cycle capabilities of the reactivity control and open test assemblies are in conformance with established criteria, codes, standards, and specifications acceptable to the staff. The use of these criteria provide

reasonable assurance that the systems will function reliably when required, and satisfies the intent of Standard Review Plan 3.9.4, "Control Rod Drive Systems."

3.9.5 Pressure Vessel Internals

The core support structure provides axial or vertical support and positioning for major reactor components such as the fuel assemblies and lateral support for the instrument tree. It provides the pressure barrier between the sodium inlet and outlet plenums. The core support structure has been analyzed for thermal/hydraulic transients expected during plant and system normal, upset and emergency conditions during the life of the plant and the design basis earthquake faulted condition. Stress distributions and deflections were analyzed and vibration studies were carried out to ensure that the core support structure can withstand any potential vibration.

Mockup and assembly vibration and hydrostatic tests were conducted to demonstrate the structural capabilities of the core support structure.

These analyses constitute an acceptable basis for satisfying the intent of Standard Review Plan 3.9.5, "Reactor Pressure Vessel Internals."

3.9.6 Inservice Testing of Pumps and Valves

The inservice testing of the primary and secondary sodium pumps will be performed on a limited scale since they have no active safety function. The pump inservice test program will be directed toward those pump parameters which give assurance that the FFTF can be safely operated and shut down. These parameters include hydraulic characteristics, pump vibration, bearing temperature, and pump coastdown time. All inservice tests will be performed during the FFTF acceptance test program on a preservice basis to serve as a baseline to inservice testing.

The inservice testing of ASME Code Class 1, 2, and 3 valves will be directed toward those valves which must function for safe operation, shutdown, decay heat removal, and mitigation of consequences of an accident. The FFTF valve inservice test program has been based on requirements of ASME Code Section XI, Division I Articles IWV-3300 through IWV-3700 modified to be consistent with the operating philosophy and design of the plant. Compliance with this code requirement constitutes an acceptable basis for satisfying the intent of applicable portions of Standard Review Plan 3.9.6, "Inservice Testing of Pumps and Valves."

3.10 Seismic Qualification of Safety-Related Instrumentation and Control Equipment

The FFTF seismic qualification program was described in the document JABE/WADCO-02, Revisions 0, 1 and 2, and in Section 3.9 of the Final Safety Analysis Report. Revision 0 of JABE/WADCO-02 included requirements for testing at operating basis

earthquake acceleration levels. These requirements were dropped in Revision 2 because a technical specification requirement was added for plant shutdown upon experiencing an earthquake of 0.05g or greater to evaluate its condition and readiness for further operation. Revision 2 of the document, dated July 1975, had referenced IEEE Standard 344-1971; however later responses to questions had stated that the requirements of IEEE Standard 344-1975 were met.

In general the seismic qualification program for the Class IE instrumentation and control equipment appears satisfactory, however the following concerns were developed during our review.

(1) The description of the seismic testing of the main heat transport system breakers did not state whether any resonances in the 2 to 35 Hertz range were observed. The Project's response showed that resonances at 3, 10 and 25 Hertz were observed and that test levels at these resonances enveloped the required response by substantial margins. We have concluded that this is acceptable.

(2) The Final Safety Analysis Report had indicated that some equipment was subjected to four design basis earthquake test simulations, but that other equipment such as the safety-related batteries, racks, zero time outage motor-generator sets and zero time outage AC and DC panels were subjected to only one such simulation. Later information from the Project clarified that the original seismic qualification specification had included three operating basis earthquake simulations and one design basis earthquake simulation. However, contracts had been let on some equipment before the specification change so these items were subjected to three design basis earthquake simulations in excess of the requirements. The specification remains at one design basis earthquake simulation, therefore all tests meet this requirement. We find this response acceptable.

(3) The Final Safety Analysis Report had stated wiring was tested only over the range of 8 to 30 Hertz. In response to our question the Project provided additional information that showed the wiring had been tested and responded satisfactorily over the range of 1 to 8 Hertz. We conclude that this response is acceptable.

(4) The dump heat exchanger local control panels had additional self-powered readout instrumentation installed on them since the original seismic qualification test was done. This may change the seismic response. The Project has committed to requalifying these panels. We conclude that this will result in acceptable panels.

(5) The safety-related radiation monitors (containment isolation system monitors and cell atmosphere processing system outlet monitors) have been qualified to

meet the seismic criteria. The Project has committed to install additional safety-related radiation monitors outside containment for protecting the control room ventilation inlet. They have committed to qualify these monitors as seismic Category I. We find this commitment acceptable.

(6) The Project has been requested to furnish for our evaluation copies of seismic qualification test reports on selected representative items of safety-related instrumentation and control equipment. These submitted reports should contain: (a) identification of the tested item, (b) the test plan, (c) the maximum seismic acceleration that can exist at the equipment's installed location, (d) description of the operability verification arrangements, (e) description of facilities used to conduct the tests and (f) a digest of the test results showing that the qualified range envelopes the maximum accelerations given in (c). We will report on this concern in a supplement to this report.

With the successful resolution of open item 3.10.(6) we conclude that the Project's program for seismic qualification of safety-related instrumentation and control equipment meets staff requirements and is therefore acceptable.

3.11 Environmental Qualification of Safety-Related Instrumentation and Control Equipment

The environmental qualification program is presented in Section 3.10 of the Final Safety Analysis Report. Our review of this program has developed these concerns that have been resolved through Project action.

(1) The sodium level sensors will operate in a high radiation environment. The staff wanted to know what the effect of this radiation upon the probe would be with respect to material damage and drift of the probe's indication. The Project's response stated that the probe would be exposed to a dose of 4.8×10^{18}nvt, was constructed entirely of inorganic materials and that this dose was not likely to cause any significant change in probe output. We find this response acceptable.

(2) It was not clear from the Final Safety Analysis Report discussion that tests of the containment instrumentation and control penetrations were made using short circuit currents to qualify them, that they were tornado protected or that they would meet the requirements of IEEE Standard 317-1972 and Regulatory Guide 1.63 (1973). The Project has supplied additional information that short circuit current tests were made and that the requirements of the IEEE Standard and the recommendations of the Regulatory Guide were met. We find this acceptable.

(3) The plant protection system radiation monitoring preamplifiers in the containment are located in areas of low radiation levels and would in case of accident have performed their function before the radiation levels become high enough to damage the equipment. Therefore, no particular qualification has been

performed on the preamplifiers. We find this response acceptable since the total dose to which they would be exposed during 20 years of normal operation is about 350 Rads.

(4) Our concern about the permanent magnet flowmeters in the primary sodium loops and their auxiliary instrumentation being qualified to operate in its environment and being calibrated was answered by the Project. Their response described the flux meter probe used to monitor the magnetic field, presented the temperature qualification of the magnet, described the calibration corrections that will be made and stated that an in-situ calibration will be performed after its installation. We found this acceptable.

(5) We were concerned that safety-related instrumentation and control equipment located inside containment and in cells and pipeways had not been qualified against attack by corrosive chemicals. The Project has stated that aerosol formation in the open areas of containment is not a problem and that all of the plant protection system sensors, their field wiring and other equipment are located in inerted cells where the atmosphere is monitored for contaminants. Thus, under normal conditions no potential for corrosion damage to plant protection system equipment exists. Under accident or coolant leakage conditions, the Project has committed to perform cleanup and to inspect and certify the equipment operability prior to resuming operations. The Final Safety Analysis Report is to be modified to so state and we find this commitment acceptable.

The following items have not been resolved and are considered as open items.

(6) We were concerned about the lifetime and possible failure of the metal bellows in the NaK pressure sensors due to mechanical flexing of radiation hardened metal in the bellows since it is exposed to a high radiation environment. We have reviewed additional information supplied by the Project regarding its performance under pressure and temperature cycling in sodium that is satisfactory. The Project has provided more information on the effect of gamma radiation hardening on the life of the bellows. We find this acceptable.

(7) The staff had raised concerns about the qualification for radiation exposure of plant protection system sensors located near or on primary sodium piping and for plant protection system equipment that could be exposed to post-accident pressures greater than atmospheric. The Project is to provide additional information to address these concerns. We will review this new information and report on it in a supplement to this report.

(8) The Final Safety Analysis Report states that the loss of offsite electric power sensors will be environmentally qualified for service under the worst case conditions prior to installation. This information has not been received or reviewed by the staff. We will report on this in a supplement to this report.

(9) With regard to environmental qualification of safety-related radiation monitors, Section 7.3.5.1.4 of the Final Safety Analysis Report presented only temperature and relative humidity specifications. Additional information regarding corrosive chemical attacks, pressure and radiation qualification results should be presented. We will report on this further in a supplement to this report.

(10) The Project's program for qualifying safety-related instrumentation and control equipment for the FFTF as presented in Section 3.10 of the Final Safety Analysis Report does not appear to conform to the current staff requirements with regard to monitoring of environmental conditions in areas where the conditions resulting from loss of the heating and ventilating systems, loss of control of the heating and ventilating systems or deliberate shutdown of these systems may go beyond the limits for which the equipment located therein is qualified. The staff has recommended that either the extreme environmental conditions that can occur under the worst of these postulated events be determined and the safety-related equipment be qualified for these extremes or that the environmental control systems be specified to operate continuously to maintain the environmental conditions within the qualified limits of the safety-related equipment and provide environmental monitoring equipment that will alarm when the environmental conditions exceed those for which the equipment is qualified. This environmental monitoring equipment shall meet these requirements: (a) be of high quality, (b) be periodically tested and calibrated to verify its continued functioning, (c) be energized from continuous power sources and (d) provide a continuous record of the environmental parameters during the time the environmental conditions exceed the normal limits. The occurrence of environmental conditions exceeding the equipment qualification range shall then be reported as an abnormal occurrence and analysis shall be provided to demonstrate that the excessive environmental conditions have not degraded the involved safety-related equipment below an acceptable level for continued plant operation. The Project has stated that Final Safety Analysis Report Section 3.10, as revised, shows that the instrumentation and control equipment is qualified for the most severe environmental conditions that could result from the worst case failure of the environmental control equipment. The staff is presently engaged in reviewing this revised Section 3.10 and believes that contingent on the outcome of the review this response is probably satisfactorily. We will report the final outcome in a supplement to this report.

(11) The Project has been requested to provide for our evaluation copies of environmental qualificaton test reports on selected representative items of safety-related instrumentation and control equipment. These submitted reports should contain: (a) identification of the tested item, (b) the test plan, (c) a listing of the extreme conditons to be accommodated, (d) description of the operability verification arrangements, (e) description of facilities used to conduct the tests and (f) a digest of the test results showing that the

qualified range envelopes the extreme conditions given in (c). We will report on this in a supplement to this report.

Subject to the successful resolution of the open items identified as 3.11(5), 3.11(6), 3.11(7), 3.11(8), 3.11(9), 3.11(10) and 3.11(11) we can conclude that the Project's program for environmentally qualifying the safety-related instrumentation and control equipment meets staff requirements and is therefore acceptable.

4.0 REACTOR

4.1 Description

The Fast Flux Test Facility is a 400 MWt fast neutron test reactor with the primary function of providing the capability for testing candidate fuel and materials for advanced reactors. The core has been designed to accommodate nine instrumented test positions in its 91 hexagonal lattice positions.

The initial core has an active height of 36 inches, a volume of 1034 liters, and at the design power has a peak flux at a row 2 position of 7×10^{15} n/cm^2-sec. The initial peak fuel burnup design basis is 80,000 MWd/MT at nominal coolant temperatures of 600°F, core inlet, and 900°F, core outlet. More advanced conditions are planned for the future but have not been evaluated in this report.

The driver fuel for cores 1 and 2 is composed of mixed plutonium-uranium oxides (PuO_2/UO_2), with compositions of 22.43 and 27.37 wt% Pu/(U + Pu) for the inner and outer enrichment zones, respectively. Natural uranium is used, and the plutonium used has a fissile content of approximately 88 wt% (Pu-239 + Pu-241).

The reference reactor is illustrated in Figure 4.1-1. A plan view of the core appears in Figure 4.1-2, showing the two core enrichment zones and the reflector region.

The removable core components (e.g., fuel assemblies, in-core control rod guide tubes, reflector assemblies, open loops) maintain their axial in-core position due to their weight. Under flow conditions, the fuel and reflector assemblies are also held down due to the hydraulic holddown mechanism. The instrument tree assembly above the core acts as a backup device and would prevent the fuel assemblies from rising. The radial positioning of the in-core components is to be provided by the core restraint system.

The FFTF core has a negative prompt coefficient of reactivity (primarily Doppler) as well as a negative overall power coefficient of reactivity. There is a slightly positive sodium void coefficient in the central region of the core although the reactivity effect of voiding the entire core is negative. The nuclear control function is accomplished by the primary and the secondary nuclear control systems which utilize two independent, but similar, sets of control rods. All control rods scram in the direction of gravity and are prevented by a mechanical stop from traveling beyond the full-in position. The reactivity increase associated with the withdrawal of an absorber rod is limited to 6 cents/sec by the drive mechanism withdrawal rate of 9 inches/min. Rod ejection is guarded against by a ratchet-type "anti-ejection pawl" in the control rod drive mechanism.

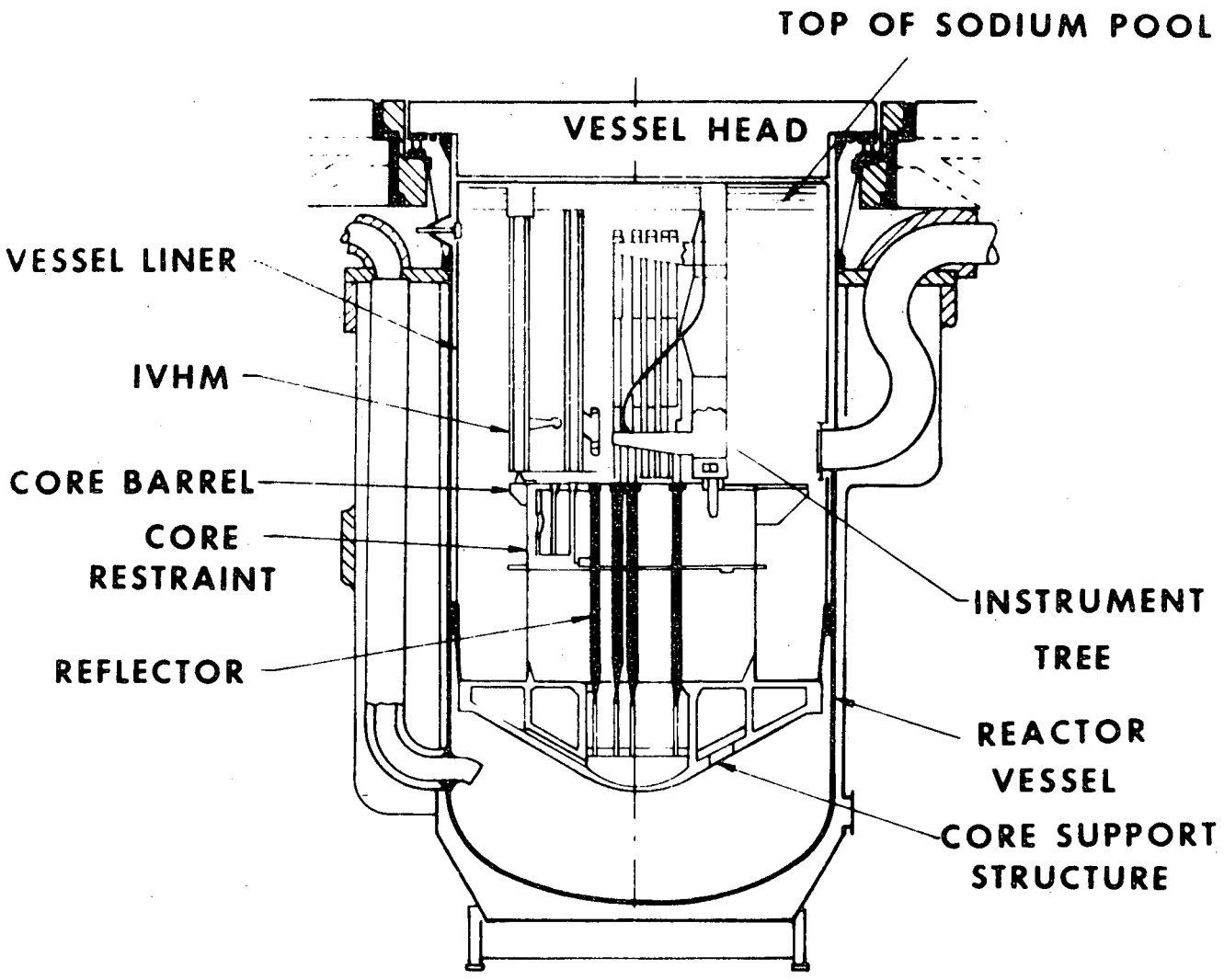
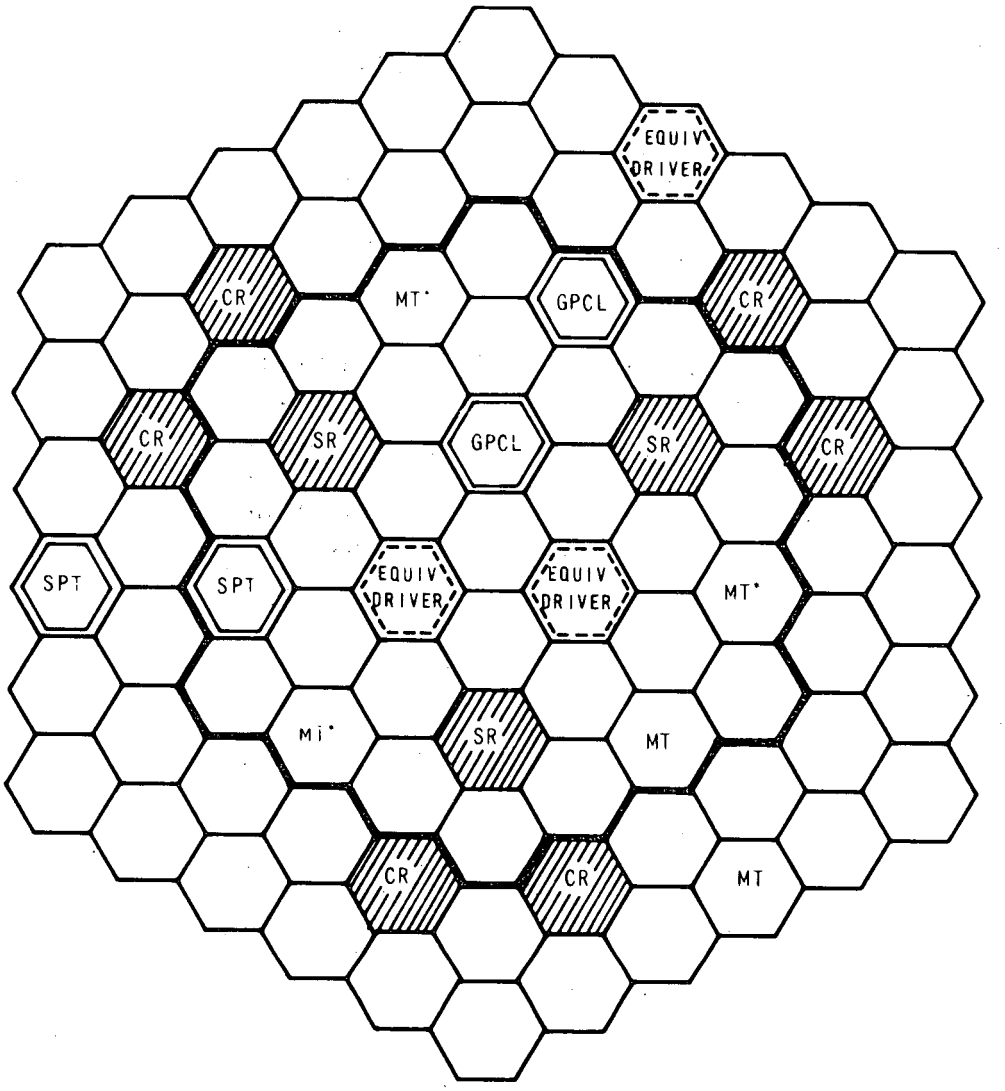


Figure 4.1-1 Reactor Elevation



SR - SAFETY ROD
CR - CONTROL ROD (THE RODS ARE HALF-INSERTED FOR BOC AND ALMOST WITHDRAWN FOR EOC)
MT - MATERIAL TEST
MT* - PRESENT FOR BOL ONLY
SPT - SPECIAL PURPOSE TEST
GPCL - GENERAL PURPOSE CLOSED LOOP

Figure 4.1-2 FFTF Core Map

Design of the core subassembly inlet nozzles makes an inlet blockage that is large enough to cause fuel failure an exceedingly remote possibility. This feature precludes, for example, a Fermi-type blockage accident. The circuitous route that the coolant must travel through the core inlet plenum and subassemblies not only reduces the probability of subassembly flow blockage, but also makes the passage of coherent large bubbles of entrained gas through the core very unlikely. In addition, each assembly will normally be monitored for flow as well as coolant outlet temperature with an alarm on abnormal flow or temperature.

4.2 Fuel System Design

4.2.1 Description

The primary function of the FFTF reactor is to provide the capability of testing candidate materials in environments approximating those anticipated for commercial fast breeder reactors. The FFTF core provides the neutron flux, establishes the power density and operating temperature, and provides locations for experiment placement. The "driver" fuel assembly, which is the basic nuclear power generating component of the FFTF core, contains the fuel, produces heat and provides a high neutron flux test environment. The fuel assembly is designed to accomplish the following:

(1) Position the fuel properly in the core for controlled nuclear reaction;

(2) Provide passages to guide and control the sodium coolant;

(3) Provide shielding to protect the core support structure from excessive fluence; and,

(4) Provide features for proper interfacing with other core components, the core support structure, and fuel monitoring and handling devices.

The fuel assembly is a hexagonally shaped component, 4.575 inches across flats and 12 feet long. It consists of an inlet nozzle, shield orifice assembly, duct, load bearing pads, handling socket, and fuel pin bundle. The bundle is comprised of 217 fuel pins which contain mixed oxide (PuO_2/UO_2) pellets. The fuel pins are positioned in a triangular lattice and spaced by a 0.056-inch diameter, helically-wrapped wire ("wire-wrap"), which is wound clockwise around the fuel pin cladding. The fuel bearing region of the fuel assembly is approximately 36 inches long and is centered 65.5 inches from the bottom end. A 42-inch long fission gas plenum (effective length 37 inches) is provided at the top of the fuel pin to limit internal gas pressure buildup. Within this plenum are a spacer, a tag gas capsule, and a spring system to keep the fuel and insulator pellets and reflectors compressed during handling prior to reactor installation. A list of fuel assembly materials is provided in Table 4.2-1.

The normal fuel pellet stack consists of approximately 150 pellets. Each pellet is about 0.24 inches long and 0.1945 in diameter, of pressed and sintered uranium and plutonium dioxide. The pellets have dished ends (0.0035-inch deep) to reduce axial thermal expansion of the pellet stack.

The driver fuel for cores 1 and 2 will have two enrichment zones: an inner zone of 22.43 wt% Pu/(U + Pu), and an outer enrichment zone of 27.37 wt% Pu/(U + Pu). Since the plutonium isotopic composition changes with time due to the different radioactive decay rates of the various Pu isotopes, these enrichments are based on the Pu isotopic composition as it existed on November 15, 1974.[1] Natural uranium is used in cores 1 and 2, but depleted uranium will be used in later cores.

TABLE 4.2-1

FUEL ASSEMBLY MATERIALS LISTING

Item	Material
Handling Socket	SS-316
Duct	20% CW SS-316
Floating Collar	SS-316
Surface	Chromium Carbide
Load Pads	20% CW SS-316
Surface	Chromium Carbide
Shield Orifice Assembly	SS-316
Inlet Nozzle	SS-316, Chromium Plated
Piston Rings	Inconel 718, Chromium Plated
End Caps	20% CW SS-316
Plenum Spacer Assembly	20% CW SS-316
Spring	Cold Drawn SS-302
Wire Wrap	17% CW SS-316
Cladding	20% CW SS-316

4.2.2 Design Bases and Limits

According to the FFTF Final Safety Analysis Report (Section 4.4.1.1.1), the basic guidelines followed in the assessment of fuel pin structural performance were that the pins (1) must operate (predictably) under expected steady-state conditions, (2) must be capable of sustaining all anticipated transients, (3) must be capable of surviving one "unlikely" (emergency) event, and (4) must maintain coolable geometry during an "extremely unlikely" (faulted) event. Thus, four aspects of operation; e.g., steady-state, anticipated transients, unlikely incidents, and extremely unlikely incidents were considered in the analysis of fuel pin performance.

Fuel pin performance is, aside from considerations of fuel melting and maintenance of "coolable geometry," generally defined in terms of cladding integrity. The integrity of the cladding refers to the ability of the cladding to retain the fuel and fission products. In terms of reactor operation, therefore, the cladding must remain intact (without perforation) throughout steady-state operation, all anticipated transients, and one unlikely (emergency) incident in order to satisfy the FFTF design guidelines outlined above. During irradiation, many factors reduce the cladding's ability to maintain integrity; i.e., to resist perforation and thus retain the fuel and fission products. These factors include (1) chemical attack of cladding by fuel, fission products, and sodium, (2) thermal gradients that cause thermal stresses, (3) thermal and irradiation-induced creep, (4) fatigue, and (5) irradiation-induced swelling. Because the fuel pin integrity is affected by the design of both the fuel and reactor, the cladding integrity limits and criteria must be defined by the reactor designer and must be supported by test data and analyses.

For the FFTF a design overall cladding "integrity limit" was proposed[2] corresponding to 0.7% inelastic deformation (due to thermal creep and plastic deformation). Based on the 0.7% inelastic strain limit, a value of 0.2% strain was selected to cover steady-state operation and an additional 0.1% was selected to cover all anticipated transient operations, leaving a 0.4% inelastic strain increment available to cover a single unlikely event at the end of fuel pin design life. These calculated design inelastic strain limits are not measurable in the reactor. Core coolant temperatures, however, can be measured, so cladding temperature limits corresponding to the calculated inelastic strains were developed.

The FFTF cladding temperature integrity limit was originally established as part of the fuel pin transient design procedure described in the Fuel Pin Final Design Support Document. The procedure for establishing the temperature limit has become known as FCF-213.[2] This limit was calculated using material properties of annealed unirradiated Type 316 stainless steel for temperatures up to 1600°F. An upper cut-off at 1600°F was placed on the integrity limit because there was a sparsity of data on prototypic materials at high temperatures at the time the original calculations were made. Subsequently, a re-evaluation of the integrity limit was performed[3] when analysis revealed that there were two possible transient events that would cause the cladding temperature to exceed the original 1600°F limit.

The highest temperature reported during the two transients was 1642°F. Based on analyses of new tensile data up to 1800°F,[3] coupled with fission gas pressure calculations as functions of burnup, the temperature limits were 1584° and 1742°F at 80 and 40 MWd/kg burnup respectively. Since the boiling temperature of the sodium coolant, assuming zero flow, at the top of the core was calculated to be 1670°F (P = 1.28 atmospheres), the temperature integrity limit was cut off at 1670°F rather than the burst pressure-calculated 1742°F (for 40 Mwd/kg fuel).

The initial core in the FFTF is designed for an average fuel burnup of approximately 45,000 MWd/MT and maximum burnup of 80,000 MWd/MT at a nominal coolant inlet temperature of 600°F (900°F outlet). At higher temperatures with nominal coolant inlet and outlet temperatures of 777°F and 1077°F, respectively (plus 15°F for instrumentation and control band error), the burnup limitation for the hot pin was predicted to be in the range 30,000 to 39,000 MWd/MT, based on fission gas loading considerations only. The maximum (hot channel) steady-state cladding temperatures are reported in the Final Safety Analysis Report to be approximately 1200°F and 1400°F, respectively, for the initial and higher temperature rated operating conditions.

Within the above framework of calculated achievable burnups and strain limits, cladding temperature limits were developed as a function of fuel burnup and transient conditions. These temperature limits were considered applicable only to loss-of-flow transients and slow overpower transients in which quasi-equilibrium of the cladding temperature and gas plenum temperature was assumed; i.e., transients in which the primary contributor to cladding stress is fission gas pressure. The transient design limits were thus set at 50°F below the cladding temperatures which would result in 0.1% strain due to one hour of transient overtemperature during the life of the pin. These correspond to 1460°F and 1580°F for 80 and 40 MWd/Kg burnup, respectively. Note that these temperatures are lower than the temperature integrity limit discussed earlier, which was based on a burst pressure analysis using tensile data on annealed 316 SS. The cladding temperature and strain design limits derived from the above considerations are provided in Table 4.2-2.

For loss-of-flow events then, the relationships between strain, time, and temperature are as follows:

(1) Thermal creep strain of 0.1% is the limiting factor for "upset" loss-of-coolant events.

(2) Temperature limits have been developed to bound strain indirectly because strain cannot be measured in-reactor.

(3) The transient temperature design limits are based on one hour of elevated temperature uniformly distributed throughout the life of the fuel, and are lower than the fuel temperature integrity limits which are based on a burst pressure analysis.

For transient overpower events, i.e., reactivity insertion accidents, fuel and cladding temperature behavior are considered to differ from loss-of-coolant behavior because of differential thermal expansion between the fuel pellets and cladding. The postulated fuel behavior during a "rapid" transient overpower event or a reactor insertion accident is described phenomenologically in Appendix D of the response to staff question 241.3. For slow reactivity insertion accidents (5 cents/second) data

TABLE 4.2.-2

CLADDING STRAIN AND TEMPERATURE LIMITS

Coolant Inlet Temperature (°F)	Maximum Fuel Burnup (Mwd/MT)	Strain (%)	Peak Cladding Temperature (°F)	Operating Conditions
600	80,000	0.2	1200	Normal
--	--	0.3*	1460*	Upsets
--	--	0.7	1580	Emergency
792	40,000	0.2	1400	Normal
--	--	0.3	1580	Upsets
--	--	0.7	1670	Emergency

*Total strain equals 0.2% (steady-state) plus up to 0.1% (transient). The associated temperature limits were selected to be 50°F below the cladding temperature which would result in 0.1% strain due to one hour of transient overtemperature during the life of the pin.

presented in support of the FCF-213 transient overpower reactivity insertion accidents model consist of results of the HEDL P-19 and P-20 experiments on power-to-melt[4,5] coupled with some European irradiation results.[6,7] Design requirements of the plant protection system were based upon a 3$/sec reactivity ramp, which was not considered credible but was used as an upper limit incident in the development of criteria for the plant protection system design. The design analysis reportedly showed that no plastic strain would be expected for any transient overpower event terminated by a primary scram trip (15% overpower), while some plastic strain below the "integrity limit" of 0.7% strain was calculated for a transient terminated by the backup scram trip (25% overpower). It is important to note that from the fuel designer's standpoint, the significant comparison that can be made with the cladding integrity limit for reactivity insertion events is whether the FCF-213 design method predicts failures to occur earlier or later than in a real event. For these comparisons TREAT data were used to determine whether predicted failure times were earlier than observed failure times. For cases where no failure was predicted, but a particular strain value was predicted, a comparison with experimental data could be made only for those tests where some strain was predicted but zero strain was measured. Meaningful comparisons could not be made in terms of relative magnitude of strains because the FCF-213-calculated strain is not a real strain.

4.2.3 Design Evaluation

The lifetime prediction methodology used for the FFTF fuel cladding is known as the "FCF-213 methodology" (the name is derived from the number of the report[2] in which the procedure is first presented). In the FCF-213 methodology, designated cladding plastic strain limits (see Section 4.2.2) are translated into cladding design temperature limits which are related to calculated steady-state or transient lifetimes (steady-state burnup or transient time to failure). These calculated values are meant to be conservative; i.e., they are intended to be less than the true capability of the fuel rods in the FFTF. Thus, they are meant to provide a reasonable margin to failure. The paramount question, therefore, is concerned with whether the FCF-213 methodology, coupled with other FFTF design limits, provides a conservative approach to preclude fuel failure in the FFTF.

Because the FCF-213 methodology establishes the primary bases for the FFTF fuel design, the technical review of the FFTF fuel design is by and large a review or critique of the FCF-213 methodology. Similar methodology was first encountered in reactor licensing during the review of the fuel design analysis in the Clinch River Breeder Reactor (CRBR). Because the FFTF fuel design is virtually identical to that of the CRBR, and since the experimental data used in the derivation and support of the FCF-213 design method for the CRBR is essentially unchanged for the FFTF, the criticism and comments provided in our safety evaluation[8] of the CRBR fuel design analysis are, to a large degree, also applicable to FFTF. Some differences do exist, however, between the planned modes of operation of CRBR and FFTF fuel and between the proposed operational limits. Because of these differences, the conclusions reached during our review of the Clinch River Breeder Reactor are subject to some modification. The following discussion, therefore, is in the form of a critique of the FCF-213 methodology and supporting liquid metal fast breeder reactor transient technology data base and an explanation of the effects that differences in the CRBR and the FFTF design limits and modes of operation have on the acceptability of the FCF-213 approach and FFTF fuel design.

The basic assumption in the FCF-213 methodology is that it is conservative to substitute the creep rate and tensile stress strain relationships of unirradiated, solution-annealed 316 stainless steel for those of highly irradiated 20% cold worked 316 SS; that is, that the former is weaker than the latter and will, therefore, fail sooner. Thus, the proposed cladding strain and temperature design limits are intended to serve only as guidelines to assure that fuel pin actual lifetimes are greater than the design lifetime. To demonstrate that the FCF-213 methodology is conservative for FFTF fuel rods, it must be shown, therefore, that a significant margin to failure would be expected for reactor operations through the single plant protection system-terminated emergency event. That is, it must be demonstrated that the plant protection system will initiate scram and that negative reactivity will be inserted sufficiently rapidly that the calculated 0.7% cladding strain integrity limit will not be violated and that, even if this limit were to be violated,

a significant margin to failure would still exist such that cladding integrity would still be retained in all but a small fraction of the total number of fuel rods.

Models and Analyses Used in the FCF-213 Methodology

The models and cases considered in the FCF-213 methodology are presented in the Final Safety Analysis Report subsection 4.4.1.1.1 for steady-state operation, in subsection 4.4.1.2.1 for transient operation and in Refs. 9 and 10 (as well as the original FCF-213 document.[2] Additional discussion is provided in subsection 15.1.1.1.2. When these models were first encountered and reviewed as part of the CRBR review, it was concluded that several important models had not been demonstrated to be conservative. The reasons for this conclusion are given in the following sections.

Steady-State Models

The assumption that the thermal creep rate of unirradiated annealed 316 SS provides a conservative model for (i.e., is faster than) the thermal creep rate of irradiated 20% CW 316 SS during in-core steady-state operation does not have support via applicable experimental data. The experimental data presented to support this model (Fig. Q241.85-3, CRBR PSAR Amendment 30) compared FCF-213 calculated 0.2% strain with postirradiation stress rupture data for 20% CW 316.[11] This comparison attempted to show that the FCF-213 methodology would have prevented failure since the calculated time to achieve 0.2% strain was less than the actual rupture time. We continue to disagree with the contention for the following reasons:

(1) The referenced data are for low fluence levels (0.2 to 0.9 x 10^{22} n/cm^2, E>0.1 Mev, compared to FFTF peak fluence of approximately 1 x 10^{23} n/cm^2 for 80,000 MWd/t burnup).

(2) Long-term elevated temperature irradiation causes recovery of cold work and loss of strength. Because this results in yield strength values of highly irradiated annealed 316 SS about the same as that of unirradiated annealed 316 SS at test temperatures > 1400°F,[12] it would be expected that creep rates (or creep strengths) would also be roughly equivalent at these temperatures. Moreover, since creep may cause some recovery of cold work, it would be expected that, under the combined affects of long-term irradiation and creep, the creep rates of unirradiated annealed and irradiated cold worked 316 SS would be similar at relevant temperatures.

(3) High fluence leads to loss of ductility. Because rupture life depends upon both creep rate and ductility, the rupture life of highly irradiated 20% CW 316 may fall below the FCF-213 calculated time to achieve 0.2% strain with annealed 316 SS.

Because no new experimental data were presented (since the CRBR review) to support the FCF-213 steady-state models, the above conclusions remain valid for FFTF. In addition, other conclusions regarding wastage allowances, synergistic effects, etc., remain valid. For example, it has not yet been demonstrated that the gross wastage allowance is a conservative model for the notch effect from preferential grain boundary chemical attack or for the potentially enhanced effect of Pu burning in the FFTF on cladding preferential grain boundary attack by fuel fission products. Burning Pu in the FFTF produces approximately 30% less Zr than the burning of U in EBR-II. This may significantly affect the oxidation potential for the cladding. Furthermore, the cladding wastage allowance for fretting and wear, although receiving support from 316 stainless steel pin-disk tests at Westinghouse and Karlsruhe[13,14] and more near prototypic cladding and wire-wrap geometry wear tests at the Liquid Metal Engineering Center[15] has never been verified via measurements on full-length FFTF rods.

It has not been demonstrated, moreover, that irradiation effects, cladding chemical attack, and creep are independent effects (as is implied by the FCF-213 methodololgy) rather than synergistic effects. Nor has it been demonstrated that the worst combinations of irradiation induced creep and swelling and fabrication variables have been considered because of the following:

(1) The "worst" (most conservative) fuel cladding interaction loads would result from no allowance being made for swelling and irradiation creep; this is probably not the "worst" case considered (otherwise it would have been so stated).

(2) It is noted in Ref. 11 on p. 4 and shown in Figure 6-10 of Addendum III to WARD-D-D146 that, even when using an equation that "agreed with the experimental data quite well," the calculated total swelling and creep (due to swelling-enhanced irradiation and thermal creep) is much less (by 0.3% to 1.2% strain, i.e., 25-60%) than the measured AD/D strain.

Transient Models

As in the case of the steady-state models, the transient models were reviewed as part of the CRBR Preliminary Safety Analysis Report review and the shortcomings and weaknesses observed then still exist at present.

(1) The use of annealed unirradiated 316 SS to conservatively model the plastic strain behavior of 20% cold worked irradiated 316 SS during transient events is not substantiated by available mechanical property data.

   (a) The Nuclear Systems Material Handbook (Ref. 12, Vol. 2, Book 1, Section 5, property code 2127 (E-1) p.1.1) notes that above 1400°F annealed unirradiated 316 SS is recommended to describe the flow stress behavior

of cold worked irradiated 316 SS and, since no irradiation hardening occurs in annealed 316 SS at these temperatures (see Nuclear Systems Volunteer Handbook Vol. I, Book 1, Section 4, 2127 (E-1), p. 11.2), annealed unirradiated 316 SS is a suitable model for annealed irradiated 316 SS. Therefore, annealed unirradiated and cold worked irradiated 316 SS are equivalent at temperatures above 1400°F.

(b) Even at 1300°F, at the value of the transient ("upsets") strain limit, 0.1% plastic strain, the conservatism of annealed unirradiated 316 SS is only approximately 2.5 ksi (see Nuclear Systems Material Handbook, Vol. 2, Book 1, Section 5, 2107 (E-1) 1 p. 1-6 and Vol. I, Book 1, Section 4, 2107 (E-1) 1 P. 1.10).

(2) The FCF-213 methodology may be nonconservative because it attempts to preclude failure during overpower events by utilizing only fission gas pressure loading to calculate a "burst pressure." It thereby neglects all fuel cladding mechanical interaction loading. Such loading (for example, differential fuel-cladding thermal expansion and fuel expansion upon melting) has been identified as a primary transient loading mechanism for transient overpower events (see "Safety Engineering Quarterly Report, August - 1975", GEAP-13923-13, p. 24).[16]

(3) The FCF-213 methodology calculations for the design maximum temperatures for emergency events (1580 - 1670°F) may not be conservative for the FFTF because:

(a) A nonconservative material property model is used (that is, annealed unirradiated 316 SS stress strain properties for those of 20% cold worked irradiated 316 SS, as in the preceeding data).

(b) It does not model the fuel cladding mechanical interaction loadings expected for overpower emergency events, but uses, instead, only 100% fission gas release loading.

Experimental Fuel Rod Data Supporting the FCF-213 Methodology

Experimental fuel rod data used in support of the FCF-213 methodology are presented and discussed in Chapter 15 and Appendix C of the Final Safety Analysis Report and in Ref. 9 as well as in several CRBR references. We conclude that these data do not provide adequate verification of the purported conservatism of the FCF-213 methodology coupled with the FFTF design limits as discussed in the next two paragraphs.

Steady-State Fuel Rod Data

Table 15.1.1-3 of the Final Safety Analysis Report contains a comparison of experimental fuel pin lifetimes with predicted lifetimes using the FCF-213 design procedure.

The ratio of attained burnup to design lifetime (i.e., design burnup) is, in most cases, greater than one. These "conservative" ratios of actual lifetime to calculated design lifetime, however, may not reflect the potential reduction in actual lifetime that might result in the FFTF due to some of the design differences in the FFTF rods as compared to EBR-II type fuel rods. FFTF fuel rods, for example, may suffer greater cladding degradation (that is, more irradiation damage) due to the higher fluence-to-burnup ratio in the FFTF. They may also undergo greater fuel/cladding chemical attack from the fissioning of Pu-239 in place of U-235 (an enhanced chemical/mechanical "fuel adequacy" effect). The observed low ratio (<1) of actual lifetimes to calculated design allowables for some subassemblies listed in Table 15.1.1-3 of the Final Safety Analysis Report are ascribed to "wear". These effects should have been accounted for in the design procedure and "conservative" creep model, but they were not. It should be noted, however, that the respective burnups involved for the failed pins (~52,000-61,000 MWd/MTM) are above the design average burnup for the initial core operation and are above the peak burnup for rated operation.

Transient Fuel Rod Data

Liquid metal fast breeder reactor fuel rod transient test data originate from two primary sources: (a) fuel cladding transient tester results attained on non-fueled sections of cladding from fueled and unfueled (plenum) regions of irradiated pins, and (b) TREAT tests. The fuel cladding transient tests data were examined in detail[17] as part of the CRBR Preliminary Safety Analysis Report review. The results of that effort indicated the following:

(1) A majority of specimens failed at temperatures below those described in the CRBR Preliminary Safety Analysis Report as the maximum expected to be allowed for upset and emergency events (1450 and 1600°F respectively).

(2) A comparison of measured incremental strains at failure with the FCF-213 design procedure 0.1% incremental strain criterion (for upset events) showed that failures occurred in some cases at less than the 0.1% strain limit.

(3) The data base was not prototypic of CRBR (or FFTF) because,

   (a) The fluence for data was $\leq 4 \times 10^{22}$ n/cm^2 vs $\sim 1 \times 10^{23}$ (for 80,000 MWd/t burnup),

   (b) A higher fluence to burnup ratio is expected in CRBR/FFTF than in EBR-II,

   (c) The burning of Pu-239 rather than U-235 results in a greater fuel/cladding chemical attack potential in CRBR/FFTF,

(d) A larger potential cladding steady-state thermal creep strain at goal CRBR burnup (due to fuel/cladding gap closures) would be expected to reduce transient strain capability, and

(e) Fuel cladding transient tester data are not relevant to overpower events; tests are gas pressure only and do not treat fuel cladding mechanical interaction.

In rebuttal to these criticisms and in partial response to staff question 241.16 (see FSAR supplement 12, part 2, p. 241.16-7), the following points were made by the Westinghouse Hanford Company, an FFTF Project subcontractor.

(1) "Recent thermal analyses have caused a change in perception of the transient overpower event. It is now recognized that fuel and cladding temperatures for loss-of-flow events behave differently from all transient overpower events, rather than just for very rapid transient overpower events" and "...differential thermal expansion and resultant cladding plastic strain cannot be ruled out in slow transient overpower events."

(2) "Cladding temperature limits are not the appropriate criterion for judging whether the strain limits are met in all transient overpower events. Thus, the implication in BNWL-2071 conclusion 4.1.1 that cladding temperature limits are not adequate is correct..."

(3) "Individual features embodied in the FCF-213 design procedure cannot be extracted for examination on their own merits. Only the net result of application of the FCF-213 design procedure should be compared." (emphasis added).

(4) "The important consideration is not whether the design procedure accurately calculates the strain at failure, but rather whether the time of cladding failures in a transient overpower event was predicted conservatively or not." (emphasis added).

These above quoted responses lie at the heart of the current application of the FCF-213 methodology to the FFTF fuel design. When we reviewed the CRBR treatment of the FCF-213 approach, we noted in regard to observed TREAT tests versus FCF-213 predicted failure times (CRBR PSAR Table 4.2-E and FFTF PSAR Table 15.1.2-4) that, even though most observed failure times were "conservative", (that is, the failure occurred later than FCF-213 predictions), conservatism for CRBR (and FFTF) was not unambiguously demonstrated by the data because of an important nonprototypicality associated with testing in TREAT vis-a-vis operation of a CRBR or an FFTF. In TREAT, fuel rods are tested with fuel that is cracked from the previous EBR-II shutdown. In contrast a prototypic fuel condition (where cracked fuel pieces would

be partially fused together at operating temperature) might result in higher cladding stresses (due to more fuel clad mechanical interaction) and, therefore, shorter times to failure.

Conservatism, relative to FFTF peak burnup, is also not unambiguously demonstrated by the aforementioned low burnup data because of the effect of steady-state power and burnup on gap closure (GEAP-13923-16, p. 34).[18] Lack of steady-state gap closure would reduce transient fuel clad mechanical interaction loads. The two test rods that had a calculated gap closure of 100% (HOP-3-2A, 3-2B) and yet did not fail were terminated tests. Yet they had relatively low burnups that would not have produced prototypic FFTF cladding degradation.

Due to the results of the $3/sec terminated TREAT tests on HOP-3-2A and 3-2B, tests for which 100% gap closures were modeled, conservatism, relative to the FCF-213 strain limit, is also in doubt, for two reasons:

(1) Even though scram was initiated at 0.34 and 0.38 seconds, respectively (Figs. C-2 and C-3, Ref. 9), incremental transient strains exceeding the FFTF transient strain limit of 0.1% were observed (0.15 and 0.2%, respectively, p. C-2, Ref. 9).

(2) In addition, cracked fuel in TREAT may reduce cladding loading and the potential for additional strain.

Confidence in the degree of conservatism inherent in the FCF-213 methodology as applied to FFTF is not enhanced by any of the data because of the expected further steady-state degradation in FFTF, as discussed above.

With regard to cumulative strain during TREAT and FCTT tests (Figs. HEDL 7609-72.1 and 7609-72.4 in HEDL TI-7603503)[19] presented at a CRBR fuels meeting, October 13-14, 1976, $3/sec TREAT terminated test (HOP-3-2A and 3-2B) rods were subsequently sectioned and samples tested in the fuel cladding transient tester. These data do not appear to be conservative relative to the FFTF for three reasons:

(1) The samples tested in the fuel cladding transient tester were apparently not from the highly strained portions of the fuel elements since the incremental TREAT strains cited (0 and 0.4% respectively) do not agree with the maximum incremental strains measured on the rods (0.15 and 0.2%, respectively, Ref. 9, p. C-2).

(2) The fact that the failures of the fuel cladding transient tested samples fell within the scatter band for the test conditions reflects the low damage these samples received in TREAT, as noted above.

(3) The TREAT tests are nonprototypic (cracked fuel, see above) and the cladding degradation is nonprototypic (see discussion earlier).

Plant Protection System - Speed of Response

To ensure that fuel rod integrity will be maintained through the design emergency event, the plant protection system must introduce sufficient negative reactivity that the cladding design strain limit is not violated. The FCF-213 methodology is supposed to provide sufficient margin to failure that, if the plant protection system were to operate sufficiently rapidly, the strain limit would not be violated, and even if it were, that at most only a small fraction of failures would be anticipated.

The plant protection system response speed was based on an FCF-213 analysis of an FFTF hypothetical $3/sec reactivity insertion (Ref. 2, Section 3.3, Ref. 9, p. 15) and was correlated to the FFTF 0.4% incremental transient strain limit (0.5% if at beginning of transient life; i.e., no upset events). That is, the plant protection system would have to scram and introduce sufficient negative reactivity such that during a $3/sec transient, the 0.7% total strain integrity limit (and 0.4-0.5% incremental strain) would not be reached. Likewise, for upsets, 0.1% incremental strain would be precluded. The FCF-213 analysis of the HOP-3-3c test (in Ref. 9), indicated that 0.7% strain would be achieved in about 0.7 seconds (Ref. 9, Fig. B-13), and perhaps, therefore, sufficiently beyond the 0.45 seconds established as the necessary plant protection system response time to preclude fuel pin failures in FFTF. More recent analyses, presented in response to staff question 241.9, indicate that the termination time for a $3/second reactivity insertion event could be 0.422 seconds. Thus, analyses of the FSAR transient overpower events with the FCF-213 design procedure indicate that the plant protection system scram trip settings (15% overpower for primary scram trip, 25% overpower for secondary trip) preclude any cladding straining for anticipated (upset) events, and that cladding straining is limited within the failure criterion of 0.7% strain for unlikely (emergency) events (up to $3/second).

The key to the acceptability of the plant protection system response time and its relationship to the transient strain limits is therefore concerned with the acceptability of the 0.7% cladding strain integrity limit. This limit appears to have originated from an extrapolation of some Dounreay Fast Reactor (DFR) data (see CRBR PSAR Fig. 4.2-1). These data were interpreted to show that a creep rupture strain of 0.7% would be obtained on 20% cold worked 316 stainless steel irradiated at approximately 1000°F to a total fluence of $\sim 3 \times 10^{22}$ n/cm^2. These DFR data were obtained, however, on British steels, which have somewhat different creep-rupture properties than U.S. steels (even of the same alloy number). Moreover, as shown in the above-cited CRBR PSAR figure, many of the EBR-II data points plotted on the figure fall below the DFR data line. The result is that although the 0.7% cladding strain integrity limit appears reasonable, the data are not in total support of this value.

4.2.4 Fuel System Design Evaluation Summary and Conclusions

Margin-to-failure in FFTF fuel rods is predicted by the FCF-213 design methodology. The basic assumption of the FCF-213 methodology is that it is conservative to substitute the creep-rate and tensile stress-strain relationships of unirradiated, solution annealed 316 SS for those of highly-irradiated 20% cold-worked 316 SS; that is, that the former is weaker than the latter and will, therefore, fail sooner. Thus, it is a calculational tool that is used to predict a conservative time to failure based on preselected strain limits which bear no direct relationship to critical experimental deformation. Although we concur that the concept of establishing cladding strain may be a useful failure criterion, we believe that the conservatism of the FCF-213 coupled with FFTF limits has not been demonstrated unequivocably for FFTF fuel rods with regard to the models in the methodology or the experimental verification of the models. A better case exists for the plant protection system speed of response requirements during an emergency event (assuming that the 0.7% cladding strain integrity limit is an acceptable limit; this assumption is also disputable).

Although lacking in total support for the FCF-213 fuel design methodology, the FFTF fuel design does, nevertheless, have considerable backing in the way of applicable, if not completely prototypic, test data. In addition, FFTF, as a test reactor, will have a mode of operation not typical of commercial power reactors, which lends itself to the possibility of frequent, periodic surveillance of the fuel; e.g., refueling operations will take place at approximately 100 day intervals, at which time some surveillance, and even replacement, of fuel should be possible. We believe a strong fuel surveillance program is necessary because of the differences between EBR-II, TREAT, General Electric Test Reactor, and Dounreay Fast Reactor test conditions and FFTF; if truly prototypic steady-state and transient data were available, the behavior of FFTF fuel could be predicted with greater confidence. Surveillance itself will not supply the desired data on FFTF fuel transient behavior, but it should provide some confirmatory information on the steady-state analysis that if coupled with suitable post-irradiation examination and testing could provide additional transient behavior information. For example, portions of irradiated FFTF rods could be tested in TREAT (TREAT tests do, however, suffer from the lack of a preconditioning period).

FFTF has an additional advantage, in comparison to the CRBR, for example, in that the fuel will be operated to a relatively modest burnup; namely 80,000 MWd/t peak, ~45,000 MWd/t average for the initial core and ~40,000 MWd/t peak for rated conditions. Thus the demands on the fuel, in terms of total fuel duty and cumulative damage, are relatively unpretentious and are not expected to result in a significant challenge to fuel rod integrity. Moreover, there is still ongoing and planned research and development on liquid metal fast breeder reactor fuels (aside from that planned in the FFTF itself) which should provide further support for the design. Thus, we

conclude that there is reasonable assurance that the cladding integrity of FFTF fuel will be maintained and that significant amounts of radioactivity will not be released through fuel pin failures.

4.3 Nuclear Design

4.3.1 General

The Project summarizes the Nuclear Design and Analysis for the FFTF in Section 4.5 of the Final Safety Analysis Report. We and our consultants at Brookhaven National Laboratory have evaluated the analysis presented in this section, have assessed the FFTF engineering mockup critical experiments on ZPR-9 at Argonne National Laboratory, and have performed a number of independent physics calculations. In general, it is our opinion that:

(1) As far as safety analysis is concerned, the methods used by the Project are adequate with sufficient conservatism to cover uncertainties; and

(2) The SEFOR experiments and the FFTF engineering mockup critical experiments provide good confirmatory support to the methods and data presently used to evaluate the nuclear design, especially the safety-related parameters.

The staff's independent confirmatory analysis supports the position of the Project on key coefficients and parameters. Thus we conclude that, from a safety point of view, the FFTF nuclear design is adequate and well understood. The startup physics testing program planned for the FFTF should further confirm the Final Safety Analysis Report evaluation.

4.3.2 Power Distributions

The Project has presented a detailed discussion of the power distribution analysis and experimentation for the FFTF. The analysis includes calculation of power profiles and power peaking factors for the first six operating cycles for the FFTF using both a 2-D and 3-D diffusion theory capability. The calculated values are adjusted using bias factors (calculation-to-experiment ratios) derived from the engineering mockup critical experiments performed on ZPR-9. Measurements of the ^{239}Pu fission rate distribution, using ^{239}Pu foils, were performed over a midplane trisection and axially along the core centerline.

4.3.3 Reactivity Coefficients

The reactivity coefficients are expressions of the effect on neutron multiplication of changes in such core conditions as power, temperature, and coolant voiding. These coefficients vary with fuel burnup. The Project has presented the values of these coefficients and has also evaluated the accuracy of these calculations.

We have performed independent calculations, reviewed the Project's calculated values of the reactivity coefficients and conclude that the Project's calculations adequately represent the full range of expected values. We also conclude that the reactivity coefficients used in the safety analysis conservatively bound the expected values including uncertainties.

The predicted total power coefficient in the power-operating range is negative for all reactor conditions through core life thus satisfying the requirements of General Design Criterion 11. The Project will measure the sodium density coefficient and the power coefficient during startup tests to check the calculated values and to further ensure that conservative coefficient values were used in the accident analysis.

We have performed a number of physics calculations using the DIF2D multigroup diffusion theory code in order to compare them with the Project's values. Both HEX and R-Z calculations were made, all with 27 energy groups, ENDF/B-IV cross sections and a beginning of life core composition. The results indicate sufficient agreement (i.e., $\pm$ 20%) for the reactivity-coefficient parameters considered, namely, sodium density worth and Doppler coefficient.

We evaluated also the reactivity effects resulting from core geometry changes. The Project calculated a power coefficient from the geometry effects to be -0.12¢/MW resulting from a -0.10¢/MW contribution from axial expansion, a +0.12¢/MW contribution from radial expansion and bowing, and a -0.14¢/MW contribution from differential motion of control rod tips. When the Doppler and sodium coefficients are included, the minimum power coefficient (for BOC conditions) become -0.36¢/MW. We conclude that these values are acceptable.

4.3.4 Control Rod Worth

We evaluated the Project's analysis and interpretations of experiments in the area of control rod worth. In addition, using the calculation methods described above, we performed some spot-check calculations of individual rod worths in the beginning of life core and of all the control rods. For a single control rod worth (rod core position 3504, all other rods out), we calculated 2.85$ while the Project calculated 3.17$. For the control rod system worth (all six rods in with all other rods withdrawn), we calculated 17.14$ while the Project calculated 21.47$. This discrepancy (about 25%) seems large until the differences in the calculation techniques and models are considered. In any event, the staff-calculated values indicate sufficient shutdown reactivity.

We have reviewed the calculated rod worths and the uncertainties in these worths, based upon appropriate comparison of calculations with experiments. On the basis of this review, we have concluded that the Project's assessment of reactivity control requirements is suitably conservative, and that adequate negative reactivity

worth has been provided by the control system to assure shutdown capability assuming the most reactive control rod assembly is stuck in the fully withdrawn position. We have concluded also that the proposed control rod worths are acceptable for use in the accident analysis. In addition, the values of these control parameters will be verified in startup tests.

4.3.5 Stability

We concur with the Project's conclusion that, even with very conservative assumptions, the FFTF reactor will not experience any operating difficulties associated with inherent feedback instability. The "best-estimate" reactivity feedback coefficients at full power are all negative. A parametric study did include a maximum positive value for the bowing coefficient. Even for the worst case considered, the significant increases in power gain occurred only at frequencies < 0.01 Hertz which can be readily controlled.

4.3.6 Analytical Methods

The Project has described the computer program and calculational techniques used to calculate the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict experimental results. We conclude that the information presented adequately demonstrates the ability of these analytical methods to calculate the reactor physics characteristics of the FFTF reactor.

4.4 Thermal-Hydraulic Design

4.4.1 Design Basis

The primary thermal-hydraulic design basis for the FFTF core is that adequate cooling be available to remove both fission and decay heat under all design basis events including postulated accidents.

The nuclear core is designed to produce 400 MWt power and is cooled by upward liquid sodium flow. A list of the major core thermal-hydraulic parameters is provided in Table 4.4-1.

The heat generated in the core is removed via the three sodium loops of the primary heat transport system. The heat is then transferred from each primary loop to an intermediate heat transport system loop via an intermediate heat exchanger. The heat in each of the three intermediate loops is then transferred to the atmosphere via four sodium-to-air heat exchangers in each loop.

TABLE 4.4-1

THERMAL-HYDRAULIC CHARACTERISTICS OF THE FFTF

Design Thermal Output, MWt	400
Primary Sodium Flow Rate gpm	14,500
Secondary Sodium Flow Rate gpm	14,500
Number of Fuel Assemblies	73
Number of Fuel Rods/Bundle	217
Rod O.D., in	.23
Active Fuel Length, inches	36
Rod Pitch, inches	.286
Core Coolant Inlet Temperature °F	600
Core Coolant Outlet Temperature °F	900
Vessel Coolant Outlet Temperature °F	858
Maximum Cladding Surface Temperature	1154
Maximum Coolant Temperature °F	1140
Maximum Core Pressure Drop psi	140
Flow Area Per Assembly in^2	7.2562
Core Heat Transfer Area ft^2	2861.5
Avg. Linear Heat Generation Rate $\frac{KW}{ft}$*	8.01
Maximum Linear Heat Generation Rate $\frac{KW}{ft}$	14.4

*First Cycle

4.4.2 Thermal-Hydraulic Limits

In order to assure that adequate cooling is available to remove fission and decay heat for all design bases conditions, sodium coolant temperatures should be maintained below the boiling temperature. Once boiling is initiated, rapid vapor formation in the core channels can lead to highly degraded heat transfer and channel flow, which in turn could lead to clad failure. The FFTF Project has proposed a temperature limit of 1670°F at the inner surface of the fuel cladding for "at power" accidents. This limit corresponds to the saturation temperature of sodium at the top of the fuel assemblies under static conditions. "At power" events are those characterized by flow degradation prior to scram. For the 3-loop operation loss-of-flow accidents analyzed, predicted cladding temperatures did not exceed this limit. Since the predicted sodium saturation temperature at the time of the peak cladding temperature will be considerably higher (approximately 100°F) than the proposed 1670°F limit (due to the increase in saturation temperature because the pressure from the static head of sodium is augmented with the pump pressure), coupled with inherent conservatisms of the application of this limit (i.e., applied to inside rather than outside cladding surface), we have found this limit acceptable.

For events characterized by the simultaneous insertion of the control rods and loss of power to the pumps (loss of all power), the FFTF Project has proposed a bulk sodium temperature limit of 1705°F, applied to the hot channel coolant.

This limit is based on the sodium saturation temperature at the top of the fueled region of the core under the static head of sodium in the upper plenum and fission gas plenum regions of the core. This limit does not preclude sodium boiling in the fission gas plenum region due to the decreased saturation temperature in this region.

The FFTF Project presented analyses of sodium boiling to evaluate the consequences due to conditions of low power and flow which might exist under certain hypothetical accident situations which were not part of the design bases. These analyses indicated that cladding temperatures would not exceed the melting temperature and that the decreased density in the boiling channels would promote additional flow to those channels, providing a self-correction cooling mechanism. However, the analyses were based on inlet and outlet conditions predicted by IANUS calculations, which did not account for the effects of sodium boiling on the system hydraulics. No experimental data were presented in support of these analyses.

Based on the above, we recommend that the proposed sodium temperature limit of 1705°F be reduced to the saturation temperature above the fuel assembly outlets under stagnant conditions, which is 1670°F.

We would agree to the use of the 1705°F limit once sufficient experimental data or adequate analyses are available to demonstrate the acceptability of sodium boiling in the fission gas plenum and/or core region regarding its effects on core flow and core heat transfer.

4.4.3 Flow Distribution

The core flow is divided among three flow paths, or orifice zones, each corresponding to different power regions of core assemblies. The flow orificing is designed to minimize radial core temperature gradients so as to equalize the maximum steady-state cladding temperature in each region. Table 4.4-2 lists the flow paths and the percentage of total core flow to each path.

The Project reported the results of tests performed to confirm orifice zone flow rates. These tests were performed in water with unheated flow paths that modelled the hydraulic resistances of the fuel assemblies. The results showed that flow distributions agreed well with design code (FLODISC) predictions. The Project has also stated that pretest predictions on the FFTF core flow distributions will be performed and compared against actual data. Based on the above tests and commitment for verification, we find the flow distribution design acceptable.

TABLE 4.4-2

REACTOR FLOW AND TEMPERATURE DISTRIBUTION

Component	Location (See Fig. 4.6-1)	No. of Assemblies	Nuclear Heat Rate MW	Nuclear Heat Rate %	Flow (%)	Flow (10^5 lb_m/hr) Initial	Flow (10^5 lb_m/hr) Rated	Nuclear Heating Outlet Temperature[a] (^oF) Initial	Nuclear Heating Outlet Temperature[a] (^oF) Rated
Safety Rods	3-8	3	0.71	0.18	0.62	1.07	1.08	674	866
Control Rods, 50% In	3-9	6	1.85	0.46	1.24	2.14	2.16	797	889
Fuel Assemblies	3-10	73 + 3	380.65	95.16	82.29	142.20	143.84	900	1092
Materials Tests	3-11	2	0.37	0.09	0.24	0.41	0.42	700	892
Fixed Shims	3-12	3	0.68	0.17	0.41	0.71	0.72	707	899
Inner Reflectors	3-13	57	7.96	1.99	3.44	5.94	6.01	750	942
Outer Reflectors	2-14	48	2.90	0.72	0.95	1.64	1.66	897	989
Inner Shielding	2-15	-	2.46	0.62	1.18	2.04	2.06	735	927
Outer Shielding	2-16	-	1.50	0.38	1.37	2.37	2.40	671	863
Sodium and Structure[b]	2-19	-	0.91	0.23	-	-	-	-	-
IVS and Annulus[c]	5-6	-	(0.50)[e]	-	1.43	2.47	2.50	-	-
Thermal Liner[d]	5-18	-	-	-	6.29	10.88	11.00	600	792
Gas Vent	1-6	-	-	-	0.11	0.19	0.19	600	792
Closed Loops	-	4	(3.91)[e]	-	-	-	-	-	-
Leakage	-	.	-	-	0.44	0.75	0.76	858	1050
Reactor Vessel	1-19	199	400.0	100.0	100.0	172.8	174.8	858	1050

[a]Average outlet temperatures based on 600°F inlet temperature for initial operation and 792°F for rated operation.

[b]Axial and radial neutron leakage from core barrel.

[c]Approximately 500 kW produced if 57 IVS positions contain fuel assemblies.

[d]Actual temperature at overflow, due to heat conduction, is 892°F. Flow includes 1.56% of reactor flow at ejector nozzle.

[e]Not included in 400 MW total power capability.

At low reactor flows, significant redistribution of flow can occur due to the radial density redistribution of the coolant. This redistribution effect is shown in Figure 4.4-1 for various power-to-flow ratios. The FFTF Project contends that under loss-of-flow type of accident conditions, redistribution will tend to increase flow in the hottest coolant channels and reduce flow to the cooler channels, resulting in a flattening of the radial temperature profile. The analysis code (IANUS) used to predict accidents characterized by undercooling of the core utilizes a redistribution model based on a conservative approximation of the steady-state FLODISC predictions. The FFTF Project also provided a comparison of redistribution effects on cladding temperature, predicted by a dynamic version of the FLODISC code, to the redistribution effects on cladding temperature predicted by IANUS. The comparison showed that the IANUS calculation conservatively predicted hot channel flow compared to the dynamic FLODISC predictions. We have found the approach described in the IANUS code for calculating flow redistribution to be acceptable.

4.4.4 Thermal Power Distributions

In the Fast Flux Test Facility, high heat transfer rates exist between the coolant and the cladding surface. Because of these high heat transfer rates, the cladding temperature is closely coupled to the bulk sodium temperature. Since the peak sodium temperatures occur at the active core outlet, then peak cladding temperatures also occur at this location. Because of this, steady-state peak cladding temperatures are relatively independent of the axial flux shape. This is also true for undercooling type of transients.

Radial flux peaking determines the power in the hot assembly and therefore the hot channel coolant enthalpy rises. The radial peaking factor for the inner peak zone of the core is 1.365 for the first cycle, with a maximum of 1.411 occurring in cycle 5.

4.4.5 Upper Plenum

The upper plenum of the vessel contains approximately 5000 cubic feet of sodium with a cover gas volume of approximately 500 cubic feet to accommodate system inventory changes due to density changes, makeup, and overflow. Under steady-state conditions, the sodium in this volume will be well-mixed and at a relatively uniform temperature. Following a reactor trip, however, the core outlet temperature rapidly decreases. This can potentially result in thermal stratification of the sodium in this plenum region, wherein cold sodium exiting the core remains in the lower regions of the plenum and hot sodium remains in the upper regions of the plenum. The Project provided an evaluation of their analysis model (VARR-II computer code) based on tests performed in a 0.268 scale model of the FFTF upper plenum. These tests were performed with water and salt injection to model the sodium density change at the core outlet following scram. The approach used was the modeling of hydraulic dimensionless numbers (i.e., Euler's Number) to simulate the behavior of

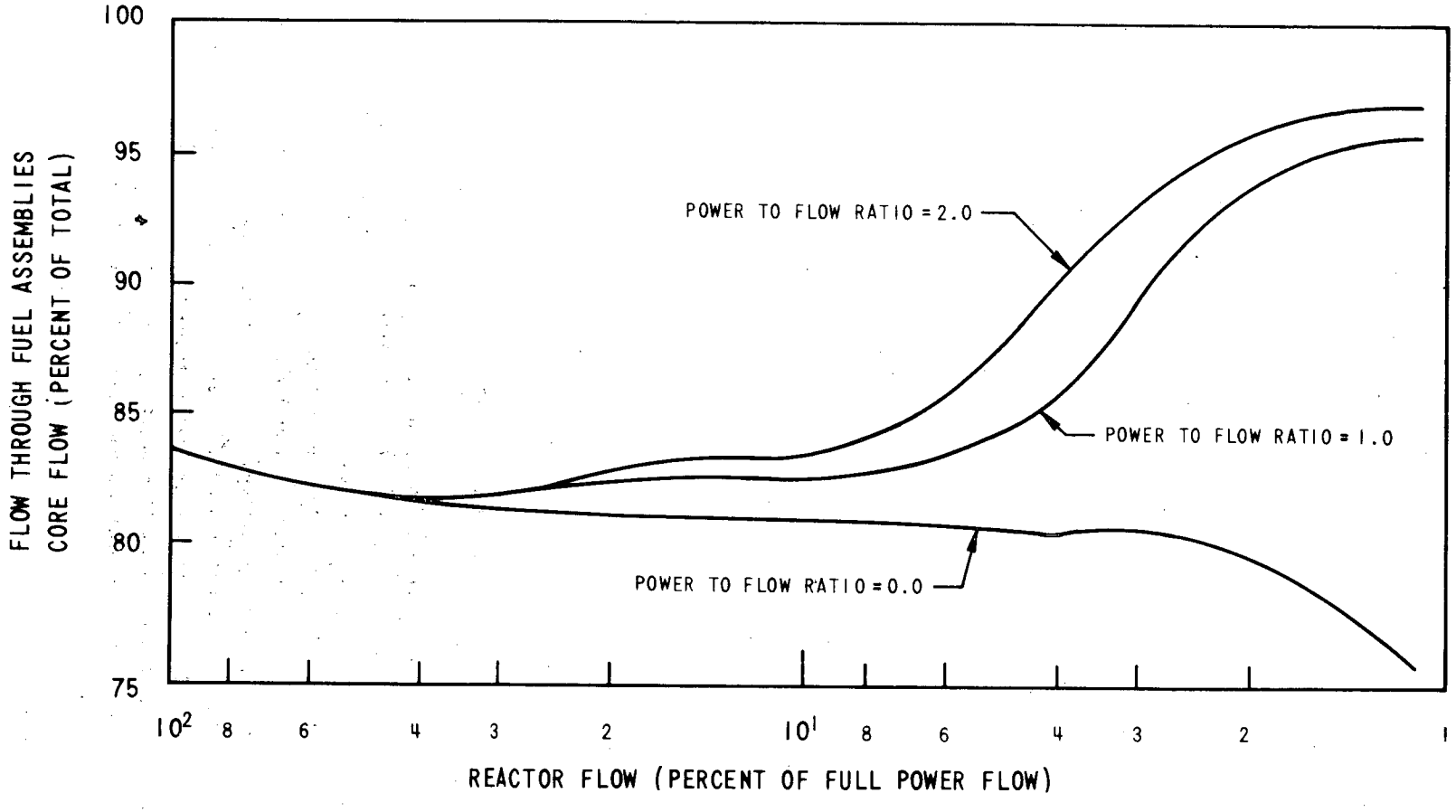


Figure 4.4-1 Fraction of Flow Through Fuel Assemblies

sodium. From the evaluations made on these tests, the Project has stated that significant thermal stratification in the upper plenum will not occur and the predicted transient plenum mixing behavior used for the structural design analysis is conservative.

For the design of the upper plenum components, the Project has compared the thermal transients predicted by the VARR-II code to the transients used in the structural design of upper plenum components. An example is reproduced from the Final Safety Analysis Report as Figure 4.4-2.

Based on our review of the comparisons made by the Project of the VARR-II code to the 0.268 scale mixing data, as well as the comparisons made of the thermal transients predicted by the VARR-II code to the thermal transients used for the structural design, we have reasonable assurance that the plenum mixing models used for structural design of upper plenum components are conservative and therefore acceptable.

While we believe that the comparisons made between the VARR-II code predictions and the 0.268 scale data help substantiate the validity of the VARR-II code to predict plenum mixing, verification of the VARR-II code should also be performed via comparisons to the actual mixing behavior of the FFTF upper plenum. This is necessary in order for the staff to find the transient behavior of the plenum as predicted by the VARR-II code acceptable for use in thermal-hydraulic safety analyses. We recommend that comparisons demonstrating the ability of the VARR-II code to predict mixing in the FFTF upper plenum be submitted to the staff for approval after completion of the startup test program.

4.4.6 Design Analysis Methods

The FFTF Project has utilized a number of computer programs in both the steady-state and transient analysis of the design.

We have not reviewed these codes in the detail consistent with a normal review for light water reactors. Brief reviews of the IANUS, FORE-II, and FLODISC (steady-state) codes were performed, based on available documentation and responses to staff questions. Because of the design requirement that the coolant remain in the liquid phase (no boiling) for all normal and expected operating conditions, and because of the physical properties of liquid sodium (low heat capacity and high heat transfer coefficients compared to water as the coolant), many analysis areas of concern in light water reactor thermal-hydraulic design (e.g., critical heat flux) are not factors in the FFTF analysis. The Project has identified the significant parameters in the analysis and conservative values were assigned to them for the design analyses.

Additional sensitivity studies on the effect of other parameters and/or assumptions were also performed at our request.

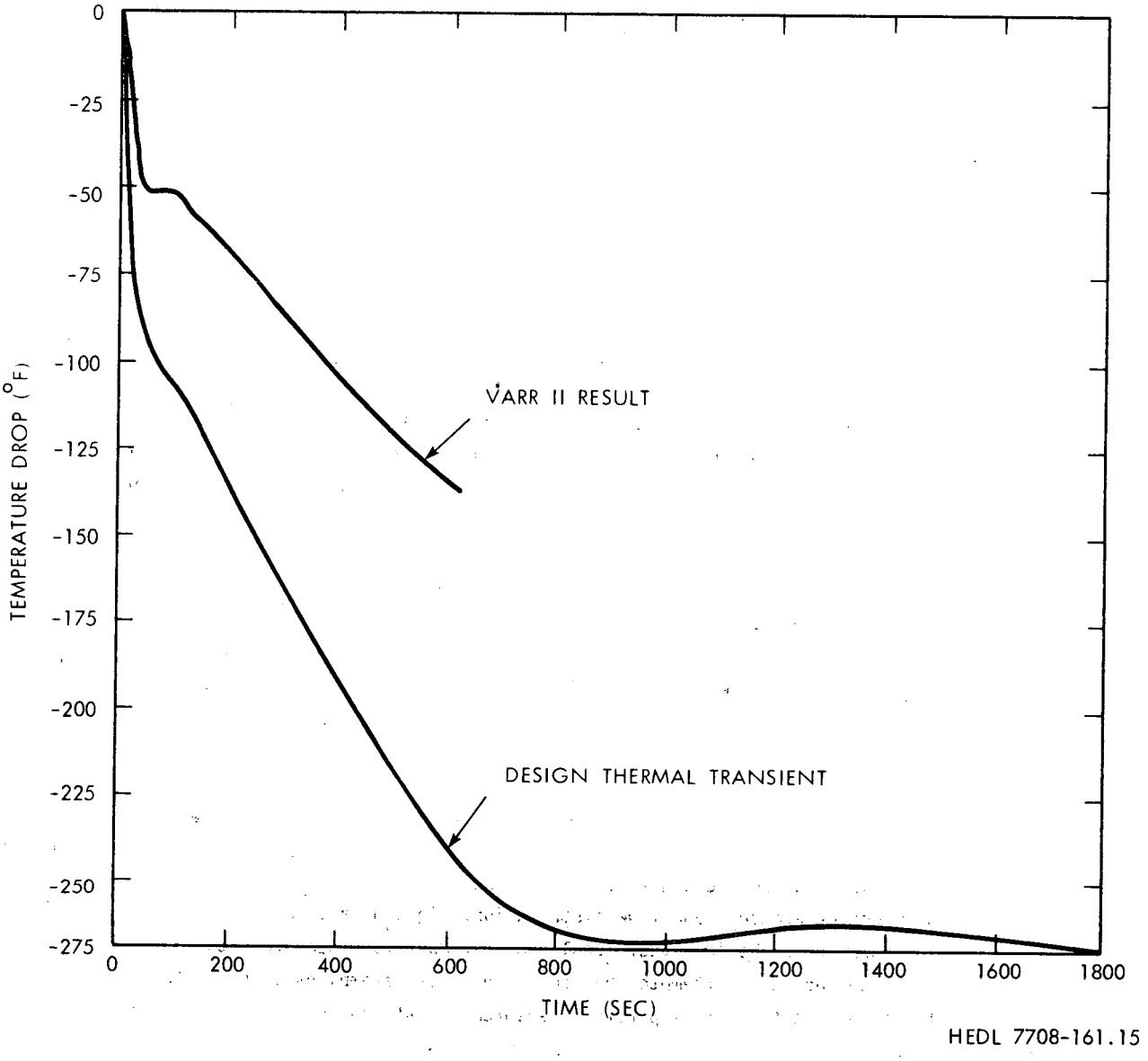


Figure 4.4-2 Reactor Vessel Outlet Temperature Drop (Normal Scram)

The sensitivity studies were performed for hot leg piping ruptures which provided a severe undercooling of the core.

Process variables investigated include:

- stored energy in the fuel (gap conductance)
- axial and radial nodalization in the fuel
- broken pipe reverse pressure drop
- power level
- protection system response time
- decay power
- plenum mixing
- fuel assembly duct effective mass.

Additional calculations in non-piping rupture transients included minimum decay heat and the application of hot channel factors to average channel dynamic calculations. The results of this study indicated that no significant changes in the predicted cladding temperature would be expected if parameter values were varied within their estimated uncertainty band.

Since none of the thermal-hydraulic safety analysis codes have not been verified under prototype conditions, we cannot conclude at this time that they are acceptable for safety analyses. While it is not our intent to perform a detailed review of these codes, we recommend that they be verified during the startup testing phase of plant operation, and any discrepancies between code predictions and test data be resolved prior to full power operation.

The Project provided preliminary plans for code verification during both startup testing and during power operation. These plans provide primarily for verification of the IANUS and FLODISC codes hydraulic response, etc., in the areas of pressure drop, power generation, and flow distribution. In general, we concur in the approach taken; however, we believe that local sodium temperature measurements in the active core region should be attempted by use of the test assemblies in order to allow better interpretation of temperature measurements at the core exit measurement station.

Because we have not reviewed, nor has verification been provided for, the safety analysis code used in support of the Final Safety Analysis Report analyses, we request that pretest predictions be performed with all of the major analysis codes. For those codes which are core models only, inlet and outlet conditions should be obtained from the system code IANUS predictions. We recommend that a report documenting the results of these pretest predictions be made available for staff review prior to the performance of the test.

4.4.7 Thermal-Hydraulic Uncertainty Analysis

Uncertainties in the thermal-hydraulic performance of the core are accounted for in the hot channel factors applied to the design. A table of the hot channel factors used in the thermal-hydraulic design are given in Table 4.4-3.

We have reviewed these factors against available experimental information. In many cases, insufficient data were available to confirm the uncertainty factors reported. For those factors where insufficient data were available for a statistical representation of the uncertainty, the Project has chosen an uncertainty factor consistent with available data. The hot channel factors are divided into direct and statistical factors. The method of combining the statistical factors is consistent with light water reactor practice. We have taken exception to the treatment of power level uncertainty as a statistical factor and have recommended its treatment as a direct factor, consistent with light water reactor practice. Revised estimates by the Project of the power level uncertainty are considerably lower, and treatment of the revised power level uncertainty as a direct factor results in approximately the same value of overall hot channel factors originally proposed.

The FFTF Project stated that since the thermal hydraulic design was based on the hot channel factors originally submitted in the Final Safety Analysis Report, any changes to them in this stage of the design would require extensive reanalyses of the design in order to assure consistency with the new values. Because the revised values coupled with power level uncertainty treated as a direct factor resulted in negligible changes to the predicted thermal-hydraulic performance, the Project chose not to revise the original hot channel factors at this time.

We have reviewed the hot channel factors and their method of application in the safety analyses, and find them acceptable.

4.4.8 Hydraulic Stability

The FFTF Project has stated that no hydraulic instabilities are expected in the FFTF core since two-phase flow is not permitted during any modes of operation, and therefore did not propose any criteria for hydraulic stability.

Based on staff experience with light water reactor hydraulic behavior, we agree that under conditions of prescribed power operation with single phase flow, hydraulic instabilities would not be predicted to occur in the core. The Project has performed steady-state flow distribution analyses, which are shown in Figure 4.4-1. They do not predict any changes in the percent of vessel flow entering the core until vessel flow rates are decreased below about 40 percent. Since technical specifications preclude operation with flow below 60 percent of full flow, no instabilities are expected during power operation.

TABLE 4.4-3

HOT CHANNEL FACTORS AT INITIAL OPERATING CONDITIONS

		Coolant 1	Film 2	Cladding 3	Gap 4	Fuel 5
DIRECT						
1.	Inlet Flow Maldistribution	1.050	1.012			
2.	Intrasubassembly Flow Maldistribution	1.140	1.035			
3.	Interchannel Coolant Mixing	1.000[a]				
4.	Power Control Band	1.020	1.020	1.020	1.020	1.020
5.	Wire Wrap Peaking[b] Direct Combination[b]	1.221	2.000(1.214) 2.137(1.297)	1.020	1.020	1.020
STATISTICAL						
1.	Fissile Fuel Maldistribution	1.035	1.035	1.035	1.035	1.035
2.	Power Level Measurements[c]	1.071	1.071	1.071	1.071	1.071
3.	Nuclear Power Distribution	1.060	1.065	1.065	1.065	1.065
4.	Rod Diameter, Pitch, and Bow	1.011				
5.	Film Coefficient		1.140			
6.	Gap Coefficient				1.470	
7.	Fuel Conductivity					1.100
8.	Cladding Conductivity and Thickness			1.120		
	Statistical Combination of Items 1-8	1.100	1.173	1.158	1.481	1.143
	Products of Direct and Statistical Combination[b]	1.343	2.507(1.521)	1.181	1.511	1.166

[a] Worst condition.

[b] Numbers in parenthesis should be used only for calculating fuel temperatures, other for fuel cladding and coolant temperatures.

[c] Value is 1.087 at rated operation.

Table 4.6-5 of FFTF FSAR.

The Project has argued that any perturbations at low flow which tend to decrease flow to the hot assembly will be self-correcting because any decrease in coolant density due to coolant heatup will decrease the gravitational pressure drop in that assembly and promote additional flow to that assembly.

During conditions of low flow, redistribution of the flow becomes significant, particularly during conditions of natural circulation. Because the most severe power-to-flow mismatch occurs for this event, the Project investigated the potential for flow instability for this case, utilizing a multiple channel dynamic representation of the core. These analyses did not result in any predicted flow instabilities. The startup program for the FFTF includes natural circulation testing. We recommend that these tests be used to verify the analysis methods used.

Based on the Project's commitment to verify the dynamic hydraulic behavior of the core flow, we find the hydraulic stability analyses acceptable.

4.4.9 Loose Parts Monitoring System

Section 4.4 of the Standard Review Plan for light water reactors requires the review of the loose parts monitoring system proposed by the applicant. The FFTF presently does not have an installed loose parts monitoring system. The Project has stated the reasons for this are due to the advanced stage of construction and the cost and schedule penalties that would be imposed in order to install such a system. In addition, the Project discussed difficulties with regard to application of state-of-the-art systems to sodium cooled reactor design. These included difficulties due to the high temperature and high neutron fluence environmental conditions, as well as installation difficulties on installed components.

The Project has stated that they are participating in a noise surveillance program at Argonne National Laboratory. We recommend that the continued participation by the FFTF Project in this program be regularly assessed with an objective of evaluating loose parts detection capability and with additional consideration of the installation of an acoustical system for loose parts detection. It is the staff's position that loose parts monitoring systems will be required on future commercial liquid metal fast breeder reactors and that the FFTF should be utilized to the maximum extent possible to demonstrate the feasibility, operability and maintainability of such systems.

4.4.10 Evaluation and Findings

Based on our review, we conclude that the thermal-hydraulic design of the FFTF core is acceptable for operation up to 100 percent of rated power with the condition that the thermal-hydraulic analyses are confirmed at each appropriate step of the power escalation by the following:

(1) Verification of IANUS code predictions for natural circulation and other (e.g., reactor trip) transient behavior.

(2) Verification of the FLODISC code for dynamic redistribution of core flow.

(3) Verification of the VARR-II code for dynamic thermal-hydraulic behavior of coolant in the upper plenum.

To support the thermal-hydraulic design verification, we recommend that reports documenting the pretest predictions made for startup tests be submitted for staff review prior to the startup testing and that results of the thermal-hydraulic test program be submitted upon completion of startup testing.

4.5 Mechanical Design

4.5.1 Control Rod System Structural Materials

The control rod system consists of three major components. They are the control rod drive mechanism, control rod disconnect driveline, and the control rod absorber assembly. The design of the control rod system is an extension of the reactivity controls used in light water reactors with appropriate changes based on current technology associated with the environmental and service requirements of the liquid metal cooled fast reactor.

Eighteen penetrations are provided in the reactor head and instrument tree for the control rod system. Reactivity control is provided by two groups of moveable full-length absorber rods, operated by individual control rod drive mechanisms of the roller nut type design. In addition, the core reactivity is adjusted within a given range by repositioning fixed shim assemblies during the refueling operation. The moveable control rods are divided into primary and secondary systems. The three rods in the primary system are located in the third row of the core, and are withdrawn prior to reactor criticality. The secondary system of six rods located in the fifth row provide operational reactivity. The additional nine core positions located in the seventh row of the core are provided with guide tubes and head penetrations for possible use as moveable control rods at a future time.

4.5.1.1 Control Rod Drive Mechanism and Drive Line

The control rod drive mechanism is an electro-mechanical actuating device that moves and positions the primary and secondary system control rod absorber within the core. It is installed in a reactor head nozzle and is separated from the sodium environment by a bellows, sealed to the support housing and leadscrew, which interfaces with the control rod disconnect driveline. This bellows prevents reactor cover gas from leaking out. The control rod drive mechanism is designed and fabricated to conform to the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1968 Edition, supplemented by appropriate Reactor Development Technology

and Hanford Works Standards for welding materials and processes and materials for service in a liquid sodium environment.

The materials specified for the control rod drive mechanism were selected on the basis of light water reactor experience. Development tests were performed to verify critical wear and friction applications, such as bearings, guide bushings, and the torque restraint device. The materials specified include 17-4 PH, Inconel 718, Types 304, 410 and 440C stainless steel, and Haynes Alloy No. 25. The design life is 20 years with an operational capability of 20,000 feet of leadscrew travel, 1500 full-length scrams, and 30,000 start-stop cycles. Functional performance tests were simulated with hot sodium and conducted to confirm the capability of the design to meet the performance objective. In addition, test specimens representative of the stator winding and insulation material were exposed to irradiation equivalent to 10^6 rads with no deleterious effects.

The control rod disconnect driveline provides a mechanical linkage between the drive mechanism and the absorber assembly. The driveline has two disconnects which permit uncoupling of the drive mechanism and absorber assembly prior to the refueling operation and recoupling the components for resumption of reactor operations. A dashpot is provided to decelerate the system components during reactor scram. The control rod system is supported and positioned relative to the core by the instrument tree guide tubes. The control rod disconnect driveline is installed in the instrument tree guide tube after the instrument tree is installed in the reactor vessel.

The control rod disconnect driveline is primarily an Inconel 718 weldment designed to operate at temperatures between 400° and 1100°F to a fluence of 5×10^{20} n/cm^2 during a life of 10 years. Inconel 718 was selected because of its relatively high strength and resistance to both irradiation embrittlement and corrosion in liquid sodium. Auxiliary devices, such as locking nuts, are fabricated from Type 304 stainless steel.

4.5.1.2 Control Rod Absorber Assembly

The control rod absorber assembly provides neutron absorber material to control and terminate the nuclear reaction in the core. It guides and controls the flow of the sodium coolant and provides shielding to the core support structure. Basically, it is a hexagonal duct component, 12 feet long, containing 61 boron carbide pins in an inner duct compartment.

The control rod absorber assembly interfaces with the core basket within the core support structure, the control rod drive mechanism during reactor operation, and the in-vessel handling machine during the refueling phase.

A special feature of the shaft design is a breakaway joint. This permits separation of the control rod drive and the assembly in the event a pin bundle is stuck

in a partially withdrawn position or in the event the shaft fails to decouple from the assembly.

Cold worked Type 316 stainless steel is specified for the primary structural material in the control rod absorber assembly, including the cladding material for the boron carbide. Inconel 718 is specified for springs. Inconel 718 and chromium carbide are specified for wear pads and load pad surfaces to minimize galling.

The assembly is designed to operate at peak clad temperatures of 1240°F to a maximum fluence of 1.5×10^{23} n/cm^2 which is expected to produce < 1 percent clad strain. Degradation processes which are considered to limit lifetime are thermal and fluence induced bowing of the stainless steel, and cladding strain resulting from gas release and swelling of the boron carbide.

4.5.1.3 Control Rod System Materials Evaluation

The mechanical properties of materials selected for the control rod system components exposed to the reactor coolant satisfy the requirements of Section III of the ASME Boiler and Pressure Vessel Code. Scram dynamics tests and environmental life tests have been successfully performed in liquid sodium with the control rod drive system.

The controls imposed upon the austenitic stainless steel of the systems conform to the recommendations of Regulatory Guide 1.31, "Control of Stainless Steel Welding," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices performed in accordance with these recommendations provide assurance of structural compatibility with the sodium coolant during the design life of the component. The compatibility of the materials used in the control rod system in contact with the reactor coolant satisfies the criteria of Articles NB-2160 and NB-3120 of Section III of the Code. Both martensitic and precipitation-hardening stainless steels have been given tempering or aging treatments in accordance with the requirements of ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During the Construction Phase of Nuclear Power Plants," and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

Conformance with the Codes, Standards, and Regulatory Guides indicated above, and the minimum tempering or aging temperature of martensitic and precipitation-hardening stainless steel constitutes an acceptable basis for meeting in part the requirements of General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

4.5.2 Reactor Internals

The reactor internal components, exclusive of the core, reflector and control assemblies, are the core support structure, radial shield, instrument trees, and

the in-vessel handling machines. Portions of the internal components are designed to be removable for inspection and possible replacement in the event deterioration of integrity may occur as a result of reactor service.

For operation below 800°F, the reactor internal components were constructed in compliance to Section III of the ASME Boiler and Pressure Vessel Code. For operation above 800°F, the components were constructed in compliance to ASME Code Case 1331 and ARD-FRA-152. Extensive development programs were conducted by the Project to verify the component design. Design verification tests were performed to verify the integrity of the components, which include simulated tests to verify thermal-hydraulics, mechanical response and materials integrity of various subcomponents. Further system verification tests are planned as part of the overall acceptance test program during startup of the facility.

4.5.2.1 Core Support Structure

The core support structure locates the core components within the reactor vessel, and consists of the lower structure and support skirt, core barrel, horizontal thermal baffle and in-vessel storage module. The lower structure is welded to the reactor vessel. The core basket is centrally located within the support structure and provides lateral support for the radial shield, the core restraint mechanism, and the instrument trees. The in-vessel storage module is located in the annular space between the core barrel and the thermal liner. The core support structure is designed for a 20-year life.

The core support structure is constructed of Type 304 stainless steel plate and forgings and welded with Type 308 filler metal. The welding processes employed were the shielded metal arc and single-wire automatic submerged arc processes. Major emphasis was placed on process control in order to avoid distortion of the weldment. This involved controlling the rate and sequencing of metal deposition in order to balance shrinkage forces and thermal effects. The delta ferrite content of the welds was maintained between 4 and 9 percent which is well within acceptable limits.

The weld subject to the highest thermal stress joins the top plate to the forging at the top of the inner barrel in the reactor vessel and is estimated to receive a fluence of 4.7×10^{20} n/cm^2 at approximately 800°F. The maximum tensile stress of 5 ksi is estimated to occur in this weld during a normal scram transient. Fracture mechanics concepts were employed to characterize the end-of-life potential of this weld. The calculated stress intensity factor was 3.9 ksi in$^{0.5}$ compared to an assumed end-of-life fracture toughness (K_{Ic}) of 85 ksi in$^{0.5}$ indicating ample margin against brittle fracture.

4.5.2.2 Radial Shield

The radial shield provides neutron shielding to the core support structures, the radial restraint and the reactor vessel. It is constructed of laminated Type 304 stainless steel plates, vertically arranged with associated holddown and locking devices to form six inner and six outer shield blocks. The radial shield is 23.6 inches thick and extends 44 inches below and 70 inches above the core midplane. The inner shield blocks are removable through the in-vessel handling machine port. They receive a fluence of 1×10^{23} n/cm^2 during a 10-year life. The outer shield blocks are not removable and are designed to have a 20-year life.

The shield is cooled by a flow of sodium coolant between the Type 304 plates, which are separated by spacers. Clearance is provided between the inner shield and the outer row of reflectors to accommodate core dimensional changes caused by irradiation and thermal effects, as well as to permit unclamping or loosening of the core in the event of a refueling problem.

Self-welding and galling are prevented at the interface metal surfaces between the radial shield blocks and the core support structure, core barrel and radial reflectors by using wear pads of age-hardening Inconel 718. Contact surfaces of the holddown devices are chromium plated to prevent self-welding to the core support structure locking rings.

The design approach to the radial shield is to provide a mechanical assembly which minimizes thermal and seismic stresses, especially in the core region most affected by the thermal environment. The high fluence areas operate at a metal temperature of about 950°F. The radial blocks are designed to have an end-of-life ductility between 5 and 10 percent. However, the fluence damage estimation in several areas yields minimum end-of-life ductilities of $\leq$ 0.5 percent and reduction of areas of 10 percent. The relative large reduction in area indicates that the material is still ductile and will have the ability to absorb significant local straining or plastic deformation without failure.

4.5.2.3 Instrument Trees

The reactor transverse section is divided into three sectors. Each sector is serviced by an instrument tree. The instrument tree performs the multiple functions of providing support and location for the instrumentation and control rod drive shaft assemblies, and acts as a secondary holddown for the core assemblies.

The maximum thermal transient stress on the instrument trees results from either a control rod drop or a pump seizure followed by a normal scram. The instrument trees are designed to sustain the resulting transient stresses without structural damage. The design life is 10 years at a maximum operating temperature of 1110°F. The fluence limit is specified so that the minimum end-of-life residual ductility is $\geq$ 10 percent elongation.

The principal material of construction is Type 304 stainless steel. Haynes Alloy No. 273 is specified for hardsurfacing to prevent self-welding and galling of contacting surfaces. Inconel 718 is specified for the lower bearing sleeve. The cobalt content of the two alloys is held to 1.0 and 0.05 percent, respectively, to minimize activation of mass transfer products.

4.5.2.4 In-Vessel Handling Machine

The in-vessel handling machine is designed to provide accurate positioning, orientation, and identification of either irradiated or unirradiated core components within a given sector of the reactor. There are three in-vessel handling machines, each controlling the core components in a one-third sector of the reactor, operated from a common control console.

The general philosophy for the selection of materials for the in-reactor components is that Type 304 stainless steel is specified unless its strength is insufficient or that self-welding and galling of the moving parts in the liquid sodium will present a problem. Inconel 718 is specified for its high temperature strength characteristics; AISI 52100 steel and Stellite No. 3 are used for bearing and anti-galling surfaces; Alloy A-286 is specified for bolts and pins; and Type 347 stainless steel is used for bellows construction for atmospheric seals.

The arm, stem and foot assembly of the in-vessel handling machine supports, guides and aligns the component grapple during vertical and rotary motion. This is accomplished while submerged in liquid sodium at approximately 400°F shutdown temperature. During operation of the reactor, the assembly is retracted and stored in sodium at a maximum temperature of 1100°F. The assembly may be removed from the reactor for repair and maintenance or at end of life. The structural materials are estimated to receive a fluence of $\leq 1.4 \times 10^{20}$ n/cm^2 during their design life of 10 years after which they will be replaced.

4.5.2.5 Reactor Internals Materials Evaluation

The materials used for the construction of the reactor internals have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code.

General corrosion and mass transfer from these materials is expected to be negligible. As demonstrated by extensive testing and satisfactory performance in sodium systems, the materials standards and fabrication processes provide reasonable assurance that the reactor internals will be adequately protected from conditions which could lead to loss of structural integrity.

The standards used and the controls imposed during the construction of the austenitic stainless steel reactor internals satisfy the recommendations of applicable

Regulatory Guides and the requirements of Section IX of ASME Boiler and Pressure Vessel Code. The materials selection, fabrication practices, and examination procedures performed in accordance with these recommendations provide reasonable assurance that the reactor internals will be in an acceptable metallurgical condition to preclude service-induced loss of structural integrity within their design life. Conformance to these standards and service recommendations constitutes an acceptable basis for meeting in part the requirements of General Design Criteria 1 and 14.

4.5.2.6 Reflectors

There are four types of radial reflectors located in rows 7, 8 and 9. There are 108 in all. The upper and lower nozzles of the reflectors are very much like fuel assemblies. The reflectors are made of stainless 316 and Inconel 600 with 6 to 37 vertical coolant passages to remove heat. Self-welding and galling are prevented by hard surface contact surfaces.

The design life is set two to four times below the expected design life. The actual design life will be determined from examination of irradiated cups on samples up to 1-2 x 10^{23} nvt. The lower threshhold of 0.5 percent elongation for row seven appears near 2 x 10^{23} nvt. The calculated strain for design life is 0.1 percent. The radial reflectors are designed in compliance with the ASME Boiler and Pressure Vessel Code Section III FRA-152 and Division of Reactor Development Technology Standards. We find the design to be in compliance with General Design Criterion 1 and the intent of Standard Review Plan 4.5.2.

4.6 Functional Design of Reactivity Control System

The FFTF reactivity control system does not include provisions for the addition of a soluble neutron poison to the reactor coolant. The full length control rod assemblies, containing boron carbide, are the only shutdown mechanism in the event of a transient and are inserted automatically from above the core, satisfying the requirements of General Design Criterion 20. No credible single failure will result in loss of the protection system nor will a loss of redundancy occur as a result of removal of a channel or component from service. The foregoing, periodic testing, reliability and redundancy, conform to the requirements of General Design Criterion 21.

The control rods are divided into two groups: the primary rod group consisting of three centrally located control rods in the third hexagonal row, and the secondary rod group consisting of six circumferentially distributed control rods closer to the periphery of the core in the fifth hexagonal row. The control rod drive mechanisms are of the electro-mechanical leadscrew type similar to the drive mechanisms used on many Naval reactors. These drives are based on the principle of engaging the leadscrew with a split roller nut which is rotated to move the rod in and out of the core. This nut is disengaged to permit insertion of the control rod into the

core in the scram mode. Engagement of the the roller nut is achieved by energizing the drive motor field which causes the two rotor poles to move outward and move the split nut into engagement with the leadscrew. Rod movement is accomplished by rotation of the electric stepping motor which rotates the nut and inserts or withdraws a rod. Disengagement of nut and leadscrew is accomplished by deenergization of the motor field which allows the split nut to be disengaged by springs and the rod to drop into the core. Failure of electrical power to a control rod assembly will cause insertion of that assembly. Single failure of a control rod assembly is considered in transient and accident analyses which include the most reactive primary control rod assembly stuck outside the core. Analysis of accidental withdrawal of a control rod assembly is found to have acceptable results. This conforms to the requirements of General Design Criteria 23 and 25.

We reviewed studies assessing the feasibility of providing diversity in the shutdown system during the construction review stage. These studies were of a scoping nature and led to the conclusion that a concentration of effort on development of a single system would result in greater reliability than could be achieved in two or more undeveloped diverse systems. On this basis, the control rod system selected for the FFTF is one which does not possess diversity but does have redundancy. We have considered this matter carefully in our review and have concluded that the lack of diversity is acceptable in this particular experimental reactor facility. It is our conclusion that this design satisfies the intent of General Design Criterion 26.

Criticality is not allowed during normal reactor operation unless all primary control rods are withdrawn from the core. These rods are used to shut down the reactor as necessary. Movement of the six secondary control rods controls normal operating and starting reactivity changes and can also be used to control anticipated operational occurrences. For operating and scram conditions fuel design limits will not be exceeded.

Either of the two reactor control rod groups is capable of controlling all anticipated operational changes, transients, and accidents. All accidents are calculated with the assumption that only the primary rods are scrammed with the most reactive primary control rod stuck, which complies with the requirements of General Design Criterion 27. The accidents are also calculated with the assumption that only the secondary control rods are scrammed.

Additional reactivity control is afforded on a cycle-to-cycle basis through the use of fixed shim rods in the seventh hexagonal row. These rods are fixed in that for a given core cycle they do not move. Their loading pattern may be modified at fuel loadings, with rods added to or subtracted from the reactor as necessary, to account for overall core reactivity changes from fuel depletion and fission product buildup. We find the use of fixed shim rods for overall cycle-to-cycle reactivity modification to be acceptable.

References

1. Fast Flux Test Facility System Design Description No. 31, Rev. 48, Hanford Engineering Development Laboratory, December 31, 1975.

2. Sim R. and A. Veca, "FFTF Fuel Pin Final Design Support Document" (Transmitted by FCF-214) December 10, 1971, and updated (Sec. 3.3, 3.4, and 4.0) by letter FCF-272, ARD- FCF-213, February 28, 1972.

3. Eberlein W. C., R. E. Bears, J. D. Watrous, and R. J. Jackson, "Revised FFTF Cladding Temperature Integrity Limit and Update Evaluation of the FFTF Fuel Pin Transient Design Procedure," HELD-TME 76-32, May 1976.

4. Leggett R. D., et al, "Linear Heat Rating for Incipent Fuel Melting in UO_2 - PuO_2 Fuel," HEDL-SA 464, Trans. American Nuclear Soc. 15, p. 752, October 1972.

5. Leggett R. D., et al, "Influence of Burnup on Heat Rating-to-Melting for UO - PuO Fuel," HEDL-SA-737, Trans. American Nuclear Soc., 19, p. 136, October 1974.

6. Dienst V. and A. Zimmerman, "UO_2 - PuO_2 Fuel Pin Capsule Irradiations of the Mol 8B and C Test Groups: Evaluation of Measuring and Post-Irradiation Examination Results," EURFNR-3187, September 1976.

7. Mueller I., "Studies on the Mechanical Interaction Between Oxide Fuel and Cladding in the Case of Power Changes in Fast Breeder Fuel Rods," EURFNR-1409, November 1976.

8. Ross D. F. (NRC), Memorandum to R. P. Denise, "CRBR SER Input," September 19, 1977.

9. Baars R. E., "Evaluation of FFTF Fuel Pin Transient Design Procedures," HEDL-TME-75-40, September 1975.

10. Jackson R. J., "Evaluation of FFTF Fuel Pin Design Procedure Vis-a-vis Steady State Irradiation Performance in EBR-II," HEDL-TME-75-48 Addendum, November 1975.

11. Travis M. L, "Exposure Dependent Cladding Deformations," WARD-D-0146, July 1976.

12. Nuclear Systems Materials Handbook, TID 26666, Hanford Engineering Development Laboratory, Richland, WA.

13. Bowen P. H., "Adhesive Wear (Galling) of Materials in Liquid Sodium," Report 70-1BS-HAWAR-R1, Westinghouse Research Laboratories, May 1970.

14. Kleefeldt K., "Investigation of Wear and Friction Between Fuel Rod Bundle and Spacer Grids in Sodium," Report KNK-1290, Karlsruhe Nuclear Center, October 1970 (EURFNR-854).

15. Liquid Metal Engineering Center, "Annual Technical Progress Report, January-December 1972," LMEC-73-1, February 28, 1973, pp. 1-135 to 1-144.

16. Liquid Metal Breeder Reactor Program, Thirteenth Quarterly Report, August-October 1975, GEAP-13923-13, November 1975.

17. Wald J. W. and I. S. Levy, "LMFBR Fuel Pin Cladding Transient Performance Capabilities: An Analysis of FCTT Data," BNWL-2041, March 1976.

18. Liquid Metal Breeder Reactor Program, Sixteenth Quarterly Report, May-July 1976, GEAP-13923-16, August 1976.

19. "HEDL Transient Irradiation Testing Program," HEDL TI-76035-3, October 13 and 14, 1976.

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Reactor Vessel System

The reactor vessel is the central component in the reactor vessel system, and distributes the flow of the liquid sodium coolant to remove heat from the core, reflector, shield and other reactor internals. The primary components in the reactor vessel system are the reactor vessel, including the interfacing pipe up to the vertical weld at the horizontal run, the vessel support structure, reactor head assembly and the guard vessel.

The reactor vessel system is designed to accommodate steady-state operating conditions of 600°F to 792°F inlet temperature at 225 psig sodium pressure and 858°F to 1050°F outlet temperature at 15 psig sodium pressure, and a maximum core temperature differential of 400°F. The average temperature rise is 300°F. The Project has stated its desire to operate at a cold leg inlet temperature of 680°F rather than 600°F. Our review is based on the Final Safety Analysis Report which is for 600°F.

The reactor vessel system materials have been selected to provide properties which will assure non-brittle behavior and avoid conditions that might result in rapidly propagating failures in the reactor vessel system. Consideration was given by the Project to ductility and notch-toughness behavior of the materials, stress intensities under static and transient loadings, specifications and material fabrication procedures, and the environmental effects of service temperatures and irradiation damage. The design of the reactor vessel and the primary coolant system permits limited or remote access to conduct a surveillance and inservice inspection program. Corrosion by sodium is minimized by controlling the oxygen concentration in the sodium.

The reactor vessel system is designed for an operating life of 20 years. The design, materials, fabrication, erection, installation, inspection and testing of the components conform to the applicable requirements of Sections II, III and IX of the ASME Boiler and Pressure Vessel Code, 1968 Edition, including Code Cases and Addenda through Summer 1970, FRA-152-2, and ANSI B31.7, as supplemented by Hanford Works Standards and AEC Reactor Development Technology Standards.

5.1.1 Reactor Vessel

The reactor vessel is constructed of Type 304 stainless steel plate welded with Type 308 weld metal. The cylindrical portion of the vessel has an inside diameter of 20 feet 3 inches and a wall thickness of 2-3/8 inches. The barrel is welded at

about the core midplane to a forged ring to which is attached the core support structure. The ring is welded to a torispherical bottom closure serving as the inlet plenum for the reactor coolant. The inlet plenum section has a height of 114 inches, and a wall thickness of 2-3/4 inches. The overall height of the reactor vessel is 43 feet 1-1/2 inches.

A ring forging is welded into the top of the barrel directly below the upper flange. The reactor vessel is supported by thirty 18-3/4-inch wide support arms which are welded to the forging and bolted to the main vessel support structure. The vessel support structure is supported by the concrete ledge in the wall of the reactor cavity. There are 96 Lubrite bearing pads in the interface to enable the main support structure to slide with respect to the fixed structure to account for thermal expansion. The maximum interface temperature is 150°F.

A vessel liner protects the reactor vessel from severe temperature transients and reduces the steady-state operating temperature of the vessel. The thermal liner is attached to the core support structure and extends above the normal sodium level in the reactor vessel. The annulus between the reactor vessel and the thermal liner is 1-1/4 inches. Sodium is diverted to the annulus from the inlet plenum. This bypass flow serves to cool the wall of the reactor vessel.

Sodium coolant enters the reactor vessel from the heat transport system through three 16-inch diameter pipes in the inlet plenum at the botton of the reactor vessel. The coolant is mixed within this region and distributed through the core components and liner cooling annulus. Core flow and bypass flows are mixed in the outlet plenum above the core and are discharged through three 28-inch diameter Type 316 stainless steel pipes to the heat transport system.

In addition to the inlet and outlet pipes, nozzles are provided in the upper portion of the reactor vessel for sodium makeup, sodium overflow and sodium sampling.

5.1.2 Reactor Head Assembly

The reactor head assembly provides the closure for the reactor vessel. It is 25 feet in diameter and 21.81 inches thick. The assembly is fabricated from two ASME Section II SA-508, Class 2 forgings which are butt-welded together. There are 51 major penetrations in the reactor head assembly. The temperature level and distribution in the closure head are controlled by two independent ventilating systems.

A series of metal plates is suspended beneath the head to provide thermal and radiation shielding, and to suppress gas entrainment at the surface of the sodium coolant. The biological shielding is supported by steel cylinders welded to the head at each of the three penetrations for the in-vessel handling machine. The

plates going toward the sodium pool are as follows: two 4-inch thick carbon steel plates; one 4-inch thick Type 410 stainless steel plate; one 5-inch Inconel 600 plate; seven 0.1-inch Type 304 stainless steel plates; and one 0.25-inch Type 304 stainless steel plate. The total thickness of the shielding is 29-1/2 inches.

The head assembly is supported at the edges by an "I" cross section ring which rests on the main vessel support structure. Sealing at the head is made by an omega seal welded to the upper flange of the reactor vessel and to the "I" ring, and by two concentric copper plated Inconel 600 metallic O-rings between the vessel and the head and the other between the head and the support structure.

5.1.3 Reactor Guard Vessel

The reactor vessel is surrounded by a guard vessel which is supported by the floor of the reactor cavity. The purpose of the guard vessel is to assure that the core and reactor vessel outlet nozzles are submerged in sodium in the event of a major coolant leak in the piping connections below the top of the guard vessel. The guard vessel is sized to provide a nominal clearance of 8 inches between the walls of the reactor and guard vessel which would control the sodium lost in the event of a leak.

The guard vessel is designed, analyzed, and constructed to the requirements of Class 1 components of Section III of the ASME Boiler and Pressure Vessel Code. It is constructed of Type 304 stainless steel and welded with Type 308 weld metal. The guard vessel is insulated on the outer surface in order to conserve heat and protect the reactor concrete cavity wall from heat. The thickness of the guard vessel varies from 1 to 2-1/2 inches to maintain structural integrity for loading combinations at the design temperature of 1050°F.

5.1.4 Reactor Vessel System Materials Evaluation

The materials used for construction of the reactor vessel and its appurtenances have been identified by specification. The design, material specifications, fabrication, inspection, and quality assurance requirements for the components conform to recommendations of Sections II, III and IX of the ASME Boiler and Pressure Vessel Code, 1968 Edition, including ASME Code Cases and Addenda through Summer 1970 and supplemented by applicable Hanford Works Standards and Reactor Development Technology Standards for material use from 800°F to 1250°F. Since certification has been made by the Project that the materials and fabrication requirements of the ASME Boiler and Pressure Vessel Code have been complied with, the special processes used are acceptable.

The methods used for nondestructive examination of the reactor vessel and its appurtenances have been identified and have been found equivalent or superior to the techniques described in Section V of the ASME Code. Demonstrations have been

made using these special techniques and have satisfied all requirements of the ASME Code. The special methods of nondestructive examination are deemed acceptable.

Special controls and special welding processes used for welding the reactor vessel and its appurtenances have been identified and found to be qualified in accordance with the requirements of Sections III and IX of the ASME Code. The controls imposed and the examination procedures used provide assurance that welds fabricated by the process will have high integrity and will have a sufficient degree of toughness to furnish adequate safety margin during operation, testing, maintenance, and postulated accident conditions. The controls imposed upon austenitic stainless steel welds are in conformance with Regulatory Guide 1.31, "Control of Stainless Steel Welding."

The results of fracture toughness tests reported by the Project on stainless steel weldments provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of the results of fracture toughness tests performed in accordance with accepted test procedures will provide adequate safety margins during operating, testing, maintenance, and postulated design basis accident conditions. Compliance with these provisions constitutes an acceptable basis for satisfying the requirements of General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

The integrity of the reactor vessel system is assured because the reactor vessel, the closure head and the guard vessel:

(1) have been designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code, pertinent Code Cases as supplemented by applicable Hanford Works Standards and Reactor Development Technology Standards;

(2) have been made from materials of controlled and demonstrated high quality;

(3) have been subjected to extensive preservice inspection and testing to provide assurance that the vessel will not fail because of material or fabrication deficiencies;

(4) will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation and that the vessel will not fail under the conditions of the postulated accidents;

(5) can be subjected to inspection of the exterior vessel surface and nozzles remotely by closed circuit television to verify that the high initial quality of the reactor vessel has not deteriorated significantly under service conditions through magnification of the television image; and

(6) will be subject to a Project surveillance program to monitor the material behavior within the reactor with irradiation.

5.2 Heat Transport System

The cooling capability for the heat transport system is 400 MWt when the reactor is operated with three cooling circuits. Each circuit contains identical and equal primary and secondary sodium cooling loops so arranged to permit natural convection for shutdown decay heat removal. An intermediate heat exchanger separates the radioactive from the nonradioactive sodium; and a dump heat exchanger rejects the sensible heat from the secondary sodium to the atmosphere.

The components in the primary loops are located in separate cells in the reactor containment building so that a fault in one circuit can be isolated from the two remaining circuits. For maintenance purposes the components in the secondary loops are located in equipment cells in the heat transfer services building outside of the reactor containment building.

The heat transport system was designed, analyzed and fabricated in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section II, III and IX, and supplemented by FRA-152, "Supplementary Structural Design Criteria for FFTF Application," Code Case 1331, and appropriate Hanford Works Standards and Reactor Development Technology Standards. The design was verified and the structural adequacy determined under a number of variable loading conditions, both normal and off-normal, including a load histogram for the anticipated events during a 20-year design life of the plant. In addition, an extensive development and test program was conducted to verify design and component performance capability. Prior to startup, the Project is planning extensive construction and elevated temperature acceptance testing to further prove and confirm the structural behavior and integrity of the components and system design.

5.2.1 Primary Sodium Loop

Heated sodium flows from the reactor outlet nozzle through the hot leg isolation valve to the suction of a centrifugal, free surface pump. The pump discharge is circulated to the shell side of the intermediate heat exchanger where the heat is transferred to the secondary loop. The flow continues from the intermediate heat exchanger through a flow meter, a check valve and a cold leg isolation valve. The sodium then enters the reactor vessel and flows through the reactor core, thus completing the flow circuit.

The piping from the reactor vessel to the pump suction is 28 inches outside diameter; the pipe from the pump discharge to the reactor inlet is 16 inches outside diameter.

The hot leg is fabricated from 3/8-inch thick seamless Type 316H stainless steel pipe. The cold leg is fabricated from 3/8-inch thick seamless Type 304H stainless steel pipe. The components are fabricated from Type 304 stainless steel. In addition to Types 304H and 316H stainless steel and the corresponding weld metals, Types 308, 316 and 16-8-2, limited quantities of Inconel 718 and Stellite 6 are specified for use within the heat transfer system

The hot leg of the primary sodium loop is designed to operate at a maximum temperature of 1050°F and pressures ranging from 15 psig at the outlet of the reactor to 225 psig at the inlet of the intermediate heat exchanger. The cold leg is designed to operate at a maximum temperature of 830°F and a pressure of 225 psig at the inlet to the reactor.

The primary sodium loop cover gas operating pressure is 10 inches water gauge, which is maintained by the reactor cover gas purification system using argon cover gas in the reactor vessel, reactor vessel overflow tank, and the pump tank gas volumes. All primary sodium loop gas pressures are equalized and controlled from a single source. The fixed reference sodium level within the reactor vessel is 14 feet 1 inch above the outlet nozzle center line.

5.2.2 Guard Vessels

Guard vessels are provided for the circulation pump and intermediate heat exchanger in the primary sodium loop. The guard vessels are constructed to the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1968 Edition including Summer 1970 Addenda and supplemented by Code Case 1331 and FRA-152. Each guard vessel is fabricated from Type 304H stainless steel and consists of a tank and cover. The guard vessels assure that a minimum safe level of sodium is maintained in the reactor vessel in the event of a major sodium leakage. The guard vessels are supported by conical support skirts welded to the vessels and interconnected with the foundation walls.

5.2.3 Secondary Sodium Loop

The secondary loop circulates nonradioactive sodium and transports heat from the tube side of the intermediate heat exchanger to the air-cooled dump heat exchanger. The inlet and outlet piping adjacent to the intermediate heat exchanger is inside the reactor containment building. The balance of the secondary loop, including the sodium expansion tank, secondary pump and associated equipment, is located in the accessible, unshielded, reinforced cells within the service building. The service building provides weather and missile protection for the components in the east loop.

Heated secondary sodium flows from the intermediate heat exchanger through the containment penetration to the dump heat exchanger in 16-inch outside diameter

pipe. In the vicinity of the dump heat exchanger, the 16-inch pipe branches and reduces down to an 8-inch outside diameter pipe, which connects to a hot leg isolation valve. The pipe then connects to an inlet header to one of four dump heat exchanger modules serving the secondary loop. The outlet from the dump heat exchanger module connects to an 8-inch cold leg isolation valve. The pipe then expands to a 16-inch outside diameter. In the vicinity of the secondary circulation pump, the 16-inch pipe expands to a 28-inch outside diameter and connects to the suction of the secondary pump. The pump discharges to 16-inch outside diameter pipe, through a penetration in the reactor containment building, and connects to the intermediate heat exchanger to complete the sodium circuit. Sodium flow is measured by a permanent magnet flowmeter located downstream from the pump discharge.

A sodium expansion tank is provided in each secondary loop to accommodate thermal expansion and pressurize the system. An argon cover gas operating pressure of 175 psig maximum is maintained by control of the argon supply. The sodium pressure in the intermediate heat exchanger is maintained approximately 20 psig above the primary sodium loop pressure in order to avoid radioactive contamination of the secondary loop in the event of tube leakage in the intermediate heat exchanger.

The materials specified for the construction and fabrication of the primary and secondary sodium loops are equivalent. The 16-inch outside diameter hot leg pipe and the 28-inch outside diameter section of the pipe at the inlet of the circulation pump in the secondary loop are 3/8 inch thick, seamless Type 316H stainless steel. The pipe and components in the balance of the secondary loop are seamless Type 304H stainless steel. The hot leg of the loop is designed to operate at a maximum temperature of 1000°F, and the cold leg at 830°F. Both hot and cold legs are designed to operate at a maximum pressure of 250 psig.

5.2.4 Insulation

Tabular electric resistance heating elements are provided for the primary and secondary loop piping and valves, the primary sodium circulation pump and guard vessel, the secondary sodium circulation pump, the intermediate heat exchanger and guard vessel, and the headers to the secondary dump heat exchangers. These heaters provide the capability to preheat and maintain temperature of the piping and components during periods when nuclear heat is not available, such as during initial startup, or when nuclear heat is not sufficient to maintain desired system temperatures.

The electrical heaters are supported around the piping and components by prefabricated stainless steel heater support assemblies. The heater supports are attached to the surface of the piping or component with stainless steel banding straps, allowing the supports to expand or contract with system temperature changes. The heater supports are covered with 0.025-inch thick Type 302 or 304 stainless steel sheet, forming an annular oven between the piping or component surface and the insulation. Leak detector and thermocouple fittings are supported from this

sheet. Insulation is applied over the stainless steel sheet, fastened in place, and covered with an outer jacket of 0.010-inch thick stainless steel sheet. The outer jacket is banded in place with stainless steel banding straps.

Various types of insulation materials are used in fulfulling the design requirements and essential features of the insulation. The insulation materials were purchased to standards for high temperature sodium sytems: RDT Standard M-12-6, "Mineral Fiber Thermal Insulation, Rigid, Flexible and Loose Fill; "RDT Standard M-12-2, "Calcium Silicate Block and Pipe Thermal Insulation," and RDT Standard M-12-5, "Thermal Insulation, Flexible or Molded, High Temperature, Low Conductivity." The insulation on the heat transport system was tested for sodium compatibility and for stress corrosion cracking in accordance with RDT Standard M-12-1, "Test Requirements for Thermal Insulating Materials for Use in Austenitic Stainless Steel."

5.2.5 Verification and Acceptance Test Programs

Design verification tests were conducted to confirm design features of the heat transport system not directly amenable to analyses. Sodium tests of full scale prototype or plant units of the sodium pumps, dump heat exchanger module and isolation valves and the primary check valves have been conducted. The purpose of the testing was to evaluate the design and the structural integrity of the system. Two areas in the heat transport piping have been the subject of extensive tests: the 16-inch tees in the secondary loop downstream from the dump heat exchanger where significant cyclic thermal stress may occur and the thermowell penetration where sodium flow may induce cyclic stresses. Design life expectancies were verified.

Acceptance testing of the heat transport system will validate the design, construction, performance, and operating procedures. The integrity of the system will be ensured by a series of leak tests prior to sodium fill. Functional testing of the components and the systems will be performed over the temperature range of 400° to 800°F after sodium fill. An important test will include testing to confirm decay heat removal by natural convection with fission power during startup.

5.2.6 Leak Detection

The Project has stated that diverse methods will be employed to detect sodium leakage from the reactor coolant pressure boundary. The methods described are contact detectors (electrical conductivity of leaked sodium), aerosol detectors (sodium ionization and differential pressure across filters), chemical analyses (cavity and cell monitoring of the atmosphere) and visual inspection in guard spaces (closed circuit television camera and periscopic examination).

The Project and the staff place little reliance on the use of contact detectors located in the annulus between the reactor vessel and guard vessel. Sodium ionization and differential pressure devices were recommended by the staff and probes

have been incorporated in the design near the three reactor inlets. In addition, chemical analysis of the atmosphere in the reactor cavity is considered by the Project as the third system for sodium leakage detection. During reactor shutdown, the closed circuit television camera may provide a method to locate sodium leakage from the reactor vessel.

The reactor cavity, including the elevated heat transport piping and isolation valves, will be monitored by chemical analysis of the ambient atmosphere. The atmosphere will be analyzed for concentrations of oxygen, moisture, hydrogen and Na24 gamma activity. Contact detectors will be installed at specified low points within the pipe insulation.

The heat transport cells, which contain the primary pump and intermediate heat exchanger, are monitored by chemical analysis of the ambient atmosphere, contact detectors installed at specified low points within the piping and at the bottom of the primary pumps and intermediate heat exchanger, and remote visual examination of the cells and guard vessels.

Local alarms are provided in the containment and service buildings for each leak detector. Common alarms are provided in the control room with a retransmission feature to annunciate signals from a particular location. After confirmation of the initial signal, the operator will initiate a specific program to shut down the reactor, to isolate the leak, and to drain the sodium from the affected loop in order to stabilize the plant under known safe conditions.

An estimate of the sensitivity of the leak detection systems in the FFTF primary heat transport system was provided in Appendix B of the Final Safety Analysis Report. The response and sensitivity of the leak detection systems remain to be demonstrated in a prototypical leak detection system prior to startup.

The contact detector was estimated by the Project to detect a sodium leak of 160 gallons per hour. This leak is approximately equivalent to a throughwall crack of 0.7 inch at 800°F and 140 psig pressure. The sodium ionization detector was estimated to detect 3.3×10^{-12} g/cc sodium in the ambient temperature evaporated from a 1-square-inch sodium pool at 800°F. The differential pressure detector was designed to serve as a backup to the sodium ionization detector and was estimated to have a sensitivity of 50 mg of sodium. The analysis of the ambient atmosphere was limited to gamma radiation and hydrogen detection. The radiation detector was estimated to have a sensitivity amounting to the amount of sodium evaporating from a 34-square-inch sodium pool. The hydrogen detector is sensitive to a 10 percent inventory change which is estimated to occur in 68 minutes of reaction time between moisture in the ambient atmosphere and a 1-square-inch surface of sodium at 800°F.

Visual examination in guard vessel spaces with either periscope or television camera can be conducted only during reactor shutdown.

5.2.7 Heat Transport System Materials Evaluation

Information was presented by the Project for the concept of "leak before break" in the FFTF heat transport system. It is our judgment that a double-ended instantaneous pipe break in the cold leg is not expected to occur and is not considered the initiating event for a core disruptive accident. Our judgment is based on the inherently high fracture toughness and ductility of Types 304H and 316H stainless steel, including the weld metals, and the expectation that these properties will be retained throughout the service life of the facility. Tests will be conducted to verify sustaining these properties by adequate materials surveillance programs. Our judgment is also based on the assumption that preservice and inservice inspection methods would be effective in identifying and locating small flaws and correcting them before they experience significant growth to cause sodium leakage in the heat transport system.

It is our judgment that the information presented by the Project has not adequately addressed the areas of validation and redundancy of the leak detection systems, and that the materials surveillance and inservice inspection program should be improved to ensure continued heat transport system integrity. The Project has not addressed to our satisfaction the long-term material integrity verification procedures or the prototypical reliability and maintainability of the sodium leak detection systems. These areas remain open at this time.

General Design Criterion 30, "Quality of Reactor Coolant Boundary," requires that means be provided for detecting and, to the extent practical, identifying the location of the source of sodium coolant leakage to the atmosphere. It is essential that leakage detection systems have the capability to detect sodium leakage as soon after occurrence as practical to minimize the potential for gross boundary failure. Cracks that might develop and penetrate the boundary are expected to grow very slowly and to afford ample time for a safe and orderly plant shutdown after a leak is detected. Provisions for testing and calibrating the leakage detection systems during plant operation should be provided.

Since nuclear power plants may be operating at the time an earthquake occurs, the leakage detection systems should be designed to continue functioning after seismic events. We consider preservation of reactor coolant a safety-related function and recommend that the leak detection system be seismically qualified and on IE power.

The information describing the leak detection systems, the sensors, and the operating principles of each system and the installed system tests was reviewed. The review included the information provided by the Project describing the experimental test program at the Liquid Metal Engineering Center for the sensors and leak detection systems. We do not consider that the contact detectors have been shown to have the reproducibility of detection required in a leak detection system. In 86 percent of the tests conducted at the Liquid Metal Engineering Center, although the detector was placed at times directly below the leak, the contact detector failed to respond.

The test data evaluation is that the sodium ionization detector was shown to be capable of detecting small leaks if the aerosols generated by the leak reach the "sniffer" tube. There is no experimental evidence that the sodium ionization detector placed at the outlet pipes of the reactor vessel will detect aerosols generated from a leak in the reactor vessel or pipes some distance from the "sniffer" tube. In several of the tests at the Liquid Metal Engineering Center where there was cell monitoring of leaks in tightly insulated pipes, the leaks were detected by the cell sodium ionization detector. We suggest that consideration be given to venting the annulus between the pipe and insulation when monitoring by sodium ionization detector is utilized.

We recommend that (1) a preservice ultrasonic baseline inspection be performed on highly stressed welds in the primary heat transport system, (2) that a sensitive leak detection system be provided and pre-startup tests be performed to demonstrate the sensitivity and reliability of the sodium leak detection systems, and (3) systems materials integrity verification be provided periodically by the examination of short service life components which may be removed from the reactor or by using adequate materials surveillance specimens.

5.3 Overpressure Protection

To meet the requirements of Sub-Article 7300 of Code Case 1596-1 of Section III of the ASME Code (1971), the Project provided the report "Overpressure Protection Report for FFTF ASME Section III Class 1 Systems," HEDL-TC-793. This report defines the Class 1 systems of the FFTF and the overpressure protection design features provided for the components in these systems as required by Code Case 1596-1. Each Class 1 system is discussed and all of the components included in each system have been analyzed to determine the transient and operating events that could give rise to above normal pressures. The design measures taken to contain or relieve these pressures have been described.

The reactor enclosure system, the heat transport system, the closed loop systems, the auxiliary liquid metal system, the inert gas receiving and processing system, and the impurity monitoring and analysis system comprise the Class I fluid systems of the FFTF plant. The purpose of the overpressure protection devices provided in the design of these systems is to prevent damage to components of liquid metal systems and gas systems. Where required, provisions have been made either to relieve gas pressure by devices or to apply instrumented controls or interlocks to ensure that the pressure of a system does not exceed the design pressure for any component. Two types of relief devices have been provided in the FFTF design for overpressure protection. One type relieves pressure directly through gas pressure relief valves and rupture disks. The other type uses indirect means to control the system pressure and limit the maximum pressure of the liquid sodium coolant to an acceptable level after an initiating event. This second type consists of the following automatic control systems: valve closure time limiters, electromagentic pump voltage limiters, heater interlocks, and liquid metal proximity detectors.

As specified by Code Case 1596-1, the following categories of potential pressure-producing events were considered in the design of overpressure protection for the FFTF: heating of an isolated fluid, abnormal pump operation, argon system failure, leaks, pressure surges, and operator error. As an illustration of the application of this Code Case in HEDL-TC-793, consider that portion of the primary coolant boundary in each of the three heat transport system loops from the primary pump discharge nozzle downstream to the core support plate ring in the lower portion of the reactor vessel. This portion of the primary coolant boundary is designed to a pressure of 225 psig. Included in this region is the highly-stressed 16-inch crossover piping of the hot leg. The maximum anticipated pressure to which this component could be subjected is 218 psig caused by an increase in pump speed due to a control system malfunction. This maximum anticipated pressure is below the design pressure for the components.

The results of the analyses of all of the components in the FFTF Class I fluid systems indicate that the maximum anticipated pressures would either be less than the design pressures or else would not exceed the allowable Code value of 110 percent of the design pressure.

Based on our review of HEDL-TC-793, we conclude that the FFTF overpressure protection design is acceptable.

5.4 Surveillance and Inservice Inspection

The objective of the surveillance and inservice program is to ensure the integrity of the components in the reactor coolant pressure boundary whose failure would either result in a release of radioactivity to the environment or adversely affect the ability to cool the reactor. The surveillance and inservice inspection program proposed by the Project in the Final Safety Analysis Report and HEDL-TME 72-106, Revision 3, "Fast Flux Test Facility Coupon Surveillance Program" emphasizes sodium leak detection and visual examination methods for continuous monitoring in order to identify and locate deterioration of integrity of components in the heat transport system.

In addition, a program was described to develop an ultrasonic capability for the volumetric inspection of selected welds in the primary heat transport system. The Project at that time proposed to include the volumetric examination of six highly stressed welds in the primary heat transport system (two welds in each primary loop) in the event that the ultrasonic development program was successful.

The surveillance and inservice inspection program proposed at that time was divided into two phases: (1) systems design verification, and (2) plant operation and maintenance.

5.4.1 Design Verification Test Program

Preservice surveillance and inservice inspection requirements are scheduled to be performed during design verification and acceptance testing. Pressure and leak testing of the assembled reactor and reactor coolant pressure boundary will be performed at ambient temperature to demonstrate the structural integrity of the welds and seals in the various components. The system pressure tests meet the requirements of Section III of the ASME Boiler and Pressure Vessel Code and will be conducted at 1.20 times the designed pressure (18 psig on the primary and 300 psig on the secondary systems). The field welds will be tested using bubble formation and helium "sniffer" methods described in Article 10, Section V of the ASME Boiler and Pressure Vessel Code. The leak rate should not exceed 1×10^{-5} atm cc/sec. Dimensional checks of components and piping are planned at ambient temperature to provide the data base for thermal expansion measurements.

After pressure testing, the surveillance and inservice inspection program states that the reactor vessel and the primary and secondary heat transport systems will be evacuated to predetermined levels. The rate of pressure increase will be monitored at each level. The reactor system will be then inerted with argon, and heated from ambient to refueling temperature. During heatup, component measurements will be obtained at the refueling temperature to determine that hanger and snubber movement is correct and that proper clearances are maintained.

The reactor vessel and the primary and secondary heat transport systems will then be filled with sodium. Dimensional checks of components and piping at 400°F, at 600°F and at the maximum isothermal temperature will be made for comparison with baseline dimensions to ensure that pipe and component movements due to thermal expansion and dead weight are as predicted. The check will provide a positive verification of the thermal design analysis. Each primary loop will be isolated at 400°F and the pump tank cover gas space pressurized to 100 psig to demonstrate the tightness of the primary isolation valves and the isolatable part of the loop.

The pumps will be operated at design flow at 800°F with the primary cover gas pressure at 10 inches (water gauge) and the secondary cover gas pressure at 75 psig. Vibrations will be monitored by accelerometers. During the tests, sodium leakage will be detected by suitable detectors or by visual observation. After the high temperature tests, the systems will be cooled to 400°F. Pipe insulation will be removed from selected pipe welds in the more highly stressed regions and the welds visually inspected for defects and examined for sodium leakage.

5.4.2 Operation and Maintenance Test Program

5.4.2.1 Visual Inspection

Closed circuit television inspection of selected welds within the reactor vessel-guard vessel annulus has been proposed for at least once during each six-year period.

This inspection was to include all reactor vessel welds below the top of the guard vessel, the reactor vessel inlet and outlet nozzle welds, guard vessel downcomer nozzle welds, and the reactor vessel support arm to forging welds. In addition, dimensional clearance measurements between the reactor vessel and guard vessel were to be made when the welds were examined. We understand that the Project now proposes to use this device only to locate leaks.

Periscopic inspection of selected primary heat transport piping and component welds within the guard vessels will be conducted at least once during each six-year period. The method will also be used to examine the position and condition of selected pipe hangers and snubbers in the primary heat transport system.

Direct visual examination of selected secondary heat transport pipe welds will be conducted at least once during each six-year period. The tubes in the dump heat exchanger will be viewed through sight ports annually.

5.4.2.2 Leak Detection

Major reliance was assigned to leak detection methods by the Project as an integral part of the FFTF surveillance and inservice inspection program proposed to ensure the integrity of the components in the reactor coolant pressure boundary. Leak detection methods were combined with visual examination and materials surveillance to detect the initiation of component degradation. Our evaluation of the leak detection systems proposed by the Project was presented in a previous section of this report.

5.4.2.3 Materials Surveillance

The materials surveillance program is described by the Project in HEDL-TME 72-106, Revision 3, "Fast Flux Test Facility Coupon Surveillance Program," December, 1975. Materials surveillance is an integral part of the overall FFTF surveillance and inservice inspection program proposed by the Project. The stated objective is to monitor inservice the condition of the plant components, and to detect abnormal behavior and material degradation in the initial phase prior to component failure.

The program is to assess the continuing serviceability of critical components by determining the effect of the reactor environment on small test coupons which are representative of the component. In summary, the components that are represented consist of the reactor vessel, core support structure, and the reactor internals for which knowledge of the changes in mechanical properties is considered to be critical to the continued safe operation of the reactor.

The Project stated that the program will be conducted in accordance with the intent of ASTM E-185-73, "Surveillance Tests for Nuclear Reactor Vessels." Coupons of the actual materials of construction will be used to monitor the response of the component to the environmental service conditions. The coupons will be suitable for determining

effects of irradiation and environment on mechanical and physical properties. The coupons will be located within the reactor vessel so that the temperature and neutron spectra duplicate as closely as possible that of the component. Inherent in the program is the accurate and detailed monitoring of neutron fluence and spectra, temperature and mass transfer phenomenon.

Since the major mechanical property of interest in the reactor structure is ductility, the principal type of surveillance coupons in the program is either standard or notched tensile specimens of Type 304 stainless steel. Referring to the program, the majority of tests will be conducted on standard tensile specimens. A summary of the type and condition of tests, the discharge schedule, and the number of coupons required are shown in ASTM E-185-73 in Tables III-1, III-2a through III-2k, III-3a through III-3k, and III-4a through III-4k.

The program described by the Project is deficient in tests on the tensile properties of Type 308 weld metal and Types 316 and 16-8-2 stainless steel plate, welds and heat affected zones. In addition, the program is deficient in tests which measure the fracture toughness properties of structural materials in stainless steel components as a function of service conditions; that is, Types 304, 304H, 316, and 316H stainless steels and the corresponding weld materials. The Project stated that methods are currently being investigated for elastic-plastic analyses and prediction of failure in flawed structures. The analyses, including J-integral and equivalent energy, generally employ sub-size linear elastic fracture mechanics compact tension specimens which fail in the wedge opening mode. In addition to the effect of temperature, sodium, and irradiation environment on the plastic notch toughness, information would be obtained from this type specimen to include slow crack growth and plastic instability, notch residual strain field metallurgical stability, fracture resistance curves and fatigue crack propagation. Commitment to use compact tension specimens in the materials surveillance program was not made. The Project stated that surveillance of critical components by use of compact tension specimens would be performed "where appropriate and supplemented at the discretion of the project engineer."

5.4.3 Evaluation of Materials Surveillance and Inservice Inspection

General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," requires that components which are part of the reactor coolant pressure boundary be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity.

The Project has provided access for the inspection of selected critical welds in the heat transport system. The majority of welds in the reactor vessel are accessible for visual examination. It is our judgment that neither sodium leak detection nor visual examination provide an equivalent degree of reliability to volumetric inspection for the detection of growth of a defect in a flawed structure.

The methods proposed by the Project indicate only through wall defects which would initiate reactor shutdown.

The materials surveillance program is weighted with tensile specimens to measure the loss of ductility as a function of environmental conditions of materials in the reactor vessel and internal components. Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10 CFR Part 50, and ASTM E-185-73, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," provide guidance for the monitoring of the fracture toughness properties of the reactor vessel and internal components. Although a limited number of compact tension specimens is included in the materials surveillance program, it is our judgment that greater emphasis should be placed in the program on monitoring the fracture toughness of materials in the reactor coolant pressure boundary.

5.4.4 Recommendations

5.4.4.1 Leak Detection

Since contact detectors have not been validated for reliability by prototypical test data, we recommend the installation of the more sensitive aerosol type sodium leak detectors in certain sections of the reactor coolant pressure boundary. It is recommended that the sodium ionization and differential pressure detector systems be installed (1) over the inlet pipe between the reactor vessel and the guard vessel and (2) near the two highly stressed welds in the crossover pipe in the elevated section of the heat transport system. Furthermore, there should be interconnection for aerosol dispersion between the pipe supports in the insulation in the latter area. The recommendations apply to each of the three primary sodium loops.

5.4.4.2 Material Surveillance

The thrust of the Project's material surveillance program is to detect a decrease in the service tensile ductility of Type 304 stainless steel, the major structural material in the reactor vessel. Although the property used in the FFTF reactor design to limit component life is an estimate of material ductility, it is recommended that the number of compact tension specimens in the program be increased to monitor the change in fracture toughness of materials in the reactor coolant pressure boundary. In addition to Type 304 stainless steel, the program should include compact tension specimens of Types 304H, 316 and 316H stainless steels and their corresponding weld materials for J-integral measurements. Inconel 718 should be included in the program. The change in fracture toughness of these materials should be measured as a function of fluence. The effect of other environmental conditions on fracture toughness, such as thermal and mass transfer effects, which may not be strongly related to fluence, may be measured on out of reactor specimens, documented, but included as justification as part of the revised program. The

program should include tests that will be conducted on short-lived components which are to be removed from the reactor.

5.4.4.3 Preservice Inspection

The Project proposed to include the volumetric examination of six highly stressed welds in the primary heat transport system in the event that the ultrasonic development program was successful. It is recommended that the two welds in each loop be ultransonically tested as part of the operation and maintenance test program. The welds should be tested (1) at ambient temperature before heating the loops and (2) at 400°F both before and after sodium fill. The volumetric preservice inspection is recommended for record purposes and to ensure weld integrity.

5.5 Decay Heat Removal System

The FFTF Project has proposed that, following a reactor shutdown, decay heat be removed from the FFTF core via the same heat transport paths that are used for heat removal during normal reactor operation.

Decay heat will be transferred from the fuel elements to the primary liquid sodium coolant. The sodium coolant is circulated through the three parallel primary heat transport loops where the heat is exchanged through the intermediate heat exchangers (one per loop) to the secondary sodium coolant for ultimate rejection through sodium-to-air dump heat exchanger modules (four per loop) to the atmosphere.

Since the system components used to transport the decay heat from the core to the atmosphere are the identical components used for heat transport during normal reactor operation, the decay heat removal system is not a totally redundant system. Some redundancy does exist, however, in that the three secondary sodium loops, each with its associated dump heat exchanger modules, are distinctly separate from one another. Moreover, within each secondary loop, the four dump heat exchanger modules are themselves redundant. Thus, barring a major primary sodium loss or the blockage of core flow of such a large magnitude that the entire primary system flow is impacted, a sufficient degree of flowpath reuundancy for effective decay heat removal exists within the heat transport system.

Functional diversity also exists since each primary and secondary loop main sodium pump is equipped with a small pony motor which has a power supply that is independent from that of the main pump motor. The Project has performed an analysis which indicates that sufficient decay heat removal capability exists with pony motor flow in only one of the three heat transport system loops (i.e., one primary loop and its associated secondary loop). Furthermore, since it is possible to lose all main and pony motor power supplies under tornado conditions, the Project has proposed that natural convection and natural circulation be relied upon for decay heat removal in the event of a reactor shutdown with a total loss of pump motive power.

We and our consultants have reviewed the proposed method for decay heat removal under forced circulation conditions.

The acceptability of relying on natural circulation is dependent on the satisfactory demonstration of natural circulation heat removal capability during the Natural Circulation Verification Test Program (see Section 14.2 of this report for a detailed description of the test program). Should the FFTF heat transport system fail to show adequate heat removal capability under natural circulation conditions, we recommend that the electric power supplies to the pony motors be enclosed in a tornado missile hardened building and that the necessary circuitry and components be upgraded to satisfy the requirements for a Class IE system.

Based on our review, we would make several recommendations regarding the functional performance of the decay heat removal system if the forced circulation mode is required for decay heat removal. We are concerned about the capability of the primary and secondary sodium pumps to run continuously at pony motor speed without any forced lubrication, or cooling of the pump lubricant. This capability is necessary to assure effective long-term decay heat removal in the forced circulation mode. Although the Project has claimed that the sodium pumps have this capability, we believe that the test data provided by the Project for justification do not fully support this claim. Consequently, we recommend that a test program be conducted under long-term, forced circulation decay heat removal conditions in sodium to confirm that the sodium pumps can indeed run continuously at pony motor speed without forced lubrication.

We have reviewed the ability of the heat transport system to remove decay heat from the core following a reactor shutdown. This review included the heat removal reduction events identified in Section 15.1.3 of the Final Safety Analysis Report as well as plant transients initiated by natural phenomena, such as tornadic and seismic events. The analyses of these events were performed using computer codes which are essentially unverified. Our recommendations for adequate verification of these computer codes are presented in Sections 4.4.6 and 14.2 of this report. Conditioned on adequate code verification, we conclude that the FFTF design provides for sufficient decay heat removal following the heat reduction events identified in Section 15.1.3 of the Final Safety Analysis Report to preclude fuel damage in excess of the specified limits.

Regarding plant transients initiated by natural phenomena, the Project and the staff agree that potentially the most severe heat transport system transient is associated with a tornado event which causes the loss of integrity of the two non-tornado missile hardened secondary loops (loops 2 and 3) and a total loss of all pump motive power. In addition, the Project has submitted a report which included an analysis of FFTF natural circulation behavior under tornado conditions. The analysis, using unverified computer codes, indicated an acceptable core thermal transient for this event.

Also included in the report were the analyses of contrived cases in which the pony motors do not all fail immediately from the tornado effects but rather fail in a staggered manner. These contrived cases resulted in significantly higher primary sodium coolant temperatures than were calculated to occur for the case which assumed immediate failure of the pony motor power supply, indicating inadequate decay heat removal capability. We conclude that it is possible that power to the pony motors may indeed be lost a short time after the rupture of the unhardened secondary loops following a tornado event, giving rise to a more severe core thermal transient than would an immediate power loss. The Project has proposed to bring the plant to a shutdown status whenever meteorological conditions in the site vicinity are conducive to spawning a tornado. We conclude that this is an acceptable resolution of the issue and recommended implementation of this proposal.

6.0 ENGINEERED SAFETY FEATURES

6.1 Summary Description

Engineered safety features are those structures, systems and components included in a given design to mitigate the consequences of design basis accidents. In the case of the FFTF a residual risk is associated with low probability events such as the hypothetical core disruptive accident. The principal engineered safety features in the FFTF are the containment vessel, substructure, containment isolation system, piping integrity-leak detection system and control room habitability system. These features are discussed in the following sections. Cell liners are also included because of their importance to plant safety, even though they are not strictly classified as an engineered safety feature.

6.2 Containment System

The containment system for the FFTF includes the upper containment steel shell structure, concrete substructure including steel cell liners, building heating and ventilating system, containment isolation system and provisions for leak rate testing.

6.2.1 Containment Functional Design

The FFTF containment building is a cylindrical carbon steel pressure vessel (186 feet 8 inches high by 135 feet diameter by 1-3/8 inches thick) with a hemi-ellipsoidal top and bottom head and is designed as a seismic Category I structure and to the requirements of Subsection B of the ASME Boiler and Pressure Vessel Code, Section III. The portion below grade level is sandwiched in concrete, while above grade the exterior steel surface is insulated with a non-flammable material. The net free volume above the operating floor is 1.5×10^6 cubic feet. The design basis internal pressure for the containment building is 10 psig at 250°F, and the maximum allowable leakage rate is 0.1 volume percent per day at the design pressure. Below the operating floor, the steel containment vessel is compartmentalized, consisting of reinforced concrete cells which contain the reactor vessel, the primary heat transport system equipment, and the auxiliary liquid metal system equipment. All cells containing primary sodium equipment are filled with inert gas (N_2 - 1% O_2) to mitigate the extent of sodium combustion with oxygen and are lined with a steel membrane to contain the inert gas as well as protect the concrete in the event of an accidental sodium leak. The reactor cavity which houses the reactor vessel is lined with steel plate attached to steel embedments and is designed for one transient of 35 psig of internal pressure from

a sodium spill. The primary heat transport system cells and other auxiliary liquid metal system cells within the containment are also lined with steel plate and are designed for 12 psig at 150°F and 10 psig at 250°F resulting from a sodium spill.

The FFTF Project contractors have taken the following measures in the design and fabrication of primary sodium systems so there is a low likelihood of sodium release accidents:

(1) Pipes and fittings for sodium service are of seamless construction with full penetration butt welded joints.

(2) All sodium containing materials are of austenitic stainless steel which precludes "brittle failure."

(3) System operating pressures are low in the range of 225 psig or less. Inert atmospheres are provided in areas containing primary sodium.

(4) Leak detectors are provided to give early warning of leaks. Four types of detectors are being used in the FFTF. These are electrical conductivity type, sodium ionization detectors, filter detectors, and cell radiation monitors.

(5) All sodium piping and equipment are covered by metal-clad (exterior jacket) insulation.

(6) Outer guard vessels and drip pans to confine sodium are located in specific areas.

(7) The inert gas processing system is designed to minimize degradation of liquid metal systems from high oxygen and/or water vapor concentrations around the radioactive sodium system.

The accidents which represent the principal challenges to the FFTF containment are sodium fires. These fires result from unexpected failure and subsequent release of sodium from the primary heat transport system equipment. Following sodium release, combustion with the oxygen in the cell atmosphere occurs resulting in increasing cell pressures and temperatures. Depending on the type of break and local system pressures, two types of sodium fires are possible, namely spray and pool.

In order to satisfy General Design Criteria 16 and 50 of Appendix A, 10 CFR Part 50, the containment design basis, including the inner cell system, must be based on considering a spectrum of postulated component and piping failures of different sizes, locations and other properties sufficient to provide assurance that the entire spectrum of postulated sodium fire accidents is covered.

The FFTF Project contractors have performed analyses reported in Chapter 15.2.4 of the Final Safety Analysis Report and described in detail in Ref. 1 regarding the containment pressures and temperatures resulting from consideration of a spectrum of sodium spray and pool fires. Spray fire analyses were performed using the SPRAY-2 computer code,[2] and pool fire analyses were performed using the SOFIRE-II computer code.[3] These codes account for mass and energy release rates, as well as structural heat sinks, to calculate the containment pressure and temperature response for postulated equipment wall breaks resulting in a sodium spray and/or pool fire.

The FFTF Project contractors have determined that 19.8 psig is the highest transient inner cell pressure. This peak pressure pulse results from a 1 ft^2 pipe break at the intermediate heat exchanger inlet causing a spray release of 1050°F sodium into the inerted primary heat transport system cell. The peak cell gas temperature was found to be 928°F. Since the inner containment cells are designed for a long-term pressure load of 10 psig at 250°F, the Project concludes that sufficient structural margin exists to accommodate the calculated higher short-term pressure pulse. The FFTF Project contractors found in Section 15.2.4.1.1 of the Final Safety Analysis Report that the accident which resulted in the highest reactor containment building peak pressure of 1.7 psig corresponded to a spill of 25,000 pounds of sodium into a primary heat transfer system cell during maintenance while the reactor is shut down (sodium temperature is 400°F), and the cell is open to the upper containment air atmosphere through three access hatches whose areas are 50, 60 and 84 ft^2. For these three openings, the Project represented the equivalent single 194 ft^2 opening with an equivalent (root-mean-square value) diameter of 6.41 feet. This representation was found to be incorrect and should be 9.07 feet. The reactor containment building peak shell temperature was found to be 194°F. Based on these results, the Project concludes that the calculated reactor containment building maximum pressures resulting from postulated design basis sodium release accidents are within the containment vessel structural design pressure capability of 10 psig at 250°F.

We have reviewed the SPRAY and SOFIRE computer codes and have determined that the assumptions and analyses methods are reasonably conservative.

We and our consultants have also performed analyses to determine the pressurization of the inner containment cells and the reactor containment building resulting from a wide spectrum of postulated sodium release accidents. The details of our analyses and calculations are presented in Ref.4. We also find that the highest cell pressures occur in the primary heat transport system cell for pipe breaks at the intermediate heat exchanger inlet because the sodium pressures and temperatures are highest at this location. While the Project used one relatively large variable rate of leakage corresponding to a 1 ft^2 pipe break, we used a spectrum of constant leak rates prior to initiation of primary sodium pump trip. Considering an operative sodium pump plant protection trip when the reactor vessel sodium level falls

6 inches in 70.8 seconds results in a peak cell pressure of 16 psig at 100 lbs/sec constant leak rate (leak area is 3.0 in^2). The corresponding primary heat transport system cell atmosphere peak temperature was determined to be 744°F and the corresponding steel liner temperature in the gas space was found to be 134°F. By considering a spectrum of leak rates, it was found that small leaks in the range of 100 to 150 lb/sec give rather high cell pressures because they result in longer spray durations without pump trip which lead to temperature-pressure transients that are more severe than some of the instantaneous larger breaks, which have shorter spray duration. The worst conceivable break, a double-ended break at the intermediate heat exchanger inlet, produced a primary heat transport system cell pressure of 20 psig. The corresponding primary heat transport system cell atmosphere peak temperature was determined to be 891°F and the corresponding steel liner temperature in the gas space was found to be 126°F. For the reactor cavity, the peak cell pressure was determined to be 10 psig resulting from a 1 ft^2 break at the top of the cold leg inlet downcomer piping at the reactor vessel. The corresponding reactor cavity peak atmosphere temperature was found to be 527°F.

During maintenance while the reactor is shut down and the sodium is maintained at a temperature of 400°F, we analyzed a spectrum of large sodium spills in the primary heat transport system cells assuming that there is communication from the cell to the containment building atmosphere through an access hatch. For a total access opening of 194 ft^2 or an opening diameter of 15.7 feet, the peak containment building pressure was determined to be 5 psig. The corresponding containment peak shell temperature was found to be 253°F. These results show higher gas pressures and temperatures than those of the Project because of the use of a larger value for the opening diameter.

With respect to inner containment cells containing sodium and NaK auxiliary equipment, these auxiliary systems are operated at lower temperatures and pressures than the primary heat transport system; therefore, the consequences of leakage are less severe in the auxiliary equipment cells.

Based on the results of our independent parametric analysis, we conclude that the containment design pressure is acceptable for sodium spill accidents occurring during maintenance activities where the sodium is at relatively low temperatures ($\sim$ 400°F). With respect to the inner cell system we find that the primary heat transport system cells would experience internal pressures greater than their design value for a range of pipe breaks as previously discussed. The Project has recently provided the staff with a structural evaluation which concludes that the inner cells can withstand internal pressures as high as 26 psig which is above the short-term peak pressure (20 psig) calculated by the staff. We have not had sufficient time to evaluate the Project's structural analysis. We will review this analysis and will advise the Project if we find that the analysis does not support an affirmative conclusion regarding the capability of the primary heat transport system cell and the containment building.

These conclusions are predicated on the reliance of the inner containment cell steel liners to prevent sodium-concrete interactions. All of the Project's design basis sodium fires analyses submitted in the Final Safety Analysis Report assume that the integrity of the cell steel liner is retained. We find that the extent of inner cell and containment building pressurization depends heavily on the existence of a steel liner which would prevent the released sodium from coming in direct contact with the cell structural concrete. In order to give credit for mitigation of accident consequences, the cell steel liners must be well constructed and periodically have their functional capability verified in a manner mutually agreeable to us. The details of our analyses considering the influence of both sodium fires and sodium-concrete interactions on the inner cell and containment pressurization are presented in Ref.4. For example, without the mitigating effects of the steel liners, our analyses indicate that a maximum primary heat transport system cell pressure of approximately 30 psig could result from a large (1 foot^2 hot-leg break) sodium spray fire when coupled with subsequent sodium-concrete interactions. Since the high pressure could cause inner cell failure, we also analyzed the effect on the outer containment building. Postulating the inner cell failure due to overpressurization allows free communication with the air environment in the containment building. If the cell liner fails, the sodium pool size is important to the subsequent containment response; the larger the sodium spill quantity, the greater the potential for reactor containment building overpressurization. Maximum calculated pressures in the containment were found to be about 12 psig and 30 psig for sodium spills of 26,000 pounds (1 ft^2 break with pump trip) and 100,000 pounds (50 lb/sec leak undetected for about one-half hour), respectively. Therefore, the inner cell and outer containment building long-term design pressures would be exceeded if large sodium-concrete interactions were allowed to take place.

6.2.2 Ex-Containment Secondary Sodium System Fire Consequences

Since the secondary sodium system is part of the normal decay heat removal path and is essentially non-radioactive, radiological consequences from secondary sodium releases are negligible; however, the effects of sodium oxide formation and release on the control room environment must be considered. Section 6.4 considers the effect of secondary sodium fires on control room habitability.

Those portions of the secondary sodium system which are within containment are located in the inerted spaces of the primary heat transport system cells so that secondary pipe breaks inside containment would not result in pressures and temperatures higher than calculated for primary system pipe breaks. The ex-containment portions of the secondary loops are housed in concrete vaults or "cells" containing an air atmosphere. The walls and roof of these concrete vaults are not lined with a steel membrane. However, the floor and lower sidewalls are protected by a steel catch pan with provisions for nitrogen (N_2) flooding to extinguish any fires resulting from a sodium spill accident. In addition, the ex-containment portions

of the secondary loops are provided with smoke detectors which annunciate a fire if one should occur resulting in shutting off the air supply and flooding the cell with N_2 gas. The nitrogen flooding system is part of the plant fire protection system. The secondary loops are physically separated from one another, so that a sodium fire in one secondary loop is isolated from the other loops.

The FFTF Project contractors' analyses of secondary system sodium fires are presented in Ref. 1. The pressures and temperatures resulting from consideration of a spectrum of sodium spray and pool fires were determined for postulated sodium releases in the various cells containing secondary sodium system equipment. It was conservatively assumed in the Project's analyses that there was no N_2 flooding to purge the oxygen. Spray releases were analyzed with the SPRAY code,[2] while pool fire analyses were analyzed with the SOFIRE code.[3] When the pressure buildup in the secondary sodium system cells exceeds 2 inches H_2O, the heating and ventilation system relief dampers open to vent the cell to the outside environment. This feature is characterized by a high leakage model and is incorporated into the analyses as 10,000 percent per day leak rate at 0.25 psig.

The FFTF Project contractors conclude in Section 15.2.4.3 of the Final Safety Analysis Report that the worst conditions from a pressure and temperature standpoint occurred in the secondary heat transport system pump cell. The spray fire resulting from a postulated failure of the 4-inch sodium level equalization line (600°F) as it leaves the expansion tank produced a peak pressure of 11.3 psig and a peak gas temperature of 803°F. The Project concludes that this break location was more limiting than others considered because the elevated location of the pipe break produced the largest spray volume resulting in higher average gas temperature and pressure. The secondary sodium system failure which results in the most severe release of sodium oxide to the environment outside of the containment is a postulated one tube break in a dump heat exchanger with sodium discharge for up to two minutes. Since the dump heat exchanger dampers are not fully closed for 120 seconds following postulated sodium spray release, a significant quantity of sodium oxide is released directly to the environment.

Our assessment regarding the sodium aerosol release, as related to the air intake to the control room, is addressed in Section 6.4.

6.2.3 Containment Isolation

The FFTF containment isolation provisions meet the requirements of General Design Criteria 55, 56 and 57, with the following noted exceptions.

There are 14 pneumatic lines supplying air, nitrogen or argon to services inside the containment, which are provided with containment isolation valves that close upon loss-of-system pressure. All the pneumatic system operating pressures are greater than the containment design pressure and the "switching" pressures for the spring actuated containment isolation valves are also set at pressure levels

greater than containment design pressure. Each of the 14 pneumatic system lines are provided with two containment isolation valves, one inside the containment and one outside the containment as required by General Design Criterion 56, "Primary Containment Isolation." Failure of the piping system inside or outside the containment would result in loss-of-system pressure and isolation of the affected penetration. If system pressure is not lost, through-line leakage is prevented by system pressure within the line. In our judgment this design provides equivalent protection to the containment isolation design required by General Design Criterion 56.

Containment isolation for the interim examination maintenance cell liquid waste line is provided by a quick disconnect valve inside the interim examination maintenance cell and a weld cap on the line outside containment. If it is ever necessary to decontaminate the interim examination maintenance cell, it would not be possible to enter the cell to make the proper connection, thus a quick disconnect with an integral valve capable of being connected by manipulators is used in the cell. Access to the pipe cap outside the containment is not restricted. In our judgment the above isolation arrangement provides protection equivalent to that afforded by the isolation arrangements recommended by General Design Criterion 56.

Containment isolation for 10 secondary sodium system lines is provided by closed systems both inside and outside the containment building. General Design Criterion 57, "Closed System Isolation Valves," requires that containment penetrations employing a closed system inside the containment have at least one containment isolation valve outside the containment building. In the event of a hypothetical core disruptive accident the secondary sodium system is required to allow post-accident heat removal. Therefore, if an isolation valve were installed outside the containment, the valve would of necessity be a remote manual valve. In our judgment, considering the post-accident operations required for the secondary sodium system, the use of a closed system inside and outside the containment provides equivalent protection to the arrangement required by General Design Criterion 57; i.e., a closed system inside the containment and a remote manual valve outside the containment.

During our review of the FFTF containment isolation design, we recommended that the two containment pressure sensing lines be provided with a sealed system with inside and outside pressure transmitting bellows. The Project has elected to use a "spin" orifice plate in place of the inner bellows as a containment isolation barrier. The orifice plate is designed such that air flow through the restrictor at containment design pressure (10 psig) and 250°F is 0.0103 cfm, which is 0.98 percent of the containment design leak rate. However, both the orifice plate and the downstream pressure transmitter are located outside the containment. The Project considers the small line from the containment penetration assembly to and including the orifice plate to be an extension of the containment boundary, and therefore not subject to postulated failures. We do not consider this to provide the equivalent protection of the system we have recommended.

Containment isolation is initiated only by the detection of high radiation levels within the containment ventilation system ducting. During the course of our review we requested the Project to consider the use of diverse signals to provide containment isolation demand. The Project, in response, has indicated that in their judgment isolation of the FFTF containment is required only when there is a release of radiation within the containment building. Considering the spectrum of events which could result in a radiation release (i.e., cover gas leakage, fuel handling accident, core disruptive accident, etc.) and the FFTF design goal to keep radiation releases from all off-normal events to as low as practicable, the Project has concluded that there are no reliable parameters, other than radiation, which will provide early indication of the need for containment isolation. Due to the nature of the potential design basis accidents which could cause pressurization of containment (e.g., sodium spills or leaks) the containment pressure buildup would be very gradual. A 5800-pound sodium spill in an open cell during maintenance of a closed loop system could result in a peak containment pressure of 1.1 psig occurring over 12 hours after the spill, while a 25,000 pound sodium spill in an open heat transfer system cell would result in a peak containment pressure of 1.7 psig after 17 hours. Therefore, the rapid containment pressurization and high rates of containment atmosphere losses prior to isolation valve closure which are of concern for light water reactor systems is not a matter of the same concern for the FFTF containment design. Even for the core disruptive accident or meltdown, radioactivity is a more sensitive parameter than temperature or pressure. We, therefore, conclude that the containment isolation demand for the FFTF need not include diversity of sensed parameters as is required for light water reactor containments.

6.2.4 Containment Leakage Testing

The Project has proposed a containment integrated leak rate test program that will satisfy the requirements of Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors." They also propose to use the reduced pressure test method. We recommend that the chilled water and Mobil-therm systems (penetrations M-1 through 4 and M-206 through 211) be drained and vented to the containment atmosphere during the integrated leak test, unless it can be demonstrated that a fluid seal can be maintained following an accident, considering the effects of the worst single active failure within each system.

The Project has described the local leak rate test program (Type B and Type C testing) to be followed at the FFTF. They have identified containment isolation valves for which the test pressure is to be applied in the opposite direction to the expected accident pressure and have provided justification for reverse pressure testing. We have identified the following list of containment isolation valves for which the Project has not proposed a Type C (local) leak rate test. We recommend that the Project provide the capability necessary to test these valves and include them in the local leak rate test program. The valves and penetration numbers are identified in Table 6.2-1.

TABLE 6.2-1

VALVES TO BE LEAK TESTED

Valve No.	Penetration No.
1HV-74650B/PSV-64128	M-1
1HV-74650A/PSV-64127	M-2
HV-70824A/PSV-70826	M-3
HV-70824B/PSV-70820	M-4
23V86	M-5
23V87	M-6
23V88	M-7
23V89	M-8
HV-74550B/PSV-64123	M-206
HV-74550A/PSV-64124	M-207
HV-74600A/PSV-64125	M-208
HV-74600B/PSV-64126	M-209
HV-74500A/PSV-64121	M-210
HV-74500B/PSV-64122	M-211
HV-07155 and HV-07156	M-46
HV-07016 and HV-07013	M-50
HV-07038 and HV-07041	M-51
HV-07124 and HV-07123	M-53

6.3 Interim Decay Storage Emergency Cooling System

The interim decay storage emergency cooling system is normally cooled by circulating pool sodium to a NaK system which transfers its heat to a Mobil-therm system which, in turn, transfers its heat to the chilled water system and finally to the atmosphere. A backup system is provided by the heating and ventilating system which would circulate nitrogen cooling through an annulus between the insulation jacket and guard vessel. Since not all of either heat transfer train is safety grade, a backup heat removal path is provided. This path closes the valves on the heating and ventilating ducts and opens valves to allow natural convection in the annulus between the guard vessel and the insulation to the containment atmosphere. The containment becomes the heat sink if this method of heat removal is used during containment isolation. We view the operation of these valves to permit this mode of heat removal to be safety-related and have recommended that they be considered an engineered safety feature powered by IE power and that position indication for the valves be displayed in the control room. These recommendations have been incorporated.

6.4 Control Room Habitability

We reviewed the control room ventilation system and control building layout and structures to determine that plant operators will be adequately protected against

the effects of accidental release of toxic and radioactive gases, and that the FFTF can be safely operated or shut down under design basis accident conditions. The evaluation was based on the acceptance criteria described in Section 6.4 of NUREG-75-087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

Relevant portions of the control room ventilation systems are described in this section, while the entire ventilation system is evaluated in Section 9.4.

6.4.1 General Description

The control room is physically separated from the reactor containment building by two adjacent structures - the reactor service building and the auxiliary equipment building (East). The control room is designed to meet seismic Category I requirements and to protect against tornados. If access to the control room is lost during an emergency, control for reactor shutdown will be transferred to primary emergency shutdown controls in the control building basement or to secondary emergency shutdown controls in the basement of auxiliary equipment building (West). The emergency controls are classified as seismic Category I and are tornado protected.

The control room is equipped with two air intakes ~320 feet apart from one another. Each air intake is equipped with a radiation monitor. When high radioactivity is detected, the normal ventilation will be stopped, the control room automatically isolated, and a backup life support system actuated in the recirculation mode. Each of the two air intakes can be manually selected and opened to draw in fresh air through an emergency high efficiency particulate air (HEPA) filter to maintain pressurization of the control room. The Project has concluded that significant quantities of gaseous iodines will not be released from any postulated design basis accident, so that charcoal filters are not required. No special provisions have been made for removal of airborne caustics.

One of the functions of the control room habitability system will be to supply non-radioactive air to the control room after a design basis accident to pressurize the control room to a slight positive pressure. This system will permit operating personnel to remain in the control room following a design basis accident. The control room habitability system is a redundant system, with each system having an intake design capacity of 1000 cfm of air and recirculating design capacity of 2000 cfm of air. Each system contains the following components: prefilter, HEPA filter and a fan. The equipment and components will be designed to Quality Group C and seismic Category I, and will be located in a seismic Category I structure. The system design does not include a charcoal adsorber in accordance with the recommendations of Regulatory Guide 1.52. However, the control room is protected from outside radioactivity sources by means of radiation detectors for automatic control room isolation.

6.4.2 Radiation Protection Provisions

We reviewed the radiation protection provisions for the control room on the basis of General Design Criterion 19, "Control Room."

The control room is shielded with three feet of concrete on the roof and in the walls between the control room and the reactor containment building. In addition, two redundant, independent filter trains, each containing a HEPA filter, are provided to mitigate airborne radioactive particulates in the recirculation mode during emergency.

We determined that the control room provides adequate radiation protection for the operator during normal operation. For postulated accidents involving the release of radioactivity, the control room personnel will be adequately protected provided that two circumstances exist: (1) the majority of the airborne radioactive species is in the form of particulates and (2) mass loading on the HEPA filter will not degrade significantly the design efficiency.

The Project has briefly evaluated the possibility that the more volatile fission products could become airborne and remain in a "gaseous" form. The Project analysis, together with results of very limited test data, indicated that only small fractions of material such as iodine might pass through filters (remain unattached to particles).

Detailed evaluations of the mass loadings of particulates on the HEPA filters that might be associated with various postulated accidents have not been provided by the Project. However, it is estimated that no more than one or two pounds of materials such as sodium oxide can be accommodated.

Our summary conclusion is that the arguments provided by the Project in support of the design are reasonable, but there is little or no margin for error. Given the developmental nature of this facility and the uncertainties associated with the control room habitability analysis, we recommend that consideration should be given to provide a greater margin of protection to the operators. Specifically, consideration should be given to installation of charcoal beds in the standby filter trains and pressurization under design basis events rather than simple isolation.

6.4.3 Toxic Vapor Protection

We reviewed the control room protection against toxic vapors on the basis of Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," revision 1.

Railroad shipments of hazardous materials in the vicinity of the FFTF site are infrequent. Truck transports of toxic chemicals include nitric acid and anhydrous ammonia in 4000 to 7000 gallon tanks, and chlorine gas in one ton containers. Chlorine gas, aqueous ammonia, and hydrazine solutions are stored at the WPPSS reactor site. No toxic chemicals are stored onsite or transported via waterways. Large quantities of sodium metal are used onsite. Circulation of high temperature (non-radioactive) sodium outside of the containment is an integral feature of this reactor concept.

The control room is provided with self-contained breathing apparatus with enough air for 45 minutes. Unlimited replenishment of bottled air is available from other areas within the plant and on the Hanford Reservation.

We determined that none of the toxic chemicals identified above, with the exception of the circulating secondary coolant (sodium) and chlorine gas, would pose significant hazards to control room habitability.

No specific means have been included for alerting the operators that sodium combustion products are entering the air intakes. The spacing between the air intakes is small enough that a fire in a dump heat exchanger or auxiliary building could result in caustics reaching both intakes. In such an event, switchover from one intake to the other would have limited value. For the reasons given in Section 6.4.2, and in view of the presently limited capabilities of the habitability system, we recommend that the design should be reevaluated. Specifically, installation of redundant detectors for sodium aerosol in each of the control room air intakes, automatic switching capability and greater separation are suggested.

References

1. Shire, P. R., J. P. Hale, and G. R. Franz, "Sodium Spill and Fire Evaluation Report for the Fast Flux Test Facility," HEDL-TME-75-122, April 1976.

2. Shire, P. R., "SPRAY-2 Code Description," Argonne Code Center, June 1976.

3. Beiriger, P., et al., "SOFIRE-II User Report," AI-AEC-13055, Atomics International, 1973.

4. Tsai, S. S., R. D. Gasser, and W. T. Pratt, "Sodium Fire Evaluation for the Fast Flux Test Facility," BNL-NUREG-24635, August 1978.

7.0 INSTRUMENTATION AND CONTROLS

7.1 Summary Description

The instrumentation and control system is presented in Sections 7.1 through 7.4 of the Final Safety Analysis Report. This presentation was incomplete and in many cases required reference to supplemental documents such as System Design Descriptions for information needed to clarify and assess the safety significance of the system. Further, the Final Safety Analysis Report did not follow the recommendations contained in the Standard Format and Content document for liquid metal fast breeder reactors dated February 1974. This made assessment of the plant safety systems difficult and time consuming.

The instrumentation and control system for the FFTF is broken down into two major subdivisions: the plant protection system and the reactor plant control system. The plant protection system contains all safety-related instrumentation and control systems including the primary reactor shutdown system, secondary reactor shutdown system, containment isolation systems, heat transfer shutdown system, dump heat exchanger emergency control system and various safety-related portions of the reactor plant control system which act as essential supporting systems.

The reactor plant control system includes all other systems used for controlling and operating the plant under normal conditions and for displaying appropriate plant parameters during both normal operation and post-accident periods.

7.1.1 Evaluation

Our evaluation of these systems will be presented in the normal format. The review of information normally presented in Section 7.1 developed the following concerns:

(1) The tables included in the Final Safety Analysis Report which were to list and identify the various safety-related systems and instruments appeared incomplete and unclear. In response to our request the Project has revised Tables 7.1-3, 7.3-1, and 8.2-10, and has provided additional clarifying material to complete the presentation of the safety-related systems and instruments. We have concluded that this response is satisfactory.

(2) Section 7.1.1.1 of the Final Safety Analysis Report referred to some unidentified critical instrumentation. In response to our request the Final Safety Analysis Report is being revised to include a reference to Table 7.1-3 which

further references Table 7.3-1 for description of the instrumentation. This was acceptable.

(3) Section 7.1.1.4 of the Final Safety Analysis Report did not present specific physical separation criteria for the instrumentation and control systems of the FFTF. The Project's response provided references to Section 8.2 of the Final Safety Analysis Report and to Tables 8.2-1 and 8.2-9 which give the specific physical separation criteria for plant protection system and non-plant protection system circuits and equipment. One instance exists where relay coil-contact isolation is used in a manner which appears not to provide proper isolation (non-plant protection system-coil, plant protection-contact); however, the Project will provide information to justify its use. Another part of the response appeared to suggest that the separation criteria could be diluted in certain cases. The Project indicated that this is not the case and the response will be modified to show that the separation criteria are being supplemented in other cases. We find these commitments acceptable.

(4) The Final Safety Analysis Report exempted certain portions of the plant protection system from meeting the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," with regard to tornado protection. We requested that these portions be identified in the Final Safety Analysis Report and recommended that the design be modified to meet our position that all portions of the plant protection system meet the criteria of General Design Criterion 2 without exception. The response identified all ex-containment isolation valves, their operators, position-indicating circuits and power supplies as not being tornado hardened or protected. The Project has agreed to shut down the FFTF when tornado conditons exist and, therefore, control power is acceptable for position indication.

(5) We had asked for comparisons to be made between the quoted reactor development technology standards and the applicable IEEE Standards. For most of the applicable IEEE Standards this has been complied with; however, in the case of IEEE Standard 323-1974 the Project is to supply an analysis showing how the environmental qualification compares with the requirements of this standard. We will report on this in a supplement to this report.

A basic difference between the FFTF and the light water reactor plant protection systems lies in the passive nature of the engineered safety features and the reactor heat removal system in that only valves are required to be positioned to isolate containment and carry away decay heat. No pumps are required to operate since reactor sensible heat and decay heat can be removed by natural convection and the containment is isolated by valves that close on loss of power or by valves that require only momentary application of power from the Class IE battery supply to

actuate a solenoid to close them. Decay heat removal rate is monitored by self-powered temperature monitors and regulated by manually powered vane-regulating mechanisms in the dump heat exchanger.

Assuming the satisfactory resolution of the open issues stated in item (5), we can conclude that the Project's description and implementation of the criteria and design bases of the safety-related instrumentation and control systems are acceptable.

7.2 Reactor Shutdown Systems

The FFTF is unique in that it possesses two semi-independent, redundant and diverse shutdown systems. These are the primary and secondary reactor shutdown systems. They are independent with the exception of common power supply sources. Each system is redundant within itself including instruments, logic, scram breakers and control rod group. Figure 7.2-1 shows this arrangement in block diagram form. The parameters used in the primary reactor shutdown system are diverse from those used in the secondary reactor shutdown system as is the equipment and logic arrangement.

The primary reactor shutdown system uses two-out-of-three coincidence logic at the logic input from the analog comparators and then uses the output of a one-out-of-thirty-two logic unit to drive the undervoltage coil of its associated primary scram breaker, which in turn opens to drop the three primary rods. Any one of these rods possesses sufficient negative reactivity to take the reactor to subcriticality and shutdown independently of any other shutdown mechanism.

The secondary reactor shutdown system uses a one-out-of-four logic unit to receive input from the analog comparators then feeds into a two-out-of-three logic unit which feeds a one-out-of-eight logic unit which, in turn, drives the undervoltage coil of its associated secondary scram breaker. Opening either one of these two breakers drops the six secondary rods which make the reactor subcritical and shut it down independently of any other shutdown mechanism.

Diversity has been provided in these two scram systems through choice of initiating parameters, type of instrumentation, logic arrangement and manufacturer of scram breaker. Table 7.2-1 shows the diversity of initiating parameters; the analog comparators are of different design; the primary reactor shutdown system uses integrated circuit logic and scram breakers manufactured by ITE while the secondary reactor shutdown system uses discrete component logic and breakers made by Westinghouse.

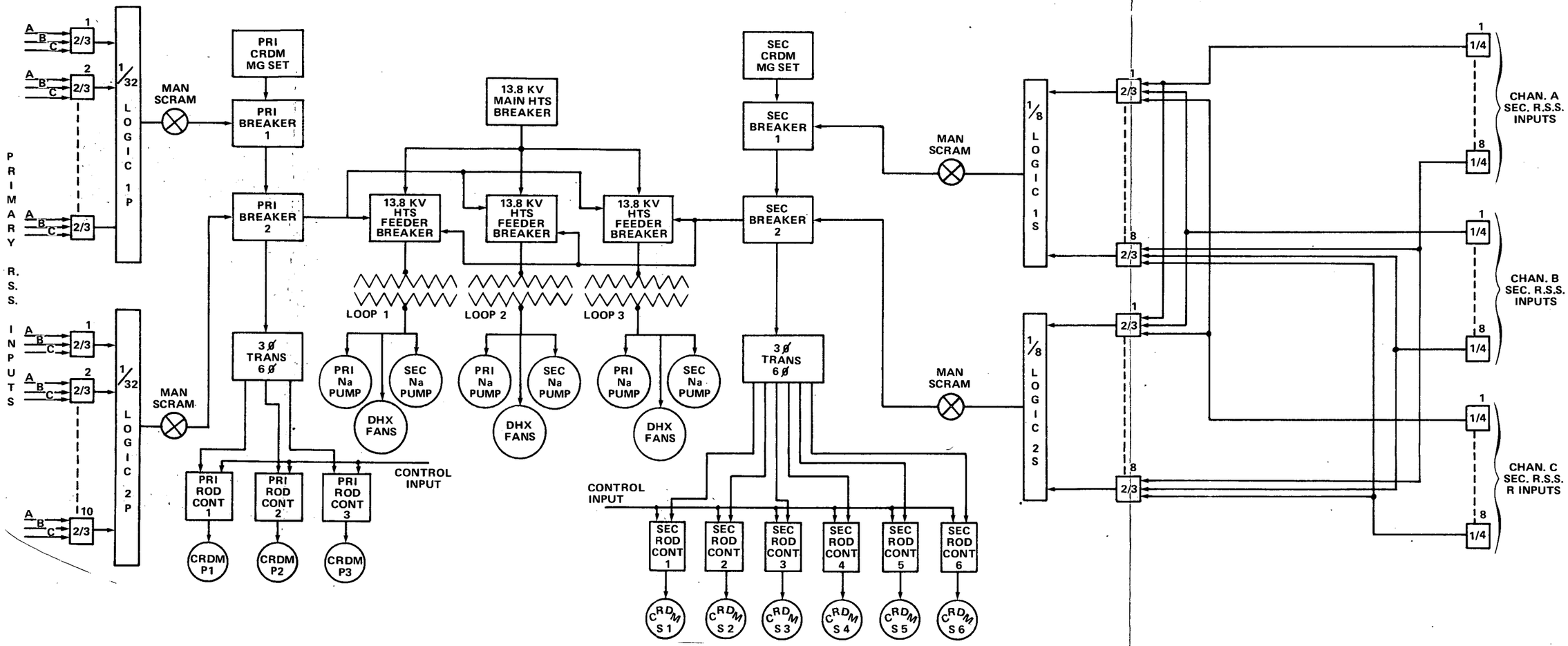


Figure 7.2-1
Primary, Secondary R.S.S. and HTS Shutdown System Block Diagram

TABLE 7.2-1

PLANT PROTECTION SYSTEM FUNCTIONS

	PRIMARY SHUTDOWN SYSTEM		SECONDARY SHUTDOWN SYSTEM
1.	Primary Power Range Nuclear High Flux (1)[b]	1.	Secondary Flux/Total Flow (1)
2.	Primary Power Range Nuclear Low Setting (1)	2.	Secondary Flux-Increasing Delayed Flux (1)
3.	High Startup Flux (1)	3.	Secondary Flux-Decreasing Delayed Flux (1)
4.	Primary Flux-Decreasing Delayed Flux (1)	4.	Low Primary Loop Flow (3)[a]
5.	Primary $Flux^2$/Loop Pressure (3)	5.	High Primary Loop FLow (3)
6.	IHX Primary Outlet Temperature (3)	6.	Low Secondary Flow (3)[a]
7.	Reactor Vessel Coolant Level (1)	7.	Loss of Off-Site Power (1)[a]
8.	Flux/Closed Loop Flow (6)	8.	Reactor Outlet Plenum Temperature (1)
9.	Closed Loop IHX Primary Outlet Temperature (6)	9.	Closed Loop Outlet Temperature (6)
10.	Experiment-Associated Trip Functions (9)	10.	Experiment-Associated Trip Functions (12)[c]

[a] Bypass required to start HTS prime movers.

[b] Numbers in parenthesis refer to actual number of functions used.

[c] Three of these protective functions are reserved for implementation, if necessary during advanced core operation, of a secondary HTS loop overpressure protective function.

Our review of these reactor shutdown systems gave rise to staff concerns as follows:

(1) It was not clear from the Final Safety Analysis Report whether the diversity in the primary and secondary reactor shutdown systems had been carried through the scram breakers to eliminate a potential common mode failure point. The Project's response stated that the breakers were of diverse manufacture and described the breaker units. We have reviewed the additional information and concluded that the response was acceptable.

(2) We requested that additional information be provided that discussed the relationship between nominal operating point, nominal trip setpoint and limiting condition for operation (safety limit) with regard to margin provided to account for errors, instrument inaccuracies, set point drift and uncertainties in the determination of the safety limit. The Project's response indicated the difficulty of relating safety limits that are integral functions of time and magnitude to steady-state levels. The response showed how early and late trip points are derived by combining all uncertainties with the nominal trip point. The safety analysis is based on the latest trip point and must show that plant protection system response based on this trip point maintains the reactor and plant parameters within the appropriate safety limits. Periodic functional testing is to be carried out at monthly intervals to verify that the actual trip point remains within the band established by these early and late trip points. Investigation into causes and recalibration will be done in the event of trip points being found to be out of limits. We have concluded that this response is acceptable.

(3) The Final Safety Analysis Report had indicated that automatic testing of the primary reactor shutdown system logic was accomplished by a system called "Monitor" but no information on this system was presented that would enable the staff to evaluate its ability to adequately test the primary reactor shutdown system logic or to evaluate its effect on independence of the reactor shutdown system. The response also was not clear on how the two types of analog instrument tests (one uses a switch to insert the test signal, the other requires sensor disconnection) were conducted and how the channels were disabled. The Project has clarified this portion of the response. The response should state that there is no way that more than one channel can be disabled without tripping the reactor and that the "Monitor" system tests only the logic channels. The "Monitor" system is not plant protection system but is buffered from the primary reactor shutdown system by photo-isolators and has been analyzed to show that no failure mode can prevent trip. We have concluded that the additional clarification provided by the Project makes this response acceptable.

(4) The secondary reactor shutdown system logic is not periodically tested but is continuously monitored for failures by noting the presence or absence of the AC logic signal. Provision has also been made for complete shutdown testing of the overall system at shutdown intervals of about 90 days. We have found this acceptable.

(5) The manual scram button provided for the FFTF was unusual in that a single actuator device could actuate all four manual scram switches. Two of these manual scram switches are assigned to the primary reactor shutdown system and two to the secondary reactor shutdown system with one switch in each logic train. We were concerned that the single actuating device could be a source of common mode failure and that sufficient physical isolation between redundant switches existed. We examined drawings of the device and the device itself on a site visit and found that each switch was located in an individual steel-walled cell and thus protected against propagation of conditions from one switch cell to another. We also concluded that no credible single failure could prevent a manual scram from being initiated since each of the four switches could be individually activated independently of the common actuator. We found this response acceptable.

(6) We requested that additional information specifying the range of radiation environment to which those plant protection system instruments exposed to high radiation are designed to operate through. The Project has provided additional information regarding the qualification of plant protection system instrumentation that is exposed to high radiation levels as well as instrumentation inside containment that may be exposed to pressures greater than one atmosphere in revised Section 3.10 of the Final Safety Analysis Report. Although we have not completed our review of this revised section, we believe there is sufficient basis to conclude that the equipment is adequately qualified to perform its function. We will report our final conclusions in this matter in a supplement to this report.

(7) We had earlier found the use of resistor buffering between safety and non-safety-related equipment and circuits to be unsatisfactory. However we reviewed additional information which showed this buffering to be used only to isolate certain test points through which portable test equipment is connected to the plant protection system comparators for test and calibration purposes, that strict procedures will limit the connection of such test equipment to no more than one channel at a time and that such channels that are being tested will be placed in a tripped condition. We then concluded that this limited usage was acceptable.

(8) We were concerned about the lifetime and replacement schedule for fission chambers used in the low level flux monitor and in the wide range flux monitor. The Project supplied additional information regarding the maximum

flux, reaction rate and reduction in sensitivity. We have reviewed this information and concluded that the lifetime and replacement schedule is acceptable.

(9) We had requested that Table 7.3-3 be revised to include system accuracies and that these values and those in Table 7.3-4 be compared to those used for the hot channel analysis. The table was revised and the comparison was presented that showed the values used in the hot channel analysis to be conservative. We find this response acceptable.

(10) The primary and secondary loop flows are measured by magnetic flowmeters. Variation in magnetic field can affect the accuracy of these flow meters; therefore we requested information on how the magnetic field will be monitored and necessary corrections made to the indicated flow. The Project's response stated the magnetic field flux will be measured continuously with a gaussmeter, correction will be a direct proportional change in flow signal with magnetic field flux change and an in-situ calibration will be performed after installation. All gaussmeter and flowmeter components are designed to withstand the radiation environment for the proposed 20-year life. Magnet specifications were shown to be adequate for the environment. We have concluded that this response is acceptable.

(11) In response to our request the Project provided additional information on the linearity and accuracy of the inductive probes used for sodium level determination. Calibration will be checked periodically against the known reactor vessel level when sodium is at the overflow level. We have found this acceptable.

(12) The Project had not made provisions in the instrumentation systems for periodic testing of the time response of its safety-related sensors after they have been installed to verify their continued ability to respond within the time range assumed in the safety analysis. However, normal response tests are considered to be impractical due to welded construction. We recommend that the various sensors be characterized during preoperational testing such that similar parameters can be compared during transient events such as scram to demonstrate that no degradation of the response times has occurred.

We have concluded that the Project's description, design and implementation of the reactor shutdown systems meet the intent of staff requirements and are acceptable.

7.3 Engineered Safety Feature Systems

This plant is unusual in that it depends largely on passive measures for decay heat removal, reduction of offsite radiological consequences and other safety-related

functions. As a result, compared to a light water reactor there are very few active engineered safety features. These active engineered safety features are: (1) the containment isolation system, (2) the decay heat removal system and (3) the interim decay storage tank natural convection cooling system.

The removal of decay heat from the reactor can be done through natural convection of the sodium coolant to transfer the heat to the dump heat exchanger. Substantial analysis has been done regarding this mode of heat transfer and extensive verification testing is to be done to confirm its viability.

Containment isolation is stated to be necessary only to confine potential releases of radioactivity from the reactor and thereby reduce the offsite radiological consequences of accidents or other design events. Its accomplishment is relatively passive requiring only a minimum of electric or other motive power to position valves.

The interim decay storage convective cooling also requires only the opening of two out of four solenoid operated valves to establish the natural circulation path into the containment building.

Our review of the information normally presented in Section 7.3 has developed these concerns:

(1) The description of the engineered safety features interim decay storage cell tank cooling system presented in the Final Safety Analysis Report was not clear with regard to the engineered safety features cooling system operation and the instrumentation provided to monitor and control its operation. The Project modified and supplemented the discussion and presented a block diagram of the instrumentation. We have reviewed this additional information and have concluded that it is acceptable.

(2) The description of the heat transfer system stated that "control" circuitry and non-Class 1E power were used in the heat transfer system prime mover trip system. The staff was concerned about the possibility of compromising the independence of the plant protection system and of unacceptable interaction between control and protection systems. In response to our request the Project supplied additional description and diagrams that showed there was no compromise of independence, no interaction between control and protection systems and that the plant protection system portions of the systems remained redundant and capable of meeting the single failure criterion. We find this acceptable.

(3) We had requested additional information on the containment isolation system regarding description of sensors, sensor location, sensitivity, trigger levels, description of signal handling equipment and identification and description of

final actuated devices. The response identified initial parameters, stated that trigger levels are given in the technical specifications and gave references to sections of System Design Description 99 to provide additional information on the containment isolation system. The response will be modified to add references to Section 2-1 and Table 2-IXA of System Design Description 99 and Table 6.5-3 for additional description of trigger levels and final actuated devices. We find this acceptable.

(4) Our concern about the relationship of trip setpoints to limiting conditions for operation (safety limits) and the incorporation of errors, inaccuracies and uncertainties into the determination of setpoints extended to engineered safety features initiation circuits also. The discussion and resolution presented in Section 7.2.5(2) of this report similarly applies to this concern.

(5) The discussion and resolution presented in Section 7.2.5(8) of this report for the reactor shutdown system applies equally to any similar uses of resistor buffering in the engineered safety features initiation system.

(6) We were concerned that the independence of the redundant interim decay storage temperature monitoring system could be compromised by the use of a common thimble in the interim decay storage vessel. It also appeared to be a potential site for a common mode failure. Additional information provided by the Project indicated that the temperature monitoring system is used only for monitoring, performs no initiating function and has no effect on the operation of the emergency engineered safety features interim decay storage cooling system. Emergency interim decay storage cooling can be initiated independently of the temperature monitor. These monitors are tested monthly and up- and down-scale alarms are provided. We therefore conclude that this is acceptable.

(7a) The operation of the heating and ventilating instrumentation system appeared to be necessary to assure closure of the heating and ventilating containment inlet and exhaust isolation valves. We requested additional information regarding the assignment of power sources in such a manner as to not compromise the independence of redundant instrumentation systems. The response stated that the operation of these valves was completely controlled by the containment isolation system and there was no connection to the heating and ventilating instrumentation system that could affect the opening or closing of these valves. We therefore conclude that this is acceptable.

(7b) We were also concerned that the heating and ventilating system instrumentation and control equipment (including the radiation monitor and the chlorine and sodium oxide detectors) that functions to isolate the control area may not be qualified to remain operable through a design basis seismic event or tornado.

We believe control room habitability is a safety function and recommend that the instrumentation and control equipment that functions to initiate isolation of the control room should be Class IE. The Project has agreed to provide Class 1E power to the radiation detectors and sodium aerosol detector. The chlorine detector will be provided by control power. We find this acceptable.

(8) We are concerned that the procedure for handling containment isolation system circuits and equipment that do not meet the reliability requirements of MIL HDBK 217 have not been stated in the Final Safety Analysis Report. We recommend that the Project identify in the Final Safety Analysis Report the equipment that did not have its reliability verified following the procedures of MIL HDBK 217 and where similarity to qualified devices or equipment is used as the basis for non-verification. The differences from the qualified equipment should be stated and shown to have no effect on the overall reliability. The Project has agreed to provide this information and we conditionally find this commitment acceptable assuming we do not find dissimilarity.

(9) The Final Safety Analysis Report implied that the interim decay storage heat removal instrumentation may not be required to operate after a tornado or seismic event. In response to our request, the Project provided clarifying information that stated that the interim decay storage temperature monitoring system was powered from Class IE power and provided with both seismic qualification and tornado protection. We conclude that this is acceptable.

(10) The Final Safety Analysis Report did not state what measures would be taken to prevent inadvertent or unauthorized disturbance or maladjustment of calibrations, gains, setpoints or sensitivities of safety-related instruments in the reactor shutdown system, engineered safety features and all other safety-related instrumentation systems. The Project has responded to this concern by stating that such controls and adjustments will be inside racks that will be locked or require tools to take apart. We conclude that with the preceding commitment this position is acceptable.

We have concluded that the Project's description, design and implementation of the engineered safety features instrumentation system meet staff requirements and are acceptable.

7.4 Systems Required for Safe Shutdown

Our review of systems that fall in this category developed these concerns that were pursued with the Project:

(1) The Final Safety Analysis Report indicated that a single 125 volt DC bus energizes the controls of the circuit breakers for the three redundant closed loop system instrument buses. In response to our request the Project supplied

additional verbal information stating that these three instrument buses are non-class IE and do not supply power to the plant protection system inputs from the closed loop system. The Project has committed to redraft this position to clarify this issue. Since the review of the closed loops has been deferred, we will report further on this subject in a supplement to this report.

(2) The Final Safety Analysis Report did not present information about the accuracy of the accelerometers, the accuracy of the trigger level, drift of the trigger units, continuity of power supply and how the single failure criterion was met for the seismic monitoring system. This additional information was provided in a table in the response and batteries with a useful life of one-half hour have been provided in case of power failure. It has been stated that the probability of occurrence of an earthquake is low. It has also been stated that in the event of occurrence of a seismic event comparisons can be made against records from at least four other sites in the immediate area. Also the peak reading instrument is completely independent of all other instruments. We have concluded that this is acceptable.

(3) With regard to meeting the requirements of General Design Criterion 19, "Control Room," appropriate controls to shut down the reactor, isolate the containment and remove decay heat have been located outside the control room. These controls have been so designed that damage to equipment located in the control room will not interfere with their use to perform their safety-related function. We conclude that the Project has complied with the intent of General Design Criterion 19.

We have concluded that the Project's description, design and implementation of the systems required for safe shutdown meet the intent of staff requirements and are acceptable.

7.5 Safety-Related Display Instrumentation

The Final Safety Analysis Report did not have a Section 7.5 in which all safety-related display instrumentation was presented and discussed. We have, therefore, attempted to draw from other sections the issues relating to this category and present them as follows:

(1) In our review of Section 7.1.1 of the Final Safety Analysis Report we had not found adequate identification of critical and post-accident monitoring instrumentation. The Project revised Tables 7.1-3 and 7.3-1 and provided references to them. We have reviewed this response and are concerned that it may not be complete in that no indication of control rod position or verification of full insertion of the rods in the core is provided. Absolute rod position is one method of shutdown verification. An alternative method is flux measurement.

The Project has committed to providing a method of determining shutdown status within the control room. We recommend that the method proposed be safety grade. The commitment is acceptable to the staff. We will address the method selected in a future supplement.

(2) Our review of Section 7.1.4 of the Final Safety Analysis Report developed these concerns: (a) the design of the containment isolation system valve position indication circuits did not protect them from General Design Criterion 2 events (tornados) and energize them from Class IE power sources and (b) sufficient description was not presented in the Final Safety Analysis Report to assure that the design of the containment isolation system valve position indicating circuits, pilot solenoids and electric operators would not compromise the independence of the systems through the power sources. Further clarification provided by the Project stated that the control circuits are Class IE but the position indicators are not. Position indication and display is provided by control power. We find this acceptable for the FFTF providing that assurance is given that false indications will not occur on loss of power and that the facility will be provided with weather information to shut down when tornado conditions prevail. We find this set of conditions acceptable.

(3) The descripton of how the bypass and inoperable condition indication as recommended by Regulatory Guide 1.47 has been implemented for the plant protection system and other safety-related systems was incomplete. The Project will modify the Final Safety Analysis Report to state that no bypass indication will be provided for the engineered safety features since it is not possible to bypass any engineered safety feature functions. Plant protection system bypasses and status are continuously indicated on the control console or on the plant protection system status panel located in front of the control console. The revision will make clear that one bypass plug will bypass all protective functions in one channel of a closed loop system. Also the status of all protective features for a closed loop is displayed on the plant protection system status panel. For "experimenter trips" the Project states that these are merely unused protective functions for which no analog inputs are provided. These channels have their comparators wired to be always reset and no bypass status is provided for them. It has been stated that when and if these trips are used appropriate plant wiring changes will be made to conform to all plant protection system requirements. We have reviewed revisions to the Final Safety Analysis Report in this regard and have concluded that the response is acceptable.

(4) Section 7.3.2.1 of the Final Safety Analysis Report did not specify what the selected off-normal parameters to be annunciated were. The Project provided a listing of these parameters. We have reviewed this listing and found it to be acceptable.

(5) The emergency hatch in the equipment airlock was stated to be manually operated and was not provided with position indication in the control room. This design has been revised to include position indication for the airlock doors and the emergency hatch which indicates and is annunciated in the control room in the event of both doors on the personnel airlock being open, either door of the emergency airlock being open with the inner door, outer door or escape hatch of the equipment airlock open. This is redundant for the function and is acceptable.

(6) We believed that portions of the plant annunciator systems functioned as safety-related display instrumentation and as post-accident monitoring since they supply information to aid the operators in taking manual safety-related actions. The Project supplied additional information stating that no part of the plant annunciator system supplies safety-related information or functions as post-accident monitoring, therefore it should not be required to meet safety criteria. The information further stated that all engineered safety features and post-accident monitoring parameters did require display in the control room under both seismic and tornado conditions and that this equipment and its power supply meet the criteria for Class IE equipment. We have concluded that this is acceptable.

We have concluded that the Project's description, design and implementation of the safety-related display instrumentation systems will meet staff requirements and will be acceptable.

7.6 All Other Instrumentation Systems Required for Safety

The Final Safety Analysis Report did not have a Section 7.6 in which "All Other Systems Required For Safety" were presented and discussed. We have therefore attempted to draw from other sections the issues related to this category and present them as follows:

(1) Section 7.1.4 of the Final Safety Analysis Report referred to "certain bypasses." We requested additional information identifying these bypasses, stating when they are initiated and removed, describing their logic and status indication and listing the circuits and equipment that would be affected by operation, failure to operate, removal or failure to be removed. The response provided a reference to a new Final Safety Analysis Report table that identified and stated when they were to be initiated and removed. Figures have been added showing the logic, bypassing and status indication. All reactor shutdown system and heat transfer system shutdown circuits are stated to be affected, the bypasses are channelized and only one channel at a time can be affected by random failure. It is stated that comparators are automatically reset after trip but the additional information states that the channel trip indication is retained and must be manually reset. We find this response acceptable.

(2) We requested additional information regarding the range of disturbance conditions to be accommodated and the control limits for the dump heat exchanger controls, interlock or permissive functions and hardware redundancy provided to limit actual plant transients resulting from causes given in Item B20 of Final Safety Analysis Report Section 7.2.1.1.2. The response presented additional information that described the actions and equipment needed to reduce plant transients due to the listed malfunctions. We found this response acceptable.

(3) The Final Safety Analysis Report stated that for the closed loop system when transfer of control to an auxiliary circuit does occur and a silent failure exists in the switched to circuit, a scram will result in some instances. We requested that these instances be identified as well as those where scram does not result and that justification be provided for those cases where scram does not result. The response is not satisfactory in that the cases have not been identified and the justification stated is that in the worst case a silent failure in conjunction with a failure of the normal control system will cause a scram. These cases should be identified and specific description of how safety is still guaranteed by the plant protection system be presented. The review of the closed loop has been deferred. This is to be pursued further with the Project and we will report on it in a supplement to this report.

(4) The information given regarding measurements made by the heat transfer system instrumentation system did not include accuracies and there was no direct comparison of these values to those used in the hot channel factor analysis. The Project's response modified Table 7.3-3 to include the accuracies and stated that the accuracies presented in Tables 7.3-3 and 7.3-4 included allowances for the indicator accuracy. The values used in the hot channel factor analysis included additional allowance beyond those for the indicators and hence were conservative. We find this response acceptable.

(5) We had requested that the accuracy of the secondary hot and cold leg temperature measurements be provided. The response provided these and showed that these accuracies were well within the required accuracies for the hot channel factor evaluation. We find this acceptable.

(6) We also asked about the safety significance of the flow measurement in the intermediate heat exchanger recirculation line. The flow measurement is made to assure that plugging of this line does not occur and block the venting of gas from the top of the intermediate heat exchanger. There are no provisions made for alarming on loss of flow but readout is to be made once per shift by an operator from a local panel in the containment. We find the commitment to verify by-pass flow once per shift to be acceptable.

(7) We had requested that brief descriptions of component instrumentation and interlocks referred to in Item I of Final Safety Analysis Report Section 7.3.2.2.2 be provided and their safety significance be discussed. The response described this instrumentation and interlocks and provided discussions of their safety significance. We found this response acceptable.

(8) We requested that information be supplied that showed the energy dependence of the radiation monitoring sensors, particularly that its response at 2.75 Mev (^{24}Na) was not significantly reduced with respect to the response at 0.66 and 1.26 Mev. The response was shown to be essentially flat out to the ^{24}Na disintegration energy of 2.75 Mev. We find this acceptable.

(9) We believed that the reactor service building exhaust monitor and control room ventilation duct monitor perform safety-related functions and should therefore meet safety system requirements. The Project indicated that the offsite radiological consequences of the worst case event, assuming the failure of the reactor service building exhaust monitor, were within the requirements of 10 CFR Part 100. Therefore the operation of this monitor is not required for safety.

We also requested additional information about the mobile continuous air monitors that are located in containment regarding their qualification for and potential use as post-accident monitoring instrumentation. The response indicated that they are not intended for use as post-accident monitoring instrumentation and therefore are not qualified as safety-related equipment. We have found the response regarding the reactor service building exhaust monitor and the in-containment mobile continuous air monitors acceptable.

We also noted that three ex-containment area monitors have been specifically designated to supplement the containment isolation system monitors for long-term post-accident monitoring instrumentation and not supplied with Class 1E power. The Project has committed to supply qualified Class IE monitors to accomplish the long-term post-accident monitoring function. We find this commitment acceptable.

(10) Section 7.3.11.1.3 of the Final Safety Analysis Report stated that portions of the inert gas instrumentation and control system were required to be powered from the zero time outage (safety) power system. We requested information showing how the power feeds would be arranged to preserve both the independence of the redundant instrumentation and the redundant power sources. The Project responded that no portion of this instrumentation was required for safety and instead received power from the non-safety power system. The Final Safety Analysis Report section was to be modified to so state. The Project erroneously modified the wrong pages and will submit a correction. We found this response to be acceptable.

(11) We have determined that the cell gas pressure, oxygen content humidity and hydrogen content monitoring systems are a function of normal plant control and need not be powered by IE power.

(12) The Final Safety Analysis Report has presented little information regarding sodium leak detection. We believe that the sodium ionization leak detectors perform a safety-related function and that more information should be presented describing the power supply, indication, seismic and environmental qualification for these devices. We believe that the sodium ionization detectors should be safety-related and remain operable following a safe shutdown earthquake.

We have concluded that the Project design, description and implementation of "All Other Systems Required for Safety" are acceptable.

7.7 Control Systems Not Required for Safety

The control systems used for normal control of the operation of the reactor, the two closed loops, and the heat transfer system have not been extensively reviewed. The review that has been done has concentrated mainly on direct interfaces with the plant protection system. We have not investigated the dynamics of the control systems to reveal possible control instabilities or failure modes that could respond in a manner that would worsen the consequences of an incident or to make the task of the plant protection system to limit these consequences more difficult. Little review of drawings for these control systems was done. We believe that more attention should be given to performing such analyses of the non-safety control systems of this reactor. The limited review performed has developed the following concerns:

(1) Additional information supplied by the Project shows that the reactor plant control system provides two independent means to limit the rod withdrawal speed to limit the transients caused by control rod runaway. Also high flux and high flux-flow rod withdrawal blocks have been provided to limit the consequences of inadvertent rod withdrawal prior to the need to scram. We have concluded that this information indicates that there is no interaction with the plant protection system and that the control system will limit the effect of this incident. This is acceptable.

(2) The Final Safety Analysis Report, in Section 7.2.1.2.1, item 5, refers to circuits for intersystem interlocks. We requested that additional descriptive information be provided so that we could determine that the redundancy and independence of safety-related systems were not compromised by such interlocks. The response stated that none of these interlocks had any connection to the plant protection system and that they act to prevent inadvertent operator action that is not operationally desirable. It also presented descriptive information about interlocks in the central room drive mechanism, pump control,

dump heat exchanger control, primary isolation valve and secondary isolation valve control systems. We have reviewed this information and found that while some of these interlocks may be derived from plant protection system signal sources they are properly isolated from the plant protection system and there is no control action that acts to inhibit or prevent the performance of a safety function. This is acceptable.

(3) The Final Safety Analysis Report, Section 7.2.2.2.3, identified several auxiliary instrumentation systems for the closed loop system but presented no descriptive information about them or no evaluation of their safety significance. We requested such additional information on these systems. The response stated that these systems are not safety-related but gave no significant information describing the systems and their control functions. The statement was made that any perturbation to stable closed loop operation by an anomaly in these systems would result in a plant protection system reactor trip, if required to protect the public health and safety. No further information substantiating this statement was given such as what perturbations were considered, potential failure modes, possible control system instabilities and how the plant protection system trip would be arrived at. We believe this response on the closed loop system control systems to be inadequate. We will require additional information as outlined above before we can evaluate the safety significance of these systems. The review of the closed loops has been deferred. We will review this additional information and report further on it in a supplement to this report.

(4) We recommended that additional review of the non-safety control and instrumentation systems comprising the reactor plant control system should be performed with particular emphasis on the discovery of potential failure modes or control system instabilities that could interact with the plant protection system to prevent or hinder the plant protection system from taking action to terminate an accident or to limit its consequences. The Project stated that the analyses for the accidents enveloped the worst case failure modes or instability modes of the control system as was determined during the design studies for the control and protection systems. We have found this response acceptable.

Assuming that the open issues presented in items 7.7(3) and 7.7(4) are satisfactorily resolved we can conclude that the Project design, description and implementation of the "Control Systems Not Required for Safety" meet staff requirements and are acceptable.

8.0 ELECTRIC POWER

8.1 General

General Design Criteria 17 and 18, IEEE Standards including IEEE Criteria for Class IE Electric Systems for Nuclear Power Generating Stations (IEEE Standard 308-1971) and applicable Reguatory Guides including Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," and staff technical positions noted in Table 8-1 of the Standard Review Plan have been utilized as the bases for evaluating the adequacy of the protection and control of electric power systems. This section of the safety evaluation report reflects the results of our review of the FFTF Final Safety Analysis Report.

8.2 Offsite Power Systems

The reactor systems for the FFTF are designed to remove decay heat by natural circulation when offsite and onsite "emergency" alternating current power are not available. The small real emergency power requirement, mainly for selective electrical monitoring and control functions, is provided by the onsite 125 V DC Class IE battery power system. The capability of the reactor heat transport systems to remove decay heat by natural circulation in a safe and reliable manner without an alternating current electric power supply is to be demonstrated by verification testing as described in Section 14.2 of this report.

The offsite power system is the preferred source of power. This system includes the grid, transmission lines, transformers, substation components and associated control system provided to supply electric power to safety-related and other equipment.

The offsite power system proposed for the FFTF consists of a 115 kilovolt main preferred line and a 13.8 kilovolt alternate preferred line. The 115 kilovolt main preferred line feeds power from the Bonneville Power Authority Benton switching station to the FFTF 115/13.8 kilovolt substation transformer. The 13.8 kilovolt alternate preferred line of limited capacity will be connected from the 300 Area Substation to the FFTF 13.8 kilovolt switchgear. The alternate offsite power line is of sufficient capacity (three megavolt amperes) to supply the power requirements of the emergency power systems.

In the event the single 115 kilovolt transmission line is lost, the second 13.8 kilovolt power source from the 300 Area can be made available automatically to the 13.8 kilovolt buses in approximately 4 seconds.

The two offsite power sources are independent and do not share offsite right-of-way. Once on the FFTF, however, the two offsite sources, now both at 13.8 kilovolt, enter a common indoor switchgear but by isolated paths. This switchgear is a two bus system, each of which serves a single train of switchgear cubicles. The two trains of cubicles of different bus systems are arranged back-to-back. Sheet metal barriers extending the full length of the switchgear provide about nine feet of physical separation between the front and rear buses. Physical separation and separate routing are provided between cables connecting the two alternate systems. The incoming "A" bus breakers for the 13.8 kilovolt alternate supply and the 115 kilovolt normal supply are controlled by direct current battery power separate from that of the incoming "B" bus breakers for the 115 kilovolt and 13.8 kilovolt supplies. Thus, no single breaker failure or loss of a single direct current control power source will result in a loss of the offsite power supply.

A grid stability analysis has not been performed for the 115 kilovolt preferred power system because in the event of complete loss of the FFTF load the effect on the Bonneville Power Authority system as a whole would be insignificant. The FFTF will not produce electricity for outflow to the Bonneville Power Authority system, and, therefore, will not contribute to the Bonneville Power Authority system instability. Should a Bonneville Power Authority system instability occur and result in a severe voltage depression then the reactor will scram through the action of the plant protection system under-voltage relays.

Our review of the offsite power system for the FFTF covered single line diagrams, physical layout drawings and descriptive information for the offsite power system. The review also included the Project's proposed design criteria and design bases for the offsite power system and its analyses of the adequacy of those criteria and bases. On the basis of our review, we have concluded that the design of the offsite power system satisfies our requirements set forth in Section 8.1 of this report and is acceptable.

8.3 Onsite Power Systems

8.3.1 Emergency Power System

The onsite "emergency" power system is not a Class IE system and, as noted above, is not required for safety. It is comprised of two redundant and independent 480 volt emergency distribution systems. Three sources of power are available to power the emergency distribution systems. They are the offsite main preferred power, the offsite alternate preferred power and two 480 volt diesel generators. If the main preferred power is not available, the diesel generators are automatically started and the alternate preferred power is automatically connected to the 480 volt emergency distribution buses. If both offsite power supplies are lost, the diesel generator circuit breakers automatically close to connect the generators to the emergency distribution buses after the units reach rated speed and voltage

and the diesel generators are sequentially loaded. Loads on the emergency buses are divided into three categories: (1) loads that receive power as soon as a diesel generator is connected to its emergency bus (after a 20-second startup), (2) loads that are automatically programmed to start following connection of the emergency generator to its bus, and (3) loads that can be started manually but do not operate ordinarily. None of these loads is required to protect the public health and safety, and only minor damage to non-critical plant equipment might result if these loads were to lose power. In addition, these loads are not designed to meet the standards for engineered safety feature equipment (e.g., seismic, environmental, etc.).

The emergency diesel generators were obtained from Oak Ridge and have a continuous rating of 1100 kilowatts. The generators have a two-hour overload rating of 1210 kilowatts. The load applied to either diesel generator will be less than its continuous rating. Each diesel generator and its auxiliary systems are separately housed. However, the diesel generator rooms are not Category I seismic installations nor are they designed to withstand the effects of a postulated tornado and tornado missiles.

Although diesel generators are not Class IE systems, a series of qualification tests were run on the units to ascertain the capabilties of the two machines to start and carry the emergency loads within their prescribed time.

8.3.2 Safety Power System

In order to provide power for those loads that serve to maintain the plant in a safe condition at all times, a safety power system which is designed as a Class IE power system is provided. This system consists of two independent and redundant parts. Each system provides 125-volt direct current and regulated 120-volt alternating currents to the various safety power system loads.

The 125-volt direct current system is comprised of two redundant and independent 125-volt batteries. Each battery system includes a battery, a battery charger and a main distribution panel. The chargers are supplied from redundant 480-volt emergency buses. There are no bus ties between any division and each division is physically separated from the others. This complies with the recommendations of Regulatory Guide 1.6. Each battery system also supplies power to two inverters which convert the direct current power to 120-volt alternating current power for protection system instrumentation.

Each of the batteries has sufficient capacity to power the interruptable loads continuously for 15 minutes and critical loads continuously for one week without dependence on any power from any emergency system. It is planned that at about 15 minutes after a total loss of normal and emergency power all of the loads connected to both batteries, except the monitoring instruments, will be tripped, thus supplying

adequate power to instruments for one week. The batteries are provided with hydrocaps and are located in separate rooms of seismic Class 1E design.

120-volt alternating current instrument power is supplied from four independent panels. Two of these panels are connected to inverters which provide 120-volt alternating current power from the Class IE batteries upon sustained loss of preferred, alternate and emergency power and the other two panels are connected to redundant and independent buses energized by the zero-time outage motor-generator sets. The inverters supply the critical loads while the loads supplied by the zero-time outage motor-generator sets are required during operation of the reactor and 15 minutes after the loss of all power from the emergency bus. After 15 minutes power need not be provided to these loads. So long as power is available from the emergency power system, the zero-time outage motor-generator sets are loads of the respective emergency buses. If the emergency power is not available, the zero-time outage motor-generator sets become loads on the direct current batteries and are tripped after 15 minutes to conserve battery power. The design of the instrument power system affords adequate independence to assure that no single failure will result in a loss of power to redundant instrument channels.

Our review of the onsite power system for the FFTF covered single line diagrams, physical layout drawings, functional logic diagrams and descriptive information for those systems and auxiliary systems that are vital to the proper operation of the onsite Class IE system and its connected loads. The review also included the Project's design bases and proposed design criteria for the onsite Class IE system and for the vital supporting systems, and the analyses of the adequacy of those criteria and bases. On the basis of our review, we have concluded that the onsite power system satisfies our requirements set forth in Section 8.1 of this report and is acceptable.

8.3.3 Physical Independence of Electrical Systems

The Project provided in the Final Safety Analysis Report criteria for physical separation of electrical equipment to preserve the independence of redundant equipment. In addition, they have stated that the FFTF design and construction was in an advanced stage at the time IEEE-384 and Regulatory Guide 1.75 came into effect and, therefore, the design would not fully meet the recommendations of Regulatory Guide 1.75, "Physical Independence of Electric Systems."

We have reviewed the proposed design criteria and conclude that the Project design criteria appears to be satisfacotry. However, we have not had the opportunity to verify the implementation of their design criteria since the Commission does not have an inspection or inforcement responsibility for the FFTF.

8.3.4 Environmental Qualification of Class IE Equipment

Due to the lack of information presented in the Final Safety Analysis Report, we were unable to determine the acceptability of the environmental qualification of Class 1E equipment. The Project has submitted additional information for our review. After we have reviewed this information, we will report our findings in a supplement to this report.

8.3.5 Power Lockout to Manually-Controlled Electrically Operated Valves

The Project identifies primary loop isolation valves and secondary loop isolation valves for which Branch Technical Position EICSB 18 applies. They also provided a detailed schematic diagram showing all control power sources and status indication for the primary and secondary isolation valves and their associated power lockouts. We have reviewed this information and conclude that the valve position indication system will perform its function under all conditions, including valve power lockout. We find this acceptable.

8.3.6 Electrical Connectors and Terminal Blocks

The staff is currently evaluating all nuclear facilities to determine the adequacy of the environmental qualification testing of electrical connectors and terminal blocks used in safety-related systems located inside or outside the containment. We will pursue this matter with the Project and provide our findings in a supplement to this report.

8.3.7 Electrical Penetrations

The Project has documented that electrical penetration conductor short circuit protection is designed so that the cross-sectional area of the module feed-through conductors is in all cases greater than the attached cable; thus, the feed-through conductor temperature during a short circuit event will always be less than that of the attached cable. This aspect of the design using self-fusing characteristics of the outside conductor for assuring containment penetration integrity is unacceptable because of the potential of causing a fire. The resulting fire damage to safety-related systems would produce even more hazardous safety problems. Therefore, we recommend that the design be modified to provide for independent primary and backup fault protective devices (breakers) that meet the following requirements of IEEE-279 1971.

(1) The system should, with precision and reliability, automatically disconnect power to the penetration conductors when currents through the conductors exceed the preset limits.

(2) All source and feeder breaker overload and short circuit protection systems should be qualified for the service environment and seismic requirements. The seismic qualification for non-Class IE circuit breaker protection systems should, as a minimum, assure that the protection systems remain operable during an operating basis earthquake.

(3) The circuit breaker protection system trip setpoints should be compatible with the capability for test and calibration. Provisions for test under simulated fault conditions should be provided.

(4) No single failure should cause excessive currents in the penetration conductors which would degrade the penetration seals.

(5) Signals for tripping source and feeder breakers should be independent, physically separated and powered from separated sources.

On July 12, 1978, supplemental information was provided indicating that the Project considers the design to be in compliance with Regulatory Guide 1.63 (1973) and the staff's proposed change to be not cost effective.

We will reconsider this situation, taking into account the status of construction, and will provide our recommendation in a supplement to this report.

9.0 AUXILIARY SYSTEMS

9.1 Summary Decription of Fuel Storage and Handling

The FFTF fuel storage, handling and examination facilities are substantially different from those encountered in light water reactor practice. There are four principal reasons: (1) use of non-transparent sodium as a coolant, (2) refueling temperature of 400°F, (3) non-removable reactor head and closed test loops in the reactor, and (4) presence of fuel test and examination facilities.

Our area of review has been directed to the safety aspects of the movement of fuel through the facility within the reactor containment building and not with the preservation of test specimens which may be substantially different during the life of the facility within the closed loops.

Our acceptance criteria is based on comparability in terms of light water reactor safety, i.e., compliance with the intent of General Design Criterion 2, Regulatory Guide 1.29 and Branch Technical Position ASB 9-1 and an evaluation of the fuel transfer cycle for recriticality, accidental radiological releases through fuel overheating, sodium fire potential, radioactive gaseous releases through seals, and equipment malfunction of an electrical or mechanical nature, including loss of power during fuel transfer. We reviewed each of the eight major elements of the reactor refueling system within the containment. The reactor refueling support system and loop preparation equipment were not reviewed since insufficient information was provided to perform an evaluation and at the present time shipment of spent fuel for reprocessing is not provided for. It is likely that individual irradiated pins may be shipped from the FFTF to other facilities in the future, but insufficient information exists to make an evaluation.

The elements of the reactor refueling system reviewed are as follows:

(1) In-Vessel Handling Machine
(2) Closed Loop Ex-Vessel Machine
(3) Floor Valve
(4) Interim Decay Storage
(5) Bottom Loaded Transfer Cask
(6) Conditioning Station
(7) Interim Examination and Maintenance Cell
(8) Plug Handling Fixture

Fuel subassemblies may be transferred from core positions following shutdown when they have decayed to 40 KWt, 10 to 12 hours after shutdown. It takes this much time to cool the reactor to its refueling temperature (400°F).

The core subassembly transfer operation takes place under the sodium pool in the reactor and is performed by the three in-vessel handling machines which deposit the subassemblies in one of the reactor fuel storage positions in a core component pot behind a neutron shield. Prior to beginning the in-vessel transfer operation the control rod drive lines are disconnected and the instrument tree is withdrawn where it cannot interfere with movement of the three in-vessel handling machines. Only one in-vessel handling machine is operated at a time. The transfer operation deposits the fuel subassembly into a core component pot which is immersed in sodium and allowed to remain there until it decays to a power level of 10 KWt, 10 to 12 hours after shutdown.

The closed loop ex-vessel machine is a gantry-mounted moveable machine designed to handle subassemblies at a decay power level of 10 KWt with the subassembly inserted in the sodium-filled pot. It is used to transfer driver fuel or various core components to and from reactor storage or other core positions, to interim decay storage or to the interim examination and maintenance cell and core component conditioning stations. The closed loop ex-vessel machine is also used to transfer in reactor closed loops and open test assemblies to and from the reactor. A sodium drip pan and environmental confinement closure valve are provided at the lower end of the closed loop ex-vessel machine which matches to the floor valve over the reactor. The heat transfer from the sodium-filled pot in the closed loop ex-vessel machine is by radiation to a chrome oxide (black) cold wall inside the machine and inside of its shielding. Either the outside of the cold wall is cooled by forced convection or, in the event of loss of power, the sodium pot is cooled by natural convection by manually operating dampers around the cold wall. Other ancillary safety functions in the closed loop ex-vessel machine allow for hand cranking down a sodium-filled core component pot back into the reactor, manual operation of the argon filled seals and movement of the transporter by manual jacking. Approximately 10 hours are available in the event of loss of cooling to a fuel subassembly in a core component pot allowing time for reinsertion into the reactor or establishing convective cooling before sodium boiling in the pot begins.

The floor valve is basically a shielded rotating valve. The valve disc has double seals on both internal sliding surfaces and the valve body has double seals on both external mating surfaces. Valve operations are performed from a self-contained console or from the mating machine. The dimensions of the valve are such that a full length driver fuel assembly and core component pot cannot be fully contained in the dimensions of the valve. The floor valve functionally can withstand 6 KWt for 20 hours based on a 10 KWt assembly stalled in the thru-port. This is ample time to manually crank down a fuel subassembly in a core component pot back into the reactor. A self-contained battery and gas accumulator will sustain the floor valve operation and seals for three hours.

The interim decay storage vessel consists of an upper large vessel with a rotating storage drum and a central smaller diameter vessel of longer length attached to the bottom of the larger vessel. The entire dual vessel assembly is filled with sodium. Driver fuel and test assemblies are stored in the outer rotating basket and closed loops in the central long vessel region. The rotating drum is driven by a portable motor drive at the operating floor. A guard vessel is provided to limit the loss of sodium to cool the immersed fuel in the event of an interim decay storage vessel leak. Two independent cooling systems are provided. The normal cooling system circulates sodium from the interim decay storage vessel to a NaK system which in turn transfers its heat to a Mobil-therm cooler followed by a chilled water heat exchange system which rejects the decay heat to air. A backup nitrogen cooling system is provided to the interim decay storage through an annulus between the insulation jacket and the guard vessel. In the event both cooling systems are lost, valves in the ventilation ducts in the N_2 cooling system to this annulus can be opened to allow building air to naturally convect up the exterior of the interim decay storage guard vessel to remove decay heat.

The bottom loading transfer cask primarily handles bare fuel subassemblies, new or irradiated, transferring them between the conditioning stations, interim decay storage or through containment to a future cask loading station. The bottom loading transfer cask thermal rating is 1.4 KW which corresponds to approximately 200 days decay for a driver fuel subassembly. The conditioning stations have the main function of preparing various core components, test assemblies, and hardware for operation, or storage in liquid sodium. Examined irradiated fuel pins may also be stored in the core component conditioning stations if their total power level is not greater than 1.0 KWt, and test assemblies may be stored in the test assembly conditoning station, if their combined decay heat level is less than 9 KWt. Nitrogen is circulated around the conditioning stations to cool them.

The interim examination and maintenance cell is basically a hot cell with all the ancillary equipment to non-destructively disassemble and examine core components including driver fuel assemblies and test assemblies. Component cooling is provided as well as space cooling by circulating argon which is part of the heating and ventilation system. A redundant blower is provided to the space cooling as backup to the primary blower. Loss of all power to direct subassembly cooling for more than 40 minutes would result in pin failures. Fuel clad melting could occur in two hours if directed cooling is not restored. However, all radioactive products released from pin failures would be in the interim examination and maintenance cell which is located inside the containment building.

The plug handling fixture removes, transports and replaces refueling shield plugs. The plug handling fixture works in conjunction with the floor valve and, in the event of loss of power to the plug handling fixture, the shielded valve in it may be manually closed.

We have reviewed the design criteria, functional requirements feature or prototype tests and verification test reports performed for prototype or actual components and the FFTF system safety evaluation assessment for the above components on an individual basis and on the basis of an integrated functional system. Our evaluation of the preceding fuel handling components is that physical separation always exists everywhere except within the interim examination and maintenance cell such that the K_{eff} is always less than 0.9. Administrative controls limit the number of fuel subassemblies permitted in the interim examination and maintenance cell to no more than three to preclude the possibility of criticality from a close packed array of subassemblies. The equipment has been designed structurally to survive the safe shutdown earthquake and the design basis tornado but the possibility of component seal leakage does exist for natural phenomena events. The radiation fields in routinely occupied areas during normal fuel handling operations are limited to 0.2 mrem/hr and higher values are permitted for brief time intervals during actual transfer operations at joints or mating surfaces. Pure bottled argon, or low level reprocessed argon cover gas, is provided around all sodium-wetted surfaces during fuel handling to prevent oxidation or combustion of sodium and positive directional control of the environment in the event of release of gaseous fission products from a fuel assembly. Double inflatable seals are used at matching faces and an argon buffer gas is used between the seals to control and prevent outward radioactive gaseous release into inhabited areas. Fuel clad temperatures are limited during transfer operations to a maximum of 1500°F to minimize the possibility of pin failures during handling of end-of-life driver fuel. Lower temperature limits have been set for test elements to preserve test data for test assemblies.

Mechanical indexing and electrical interlocks are provided to prevent improper insertion and withdrawal of fuel subassemblies in core positions as well as limiting the forces that can be applied to the subassemblies during core component transfer operations. The ex-vessel transfer machine is provided with redundant blowers and is also connected to onsite emergency power to minimize the possibility of loss of cooling to a fuel subassembly while in transport. The ex-vessel transfer machine is also designed for natural convection cooling around the cold wall without exceeding the 1500°F cladding criteria in the event of loss of all electrical power. Other components in the reactor refueling system are connected to emergency power as well but interrupted periods of loss of power can be tolerated in all cases except for the interim examination cell when a subassembly is undergoing examination. The interim examination cell is a hot cell and is fully contained in the event of pin ruptures.

The interim decay storage and interim examination cell are more dependent on power. However the interim decay storage pool has a large heat capacitance and can withstand loss of forced convection cooling for considerable time.

The remaining components in the fuel handling system, such as the conditioning station, floor valve, plug handling fixture and bottom loaded transfer cask, can withstand a power interruption for a matter of hours.

All of the components in the reactor refueling system have undergone extensive prototype testing and plant components will undergo further acceptance testing prior to startup of the facility.

Only one component of the fuel handling system will be subjected to core disruptive accident loadings and that is the in-vessel handling machine (actually three machines). The in-vessel handling machine has been evaluated as being able to accommodate the dynamic structural load of the core disruptive accident without becoming a missile. However, some sodium leakage may occur above the reactor head into the head access area.

Our evaluation and findings are that the FFTF fuel handling system, other than the reactor refueling support system and the loop preparation equipment which have not been evaluated, is in accordance with the analysis and descriptive material furnished by the FFTF Project and complies with the general intent of General Design Criteria, Regulatory Guides and industrial standards when considered with respect to the uniqueness of this type of reactor, using liquid metal as coolant, and that it is a low power test reactor having specialized fuel handling requirements not parallel to those encountered in light water reactors.

9.2 Facility Systems

9.2.1 Component and Compartment Cooling System

The component and compartment cooling system provides cooling capability for all plant components and compartments that require heat removal during plant operation. The system is not required to actively function for safe reactor shutdown. The component and compartment cooling system is normally powered by offsite power supplies and backed up by onsite diesel generators which are not a Class IE power supply. The system includes the gas cooling system, in-containment chilled water system, intermediate Mobil-therm system, chilled water system, and cooling water system.

The gas cooling systems maintain the temperature in the inert cells and heat compartment within the required operating range, provide cooling to certain reactor components and furnish backup decay heat removal for the interim decay storage. The gas cooling system circulates inert gas through coolers which transfer heat to the in-containment chilled water system. Normally the interim decay storage decay heat is removed by direct cooling of the interim decay storage sodium. In the event that both the sodium and nitrogen cooling system are inoperable, the interim decay heat can be dissipated to the containment atmosphere by natural convection flow of air through the interim decay storage vessel annulus.

The in-containment chilled water system is the next stage in the heat removal chain after the gas cooling systems for most of the in-containment heat loads. The in-containment chilled water system contains a water-glycol coolant which is circulated by centrifugal pumps and cooled by water chillers. The in-containment chilled water system piping and valves inside the containment are physically separated from any sodium-containing piping and equipment to preclude the possible direct contact of water and sodium. The intermediate Mobil-therm system uses Mobil-therm light heat transfer oil as a cooling medium. The system is designed to remove heat from the interim decay storage primary sodium processing system cold traps through the NaK heat exchangers. The Mobil-therm light oil does not react with NaK in the event of the accidental mixture of the two fluids. The system transmits heat to the cooling water system through Mobil-therm light to water heat exchangers.

The chilled water system uses water-glycol coolant and provides cooling for heat sources outside the containment. The system is backed up by the standby chilled water system which is powered from the diesel generators.

The cooling water system provides cooling water to the heat exchangers of the water chillers in the above systems and to the plant air and gas compressors. The system consists of eight cooling towers and eight circulating pumps.

The gas cooling system and the inside containment portions of the in-containment chilled water system and intermediate Mobil-therm system are designed to seismic Category I requirements and the remaining systems are designed to non-seismic requirements.

Based on the review of the component and compartment cooling system design, we conclude that the system can perform its design function and that a postulated failure of the system will not affect the safety function of any systems required for safe reactor shutdown and, therefore, is acceptable.

9.2.2 Inert Gas System

The inert gas system is designed to establish and maintain cover gas atmosphere and operating pressures for liquid metal systems and provide atmospheric inerting and pressure control for all cells and pipeways containing primary sodium equipment or piping. The inert gas system is not required to actively function for safe reactor shutdown. However, portions of the system serve as part of either the reactor pressure boundary or the containment boundary. Also, the radioactive argon processing system and the cell atmosphere processing system contain significant radioactive gases. This system includes the argon system, nitrogen system, radioactive argon processing system, and cell atmosphere processing system.

The argon system supplies argon gas to all liquid metal systems for use as cover gas. The argon is also used to service pressure-buffered seals and argon atmosphere cells. Argon pressures are regulated by pressure control systems and pressure relief valves which discharge into inert atmosphere cells. Vapor traps are provided to prevent liquid metal vapors and particulates from contaminating the argon supply system.

The nitrogen system provides the inert atmosphere and actuating gas for pneumatic valves located in inert atmosphere cells. A sampling and analysis subsystem is used to monitor the oxygen, hydrogen, moisture, and radioactivity content of the nitrogen atmosphere in each inert cell periodically to maintain purity of the nitrogen within specified limits.

The radioactive argon processing system and cell atmosphere processing system purify contaminated argon from the reactor and primary closed loop systems and potentially contaminated gases from inert cells. The purified argon is returned for use as a reactor cover gas and the purified gas from the cell atmosphere processing system is released to the environment.

These portions of the inert gas system that are required to maintain the pressure boundary integrity are designed to seismic Category I requirements.

Based on our review, we conclude that the system design is in conformance with General Design Criterion 2 and the guidelines of Regulatory Guide 1.29 with regard to seismic design and, therefore, is acceptable.

9.2.3 Failed Element Detection System

For the purpose of this review, the FFTF reactor will operate with no failed fuel. Since it has been determined that, from a safety point of view, the reactor can operate with up to 1 percent failed fuel, the issue of failed-fuel detection capability for low levels of failure is not of major concern for this safety evaluation. In addition the failed-fuel detection system is not part of the plant protection system, as it serves as a monitoring function only. Nevertheless, the staff has reviewed the failed fuel monitoring system planned for the FFTF and the experience with similar systems in other reactors. It is our opinion that the failed fuel monitoring in the FFTF reactor will prove effective for the detection and location of failed fuel and failed absorber pins.

The failed-fuel monitoring system provides the equipment necessary to detect, characterize, locate, and verify fuel and absorber pin failures. Thus it is capable of:

(1) Continuously monitoring the reactor cover gas for gaseous fission product activity (Xe-133, Xe-135, and the 31 Kev cesium X-ray) and indicating abnormal activity levels.

(2) Continuously monitoring the reactor cover gas for activated tag gas (Xe-125) and helium released from failed absorber assemblies and indicating abnormal activity levels.

(3) Continuously monitoring the reactor primary sodium coolant for delayed neutrons and indicating abnormal activity levels.

(4) Performing above functions 1 and 3 for the closed loop subsystem.

(5) Locating failed fuel and absorber assemblies by the analysis of the continuous "on-line" data and by analysis of the constituents of sampled primary and closed loop sodium and cover gas. The primary system for determining the location of failed fuel and absorber pins is the gas tagging failure location system. This system is based on loading a unique gas tag consisting of a mixture of enriched stable xenon and krypton isotopes (Xe-124, Xe-126, Xe-128, Xe-129, Kr-78, Kr-80, Kr-82, Kr-83) into all pins of each fuel and absorber assembly. If a pin fails, a portion of the tag gas escapes to the reactor cover gas to be sampled by the gas tag sampling trap. The identity of the failed assembly is determined by mass spectrometer analysis of the sample.

Based on the diversity and redundancy of the failed fuel detection system, the state of the detection technology and positive reactor experience with these subsystems, it is our opinion that the system will prove effective in determining pin failure well below the 1 percent failure limit for which the FFTF is adequately designed.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

The compressed air system is designed to provide both instrument air and service for the facility. The system consists of three air compressors, each sized for fifty percent of the design capacity, air receivers, filters, dryers, and standby air cylinders.

The compressed air system is classified as non-safety-related except for the portions that penetrate containment walls, including isolation valves which are designed to Quality Group B and Category I requirements. Also, standby air cylinders are provided at the heat transport system dump heat exchanger controls. (These standby air cylinders are safety-related since they provide emergency air supply to establish control for decay heat removal of the facility. The standby air cylinders and the associated piping and valves are designed to Quality Group D and seismic Category I requirements. Because of their safety significance, we recommend that the Project upgrade the emergency air supply system to Quality Group C or above.)

Further, the air cylinders at loop 2 and loop 3 of the heat transport system dump heat exchanger are not protected from tornado missiles. This is consistent with the present heat transport system dump heat exchanger design that only loop 1 is protected from tornado missiles.

Based on our review, we recommend that the compressed air system design is not acceptable with regard to the quality group design classification of the safety-related standby air cylinders.

9.3.2 Process Sampling System

The process sampling system (impurity monitoring and analysis system) is designed to provide representative samples of radioactive, as well as non-radioactive, fluid and gas streams from systems throughout the plant for chemical and radiochemical analysis. The seismic design and quality group classification of sampling lines and components conform to the classification of the system to which each sampling line and component is connected as described in Regulatory Guides 1.26 and 1.29. The process sampling system consists of piping, fittings, isolation and throttling valves, sample heaters, sample compressors, gas chromatograph, and instrumentation. The process sampling system is designed to obtain representative samples from the systems and components listed in Table 9.3-1.

Our review included the provisions to sample all principal fluid and gas process streams associated with plant operation. The review has included descriptive information for the process sampling system and the location of sampling points, as shown on piping and instrumentation diagrams.

The basis for acceptance in our review has been conformance of the design for the process sampling system to applicable regulatory guides. Based on our evaluation, we find the proposed system to be acceptable.

TABLE 9.3-1

PROCESS SAMPLE SYSTEM LOCATIONS

Primary Sodium System
Secondary Sodium System
Primary Argon Cover Gas System
Closed Loop System
Secondary Argon Cover Gas System
Closed Loop Cover Gas System

9.3.3 *Liquid Metal Receiving Storage and Processing System*

The liquid metal receiving storage and processing system is designed to supply all sodium systems with reactor grade sodium and provide sodium storage purification for the reactor, heat transport, and interim decay storage system. It also supplies NaK to two NaK systems in the facility and provides NaK storage and purification capability. The liquid metal receiving and storage system is not required to actively function for safe reactor shutdown. However, portions of the system serve as part of either the reactor pressure boundary or the containment boundary. This system includes the sodium receiving and unloading system, reactor primary sodium storage processing system, interim decay storage and primary sodium storage processing system, NaK interim decay storage cooling system, NaK cold trap cooling system, and reactor secondary sodium processing system.

The sodium receiving and unloading system provides the capability to melt the contents of a sodium tank car and transfer the sodium to the system storage vessels and to melt the contents of a sodium drum and transfer the sodium to and from the various sodium systems.

The reactor primary sodium processing system in conjunction with the NaK cold trap cooling system provides sodium overflow, storage and purification for the reactor and primary heat transportation system. The NaK serves as a cooling medium for the primary sodium cold trap.

The interim decay storage and primary sodium storage processing system in conjunction with the NaK interim decay storage cooling system provides storage, cooling, and purification capability for the sodium of the interim decay storage vessel. The NaK serves as a cooling medium for the interim decay storage vessel primary cooling loop.

The reactor secondary sodium processing system provides sodium storage and purification capability for the sodium of the secondary heat transport system and the secondary closed loop system.

All portions of the system which contain radioactive sodium are located inside the containment in shielded, inert gas-filled spaces to minimize the fire potential involving radioactive sodium. The portions of the system that are required to maintain the pressure boundary integrity are designed to seismic Category I requirements. The system is also designed to preclude the possibility of an accidental drain of the primary and secondary heat transport system, interim decay storage vessel, or the reactor vessel by a single active failure.

Based on our review, we conclude that the system design meets single failure criterion as to prevent accidental drain of the sodium system and it is in conformance with General Design Criterion 2 and the guidelines of Regulatory Guide 1.29 with regard to seismic design and, therefore, is acceptable.

9.4 Heating and Ventilating Systems

9.4.1 Control Room Area Ventilation System

The control room area ventilation system is designed to maintain the control room within the thermal and air quality limits required for operation of the plant controls and uninterrupted safe occupancy during normal operation, shutdown and post-accident conditions.

The control room area ventilation system includes a control area air conditioning system and a control area life support system. The control area air conditioning system consists of two 100 percent capacity air conditioning units, one 100 percent capacity return and exhaust fan and a duct system that distributes conditioned air. The control area life support system consists of two 100 percent capacity outside air supply filter-fan units and an air conditioning unit with two 100 percent capacity supply fans. The control area is normally served by the control area air conditioning system with the control area life support system on standby. When a high-high radiation is sensed at the outside air intakes, the control area air conditioning system is automatically isolated and the control area life support system is then operated in the recirculation mode. Manual selection of one of two outside air intakes can be made for the control area life support system outside air supply to maintain a positive pressure inside the control room. This outside air is also monitored for radiation level at air intakes downstream of the filters. The control area life support system is connected to the diesel power supplies.

Both the control room area ventilation system and the control area life support system are classified as non-safety-related systems except the portions that are necessary for isolation of the control room area (e.g., isolation dampers and their connected ductwork) and are designed to seismic Category I requirements. The instrumentation and control equipment that functions to activate the control room isolation is also safety-related and is evaluated in Section 7.6 of this report.

Since both the control area air conditioning system and control area life support system are classified as non-safety-related, the Project performed an analysis assuming complete loss of the cooling and heating function of the control area air conditioning system and control area life support system with worst local weather conditions and concluded that the temperature inside the control room at seven days after the incident will be maintained with temperature limits for operation of the plant controls and safe personnel occupancy. We agree with this conclusion.

The original plant design provided one seismic Category I and tornado missile protected isolation damper on each air intake and exhaust of the control room area ventilation system. In response to our request, the Project, in a letter dated March 27, 1978, proposed to modify the system by addition of redundant isolation dampers in each of the control room area ventilation system air intakes and exhausts. These redundant dampers are designed to seismic Category I requirements. However,

they are not tornado missile protected. We recommend that the Project provide tornado missile protection to both redundant isolation dampers in each of the control room area ventilation system air intakes and exhausts.

9.4.2 Auxiliary Equipment Area Ventilation System

The auxiliary equipment area ventilation system provides cooling and ventilation capabilities in various auxiliary equipment areas of the facility. The system is not required to actively function for safe reactor shutdown.

Local air conditioning and ventilation systems are provided in auxiliary and radwaste areas. Recirculating air coolers are also provided in those areas where supplemental cooling is required. Exhaust systems with filtering capability are provided to serve the access control area of the auxiliary equipment building, the heat transport system service building, and emergency generator areas.

The reactor service building heating and ventilating system maintains pressure control, cooling, and heating of the work areas in the reactor service building.

The diesel generator room supply fan system maintains space temperature suitable for diesel operation. The safety-related battery rooms are provided with once-through ventilation systems which are connected to power supply from the diesel generators. Each cell in the two battery systems is equipped with a catalytic hydrogen-oxygen combiner cell cap which prevents the escape of hydrogen into the room. Cap effectivity has been demonstrated by tests. The battery room ventilation system is, therefore, only a backup to control possible hydrogen buildup inside the battery room. The ventilation system will not be available if both the offsite and diesel power supplies are lost. The associated battery charger will also lose power, so there will be essentially no in-cell hydrogen production in the situation.

The auxiliary equipment area ventilation system is not designed to seismic Category I requirements.

Based on our review, we conclude that the system can perform its design function and a postulated failure of the system will not affect the safety function of any systems that are required for safe reactor shutdown and, therefore, the system design is acceptable.

9.5 Service Systems

9.5.1 Fire Protection System

During our review of the fire protection system, we requested that the Project conduct a reevaluation of the proposed fire protection provisions, and that those provisions be compared, in detail, with the guidelines set forth in our Branch Technical Position ASB 9.5-1, "Guidelines for Fire Protection for Nuclear Plants."

The Project's response is currently under review and we will report our findings in a supplement to this report.

9.5.2 Communication Systems

The communication systems provided at the FFTF include all components and auxiliary systems necessary to provide adequate and effective communication throughout the facility and other offsite emergency control centers under all conditions of operation.

The communication systems provided include an operation system, private automatic exchange system for general facility use, general plant telephone system for facility and offsite communications, public address system, and an interim examination and maintenance call system. In addition, the above communications systems are supplemented by mobile radio units patrolling the FFTF site, vehicle radio units patrolling other Hanford sites, an exclusive telephone connection between the FFTF and offsite emergency control centers, a manual switch in the FFTF control room to signal patrol headquarters in the event of an emergency, a crash alarm telephone provided in the FFTF control room and at the emergency shutdown panel to initiate or receive emergency calls, walkie-talkie equipment and messengers.

The FFTF communication systems equipment includes walls and desk type telephones, telephone jacks, permanent and portable headsets with microphones, an automatic central exchange, battery-supported power supplies with chargers, amplifiers, loudspeakers and interconnecting wiring.

The scope of review of the various communication systems included verification that the FFTF installed systems are powered from dependable power sources and are capable of providing diverse means for notification of personnel and initiation of evaluation procedures in the event of an emergency. The review also included verification that the installed communication systems provide the capability of communicating with all areas of the FFTF and with emergency control centers at the Hanford site under all conditions of operation and under maximum potential noise levels.

The basis for acceptance in our review has been conformance of the design criteria and bases and design of the installed communication systems to the acceptance criteria in Section 2.0 of the Standard Review Plan as applicable, and industry standards, and the ability of the systems to provide effective communications within the FFTF and with other emergency control centers at the Hanford site during normal and emergency conditions under maximum potential noise levels.

We have concluded that the installed communication systems at the FFTF conform to the above cited standards and criteria and are, therefore, acceptable.

Special requirements needed for the communication systems to satisfy Appendix "A" to Branch Technical Position APCSB 9.5-1 will be reviewed separately at a later date during the fire protection review of the FFTF. Additional recommendations may be proposed to further improve the capability of the communication systems resulting from the fire protection review.

9.5.3 Lighting System

The lighting system for the FFTF includes all components necessary to provide adequate lighting throughout the facility under all conditions of operation. The lighting system for the facility consists of normal and emergency lighting systems.

The normal lighting system, supplied from the 120 volt AC building electrical power system powered by offsite power, provides lighting throughout the facility.

The emergency lighting system provides lighting throughout the facility upon loss of offsite power. Four types of emergency lighting are provided. The first type consists of 120 volt AC fixtures powered from the 480 volt AC emergency power systems 1 and 2 which are fed by the emergency diesel generators. The lights are arranged so that if only one emergency power system is available, no normally lighted areas will be completely dark. Also, a few selected and distinctively identified 120 volt AC outlets powered from the 480 volt AC emergency power systems 1 and 2 are provided to power extension cord type lighting where such a need exists.

The second type of emergency lighting provided is a 120 volt AC inverter powered system. On loss of offsite and onsite emergency power, all 120 volt AC exit lights and selected control room lights will be powered by one of eight power units, each of which contains a storage battery and 125 volt DC/120 volt AC inverter. This system is automatically energized upon loss of all alternating current power. The storage battery powers the connected load for 90 minutes. Chargers are provided to maintain batteries fully charged during normal operation.

The third type of emergency lighting consists of fixed and portable wall-mounted lights, with integral batteries and chargers, each capable of providing light for 90 minutes. These units are connected to the 120 volt AC lighting circuit powered by the emergency power system and are installed in stairways and areas used for egress. All portable light units are plugged into 120 volt AC outlets to maintain them fully charged; fixed and portable units are automatically energized upon loss of alternating current power.

The fourth emergency lighting system provides a limited amount of control room lighting powered from the 120 volt DC safety power system Class IE batteries. Four special control room outlets, two from each Class IE battery, are provided for extension cord-type light with stipulation of a maximum load of 100 watts per battery.

The scope of review of the lighting systems included assessment of the number and types of lighting systems provided, assessment and adequacy of the power sources for the normal and emergency lighting systems, and verification of functional capability of the lighting system under all conditions of operation.

The basis for acceptance in our review has been conformance of the design bases and criteria, and design of the lighting systems and necessary auxiliary supporting systems to the acceptance criteria in Section 2.0 of the Standard Review Plan as applicable, and industry standards, and the ability to provide effective lighting in all areas of the facility under all conditions of operation.

We have concluded that the various lighting systems provided at the FFTF conform to the above cited standards and criteria and are, therefore, acceptable.

Special requirements needed for the emergency lighting system to satisfy Appendix "A" to Branch Technical Position APCSB 9.5-1 are being reviewed separately and will be addressed in a supplement to this report. Additional recommendations may be proposed to further improve the capability of the lighting system resulting from the fire protection review.

9.5.4 Emergency Diesel Generator Auxiliary Systems

The reactor systems for the FFTF are designed to remove decay heat by natural circulation when offsite and onsite "emergency" alternating current power are not available. The small real emergency power requirement, mainly for selective electrical monitoring and control functions, is provided by the onsite 125 volt DC Class IE battery power system. The capability of the reactor heat transport systems to remove decay heat by natural circulation in a safe and reliable manner without an alternating current electric power supply is to be demonstrated by tests.

The emergency diesel generators installed at the FFTF do not perform a safety-related function, and, therefore, they do not provide an onsite Class IE emergency electric power source as required at a commercial nuclear power plant. Actually the emergency diesel generators function only as an auxiliary non-Class IE backup power system for the offsite power system in the event it fails. The emergency diesel generators are not required for safe reactor shutdown, therefore, the diesel generators and their support auxiliary systems are designed to non-seismic requirements in accordance with commercial industry standards and codes.

Each emergency diesel generator is provided with the following auxiliary systems: (1) fuel oil storage and transfer system, (2) cooling water system, (3) air starting system, (4) lubrication oil system, and (5) combustion air intake and exhaust system. These systems are described and the results of our evaluation and findings are included in Sections 9.5.5 through 9.5.9.

9.5.5 Emergency Diesel Engine Fuel Oil Storage and Transfer System

The emergency diesel engine fuel oil storage and transfer system is designed to provide fuel oil storage and transfer capability to allow operation of each emergency diesel generator for at least seven days at rated load.

The fuel storage and transfer system consists of two separate and independent trains, one for each emergency diesel engine. The principal components for each train are: one underground 20,000-gallon main fuel oil storage tank; two motor driven full capacity final transfer pumps; one 1300-gallon day tank capable of operating the diesel engine for 16 hours at rated load; two fuel oil engine supply pumps, one integral engine driven and one motor driven; fuel oil strainer; fuel oil filter; required instrumentation, controls and alarms; and associated piping and valves required to connect the equipment. The fuel storage and transfer system is designed to non-seismic requirements in accordance with commercial industry standards and codes.

Based on our review of the fuel storage and transfer system design, design criteria and bases, we have concluded that the system has adequate capacity, and that it can perform its auxiliary support functions. At this time no system safety-related functions have been established; therefore, the system as installed is acceptable.

9.5.6 Emergency Diesel Engine Cooling Water System

The emergency diesel engine cooling water system is an integral part of the diesel generator and is designed to maintain the diesel engine jacket water and lubricating oil temperatures within safe operating range. The cooling water system is a closed loop pressure system dissipating heat to the atmosphere by a forced draft engine radiator during normal operation. When the engine is idle, an electric immersion heater and circulating pumps will maintain engine jacket water at the manufacturer's recommended temperature to increase first try starting reliability.

Each cooling water system consists of a jacket water radiator and a two-speed forced draft fan; two centrifugal circulating water pumps, one integral engine driven and one motor driven; one electric jacket water heater; one jacket water heater motor driven circulating pump; one 100-gallon pressure expansion tank; lube oil cooler; required instrumentation, controls and alarms; and associated piping and valves required to connect the equipment.

The emergency diesel engine cooling water system is designed to non-seismic requirements in accordance with commercial industry standards and codes. The emergency diesel generators are not required for safe reactor shutdown.

Based on our review of the emergency diesel engine cooling water system design, design criteria and bases, we have concluded that the system has adequate capacity,

and that it can perform its auxiliary support functions. At this time no system safety-related functions have been established; therefore, the system as installed is acceptable.

9.5.7 Emergency Diesel Engine Starting System

A starting air system is provided to supply the starting air needs for each diesel engine.

Each emergency diesel engine starting system provides for five 15-second starts without recharging. The system consists of two 100 percent capacity air compressors; two air receivers arranged so that either compressor can be operated with either or both receivers; strainers, lubricators, starting air motors; instrumentation, controls and alarms; and associated piping and valves to connect the equipment. The emergency diesel engine starting system is designed to non-seismic requirements in accordance with commercial industry standards and codes.

Based on our review of the emergency diesel engine starting system design, design criteria and bases, we have concluded that the system has adequate capacity, and that it can perform its auxiliary support functions. At this time no system safety-related functions have been established; therefore, the system as installed is acceptable.

9.5.8 Emergency Diesel Engine Lubrication System

Each emergency diesel engine is provided with an integral lubrication system designed to provide lubrication to bearings and other wearing parts. The lube oil is cooled by an oil cooler installed in the closed loop, integral, air cooled, diesel engine cooling water system. The lube oil is heated to improve first try starting reliability.

The emergency diesel engine lubrication system for each engine consists of two lubrication oil circulating pumps for normal operation, one integral engine driven and one motor driven; one electric lube oil heater; one motor driven electric oil heater circulating pump; one 300-gallon lube oil sump tank; strainless filters, instrumentation, controls and alarms; and associated piping and valves to connect the equipment. The emergency diesel engine lubrication system is designed to non-seismic requirements in accordance with commercial industrial standards and codes.

Based on our review of the emergency diesel engine lubrication system design, design criteria and bases, we have concluded that the system has adequate capacity, and that it can perform its auxiliary support functions. At this time no system safety-related functions have been established; therefore, the system as installed is acceptable.

9.5.9 Emergency Diesel Engine Combustion Air Intake and Exhaust System

The emergency diesel engine combustion air intake and exhaust system is designed to supply filtered fresh air for combustion to the emergency diesels and to dispose of the engine exhaust to the atmosphere.

The combustion air intake and exhaust system for each emergency diesel is separate and independent from the other and consists of air intake louvres, integral air filter and silencer on each diesel engine, exhaust silencer installed on the diesel generator room roof, expansion joint, and associated piping connecting the equipment.

The physical separation between the combustion air inlets and exhaust gas outlets for the diesel engines and the location of the building is such to preclude the possibility of fire extinguishing agents and other noxious gases from being drawn into the air intakes and exhaust gas recirculation with the air intakes.

The combustion air intake and exhaust system is designed to non-seismic requirements in accordance with commercial industrial standards and codes.

Based on our review of the combustion air intake and exhaust system design, design criteria and bases, we have concluded that the system has adequate capacity, and that it can perform its auxiliary support functions. At this tme no system safety-related functions have been established; therefore, the system as installed is acceptable.

10.0 CLOSED LOOP SYSTEM

The FFTF Project requested (letter from A. J. Rizzo to W. Gammill, dated June 13, 1978) postponement of the review of the closed loop system plant protection system because the design analysis has not been completed. In addition, closed loop system testing is not anticipated sooner than the seventh irradiation cycle.

Because the plant protection system impacts on the performance of the closed loop system under anticipated and hypothetical transient conditions and design modifications may result from the requirements of the plant protection system, we have postponed our review of Chapter 10.0 of the Final Safety Analysis Report, "Closed Loop System."

The closed loop system will be addressed in a supplement to this report.

11.0 RADIOACTIVE WASTE MANAGEMENT SYSTEMS

11.1 Summary Description

The radioactive waste management systems are designed to provide for the controlled handling and treatment of potentially radioactive liquid, gaseous, and solid wastes which are generated during normal operation of the FFTF, including anticipated operational occurrences. The waste management systems consist of the liquid waste system, the gaseous waste system, the solid waste system, and the process and effluent radiological monitoring system.

The liquid waste system collects wastes which are produced from lavatory decontamination sinks, component and area decontamination operations, and personnel decontamination showers. Liquid wastes are collected for transfer to the Hanford 200 Area for ultimate disposition.

Gaseous wastes are produced from the migration of radioactive gases from defective fuel to the primary system cover gas and from leakage and diffusion of radionuclides through primary and secondary system piping and components. Gaseous wastes are treated for radioactive material removal by decay, filtration, charcoal delay, and cryogenic distillation.

The solid waste system collects waste for packaging and interim storage prior to transfer to the Hanford 200 Area for ultimate disposition. Solid waste consists of air ventilation system filters, radioactive gas filters, sodium cold traps, contaminated tools and equipment, rags, and paper.

In our evaluation of the liquid waste system, we have considered: (1) the capability of the system to meet the collection demands of the FFTF during anticipated operational occurrences, (2) the design features incorporated to control releases of radioactive materials, and (3) the quality group and seismic design classification applied to the equipment and components and structures housing the system.

In our evaluation of the gaseous waste system, we have considered: (1) the capabilities of the system to keep levels of radioactivity in effluents "as low as is reasonably achievable," based on expected radwaste inputs over the life of the plant, (2) the capabilities of the system to meet the processing demands of the FFTF during anticipated operational occurrences, (3) the design features incorporated to control releases of radioactive materials, and (4) the quality group and seismic design classification applied to the equipment and components and structures housing this system.

In our evaluation of the solid waste system, we have considered: (1) the system design objectives in terms of expected types, volumes, and activities of waste processed for transit to the Hanford 200 Area for disposal, (2) the method of operation and the capability for meeting the demands of the FFTF due to anticipated operational occurrences, and (3) the provisions for temporary onsite storage prior to transfer to the Hanford 200 Area.

In our evaluation of the process and effluent radiological monitoring system, we have considered the system's capability (1) to monitor all normal and potential pathways for release of radioactive materials to the environment, (2) to control the release of radioactive materials to the environment, and (3) to monitor the performance of process equipment and detect radioactive material leakage between systems.

Based on our evaluation, as described below, we find the liquid, gaseous, and solid radioactive waste systems and associated process and effluent radiological monitoring systems to be acceptable.

11.2 Liquid Waste System

The liquid waste system consists of equipment and instrumentation designed to collect, store, and transfer liquid radioactive waste to the Hanford 200 Area for ultimate disposition. Wastes are collected based on their origin in the plant and the expected levels of radioactivity.

The liquid waste system consists of: four 55-gallon drainage drums for safety shower wastes; a 200-gallon collection tank for condensed waste from the cell atmosphere processing system; a 5000-gallon collection tank for wastes from floor drains, lavatory sinks, and personnel decontamination showers; a 20,000-gallon railroad tank car for transfer of low/intermediate level (< 1.0 uCi/cm^3) wastes; and a 500-gallon shielded transfer cask for transport of high level (> 1.0 uCi/cm^3) waste. All liquid wastes are transported to the Hanford 200 Area for ultimate disposition. Thus, there are no direct discharges to the environment of liquid waste from the Fast Flux Test Facility.

The liquid waste system includes measures to control the release of radioactive materials due to potential overflows from indoor and outdoor storage tanks. The 5000-gallon collection tank is located in a cell which is epoxy-coated on all walls. The cell is equipped with a sump pump which can direct the waste to the collection tank or to the railroad tank car. The 200-gallon collection tank is located in a cell which pumps to the 5000-gallon collection tank. The high level waste shipping cask is located in a cell which drains to the coated cell described above. The loading area for the railroad tank car contains two drains which lead to the 5000-gallon collection tank. The four 55-gallon drainage drums are located in a pit which is equipped with a sump. In addition, the 200-gallon and 5000-gallon

collection tanks are equipped with level instrumentation. We consider these provisions to be adequate for controlling the release of radioactive materials in liquid effluents to the environment.

The liquid radioactive waste treatment system will be located in the reactor service building which will be designed to seismic Category I criteria. We find the structure housing the system to be acceptable. The design capacities of principal components considered in the liquid radwaste evaluation are listed in Table 11.2-1. The two liquid waste collection tanks are designed to ASME Section III, Class 3 criteria. We find the Project's proposed liquid radioactive waste treatment system design to be acceptable.

TABLE 11.2-1

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED IN EVALUATION OF THE LIQUID AND GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEMS

Component	Number	Capacity Each
LIQUID SYSTEM*		
Collection Tank	1	200 gal
Collection Tank	1	5000 gal
GASEOUS SYSTEMS		
RADIOACTIVE ARGON PROCESSING SYSTEM*		
Vacuum Vessel	1	254 ft^3
Surge and Delay Tank	1	1000 ft^3
Charcoal Bed Adsorber	4	1300 lbs charcoal
Cryogenic Still	1	25 scfm
Storage Vessel	1	254 ft^3
CELL ATMOSPHERE PROCESSING SYSTEM*		
Vacuum Tank	1	334 ft^3
Surge and Delay Tank	1	742 ft^3
Charcoal Bed Adsorber	2	1300 lbs charcoal

*Quality group and seismic design in accordance with BTP ETSB 11-1 (Rev. 1).

11.3 Gaseous Waste System

The gaseous waste system consists of equipment and instrumentation designed to collect, process, store, monitor, recycle and dispose of radioactive gaseous wastes

generated during normal operation, including anticipated operational occurrences. The radionuclides associated with these wastes are principally noble gases. The gaseous waste system is comprised of two subsystems: (1) the radioactive argon processing system and (2) the cell atmosphere processing system.

The radioactive argon processing system processes the reactor and closed loop cover gas systems. The argon cover gas, contaminated with noble gases and small quantities of tritium, is collected in a 254-foot^3 vacuum vessel and transferred by a compressor to a 1000-foot^3 surge and delay tank. The surge and delay tank is equipped with baffles to permit decay of short-lived radionuclides. From the surge and delay tank, the gases are cryogenically cooled and treated in a series of four charcoal delay beds, each containing 1300 pounds of charcoal. After holdup of approximately four days for krypton and 260 days for xenon, the gases are directed to a cryogenic distillation column which preferentially condenses krypton and xenon while gaseous argon passes from the top of the column to the cover gas supply tank for reuse in the plant. The xenon and krypton concentrate in the bottom of the distillation column is periodically transferred to shielded compressed gas bottles and shipped to the Hanford 200 Area for ultimate disposition. Thus, there are no direct releases to the environment of radioactive material in gaseous effluents from the radioactive argon processing system.

The cell atmosphere processing system collects and processes the gaseous radioactivity that may leak or diffuse into the cells inerted with nitrogen which house the reactor, primary heat transfer system, primary pumps, and overflow vessel. The cell atmosphere processing system also collects and processes potentially contaminated air from cells housing the argon processing components and argon from various stations within the plant. The major input to the cell atmosphere processing system consists of nitrogen containing trace quantities of contaminated argon cover gas and tritium which diffuses through primary system piping and components. The nitrogen/air gas bleeds and purges from the cells are collected in a 334-foot^3 vacuum tank and transferred by a compressor to a 742-foot^3 surge and delay tank. From the surge and delay tank, the gases are passed through a dryer which condenses all tritiated water vapor. The dryer condensate is directed to the liquid radwaste system. The dried gases are processed in a series of two cryogenically cooled charcoal delay beds, each containing 1300 pounds of charcoal. Although the flow rate into the cell atmosphere processing system is variable, the flow rate through the charcoal delay beds is maintained at a constant 50 scfm by an automatically controlled variable-flow recirculation loop. The charcoal beds are operated at temperatures of -137°F to -150°F. After holdup of approximately 1.3 days for krypton and 84 days for xenon, the gases are discharged to the environment via the auxiliary equipment building east ventilation exhaust. Should the concentration of activity in the cell atmosphere processing system effluent exceed 5 x 10^{-3} uCi/cm^3, the cell atmosphere processing monitors alarm and the gases are automatically recycled back to the vacuum tank for reprocessing. Only trace quantities of radio-

activity are expected to be released from the cell atmosphere processing system to the environment.

Ventilation air from the reactor containment, heat transport system south building, reactor service building, closed loop ventilation exhaust, and pipeway and main dump heat exchanger ventilation exhaust will be released to the environment without treatment. The principal radionuclide in ventilation exhaust air will be tritium from the dump heat exchangers. Tritium will diffuse from the primary heat transport system to the secondary heat transport system and, thence, through the dump heat exchangers to the exhaust air. The FFTF Project estimates a tritium release of 0.088 Ci/day based on the diffusion cited above. We concur with that estimate.

We conclude that the gaseous radwaste treatment system for the FFTF is capable of maintaining releases of radioactive materials in effluents "as low as is reasonably achievable" in accordance with 10 CFR Part 50.34a.

The principal components in the radioactive argon and cell atmosphere processing systems are designed to ASME Section III, Class 3 criteria and are seismic Category I.

The design capacities of the principal components in the radioactive argon and cell atmosphere processing systems are listed in Table 11.2-1. We find the proposed gaseous radwaste management system to be acceptable. The radioactive argon and cell atmosphere systems will be located in the reactor service building which will be designed to seismic Category I criteria. We find the structures housing the systems to be acceptable.

11.4 Solid Waste System

The solid waste system consists of equipment and instrumentation designed to collect, package, and store radioactive solid wastes prior to transfer to the Hanford 200 Area for final disposition. Wastes are collected and packaged based on their origin in the plant and the expected levels of radioactivity.

The radioactive solid waste will consist of heating and ventilation system filters and charcoal adsorbers, radioactive gas filters, contaminated tools and hardware, cell filters, sodium cold traps, sodium samples, rags, paper, and clothing. As appropriate, the solid waste will be packaged in cardboard boxes, metal or fiber drums, wood or concrete boxes, fiberglass, metal or concrete containers, and a shipping cask. The FFTF Project estimates that approximately 11,000 feet^3 of radioactive solid waste will be transferred to the Hanford 200 Area annually for ultimate disposition. Based on anticipated maintenance activities, we concur with that estimate.

Temporary storage for packaged solid waste will be provided in the 400 Area of Building 4713-B and in the radioactive liquid waste pipeway (Area 242). The storage

facilities are in accordance with the recommendations of Branch Technical Position ETSB 11-3 and adequate for meeting the demands of the plant.

From these findings, we conclude that the solid waste system is acceptable.

11.5 Process and Effluent Radiological Monitoring System

The process and effluent radiological monitoring system is designed to provide information concerning radioactivity levels throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environs.

Table 11.5-1 provides the proposed locations of continuous monitors. Monitors on certain effluent release lines will automatically terminate discharges should radiation levels exceed a predetermined value. Systems which are not amenable to continuous monitoring, or for which detailed isotopic analyses are required, will be periodically sampled and analyzed in the plant laboratory.

We have reviewed the locations and types of effluent and process monitoring provided. Based on the plant design and on continuous monitoring locations and intermittent sampling locations, we have concluded that all normal and potential release pathways will be monitored. We have also determined that the sampling and monitoring provisions will be adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring plant processes which affect radioactivity releases. On this basis we consider the monitoring and sampling provisions to meet the requirements of General Design Criteria 60, 63 and 64.

TABLE 11.5-1

PROCESS AND EFFLUENT MONITORING LOCATIONS

Stream Monitored
Reactor Containment Building Ventilation Exhaust*
Reactor Service Building Exhaust
Reactor Service Building Unlined Cells Exhaust*
Heat Transport System Service Building South Main Exhaust
Argon Blower and Valve Cell Exhaust
Radioactive Argon Processing System
Cell Atmosphere Processing System*

*These monitors alarm and automatically terminate the release when the radiation level exceeds a predetermined value.

11.6 Evaluation Findings

In our review, we have evaluated releases of radioactive materials in gaseous effluents for normal operation including anticipated operational occurrences based on expected radwaste inputs over the life of the plant.

We conclude that the gaseous radwaste treatment system will reduce radioactive materials in effluents to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 50.34a and, therefore, is acceptable.

We have considered the capabilities of the radwaste systems to meet the anticipated demands of the plant due to anticipated operational occurrences and have concluded that the liquid, gaseous, and solid waste system capacities and design flexibilities are adequate to meet the anticipated needs of the plant.

We have reviewed the quality assurance provisions for the radwaste systems, the design codes used for system components, the seismic design applied to the design of the gaseous waste processing system, and the seismic design applied to the design of structures housing the radwaste systems. The design of the radwaste systems and structures housing these systems meet the acceptance criteria as set forth in Branch Technical Position ETSB 11-1 (Rev. 1).

We have reviewed the provisions incorporated in the design to control the releases of radioactive materials in liquids due to inadvertent tank overflows and conclude that the measures proposed are consistent with our acceptance criteria as set forth in Branch Technical Position ETSB 11-1 (Rev. 1).

Our review of the radiological process and effluent monitoring system included the provisions for sampling and monitoring all normal and potential effluent discharge paths in conformance with General Design Criterion 64, "Monitoring Radioactivity Releases," for providing automatic termination of effluent releases and assuring control over releases of radioactive materials in effluents in conformance with General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," and for sampling and monitoring plant waste process streams for process control in conformance with General Design Criterion 63, "Monitoring Fuel and Waste Storage." The review included piping and instrumentation diagrams and process flow diagrams for the liquid, gaseous and solid radwaste systems and ventilation systems, and the location of monitoring points relative to effluent release points. We conclude that the FFTF Project's radiological process and effluent monitoring systems are acceptable.

Based on the foregoing evaluation, we conclude that the proposed radwaste treatment and monitoring systems are acceptable. The basis for acceptance has been conformance of the Project designs, design criteria, and design bases for the radwaste

treatment and monitoring systems to the applicable regulations and guides referenced above, as well as to staff technical positions and industry standards.

12.0 RADIATION PROTECTION

12.1 General

The Project has provided a management commitment to assure that the FFTF will be designed, constructed, and operated in a manner consistent with Regulatory Guides 8.8 and 8.10. The FFTF Project Manager is responsible for the operation and safety of the plant in conformance with Hanford Engineering Development Laboratory (HEDL) policies to assure that radiation exposures are maintained as low as is reasonably achievable. The FFTF Plant Manager is kept informed on all pertinent health physics functions. Hanford Engineering Development Laboratory health physicists have had experience with liquid metal nuclear reactor facilities. They use this experience to provide preliminary and continuing reviews from construction through startup of systems intended to provide assurance that occupational radiation exposures are maintained as low as is reasonably achievable.

The objectives of the plant radiation protection design are to:

(1) Maintain individual doses and total man-rem doses to plant workers, including construction workers, and to members of the general public as low as is reasonably achievable; and

(2) Maintain individual doses within the limits of ERDA Manual Chapter 0524. Radiation exposure is minimized by identifying potential radiation hazards, by the use of shielding, by planning operations to minimize man-rem exposure, and by measuring and limiting time of external exposures or quantity of ingested internal radiation sources. By meeting the above design objectives, occupational radiation exposures at the FFTF will be maintained as low as is reasonably achievable.

In order to satisfy the design objectives and minimize radiation exposures, the following facility and equipment design considerations will be incorporated in the FFTF. In order to maintain low personnel doses, radioactive systems will be located in individual shielded cells. Remote handling devices, in conjunction with shielding, will be used to work on equipment in high radiation zones. In order to assure integrity of the shield walls, the Project conducted an extensive quality control and visual inspection program during construction of the facility. Operations involving high dose rates, high contamination levels, and high radiotoxicity materials will incorporate remote handling, valving, and instrument readout. Fluid transfer facilities will be designed for easy flushing and internal cleaning to prevent crud buildup. These design considerations conform with the guidelines of

Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low as is Reasonably Achievable," and are acceptable.

The Fast Flux Test Facility is designed for a 20-year operating life. At the time the FFTF is decommissioned, the Department of Energy will have authority over what procedures will be used for dismantling the plant. Based on experience available from the decommissioning of a number of small nuclear power plants the Project believes that any safety-related technological problems incurred during decommissioning of the FFTF will not differ significantly from those that may occur during reactor refueling and maintenance. The Fast Flux Test Facility is designed for safe decontamination and dismantling, and the Project believes that its decommissioning can be accomplished with complete safety.

Operating and maintenance personnel at the FFTF follow specific plans and procedures in order to assure that "as low as is reasonably achievable" goals are achieved in the operation of the plant. These are based on over 100 reactor-years of experience operating production and test reactors at the Hanford site. Personnel practice on cold mockups prior to doing maintenance on equipment located in high radiation zones. Health physics personnel record dose rates and air contamination in work areas before operations and maintenance personnel enter. Radiation work permits control all entries into radiation zones. During maintenance activities, personnel are monitored for exposure to radiation and contamination. Upon completion of major maintenance jobs, personnel radiation exposures are evaluated. The results are compared with exposures from previous jobs to determine if changes in plant procedures are needed. These practices conform with those recommended in Regulatory Guides 8.8 and 8.10 and are acceptable.

12.2 Radiation Sources

This section contains a description of the sources of contained and airborne radioactivity used as inputs for the dose assessment and for the shielding and ventilation designs. Also included are the assumptions made by the Project in arriving at quantitative values for these contained and airborne source terms.

The fast flux test reactor is the effective source of all radiation in the facility. The reactor cavity walls and walls of the surrounding cells will serve to reduce the dose rate in the personnel access areas around the reactor to less than the design levels. The sodium coolant will also be an extremely high radiation source during operation. The saturation activity of 24Na governs the design of all operating shields surrounding the primary sodium. After the decay of the sodium coolant following shutdown, corrosion products and fission products will become the primary sources of personnel exposure. The source terms contained in Sections 11 and 12 of the Final Safety Analysis Report are based on continuous operation at 400 MWt with an assumed 1 percent failed fuel and 100 percent vented fuel in the four

open test positions. Other parameters, as well as a complete description of source term development, are contained in Section 12.1 of the Final Safety Analysis Report.

The ventilation system will route air from areas of low potential contamination to areas of increasing potential airborne contamination. The radioactive effluent from the reactor and closed loop cover gas systems is collected by the radioactive argon processing system. The effluent from all cells and spaces subject to potential contamination by radioactive gases is collected by the cell atmosphere processing system. The basis for the control of radioactive gas leakage into occupied areas is based on one-tenth of the airborne radioactivity concentrations given in ERDA Manual Chapter 0524, Annex A, Table I.

12.3 Radiation Protection Design Features

The radiation protection design features at the FFTF are intended to help maintain occupational radiation exposures as low as is reasonably achievable. Many of these design features have been incorporated as a result of radiation exposure experience gained during the operation of other liquid metal nuclear reactors. In order to minimize radiation doses associated with maintenance work, all major pieces of process equipment will be located in individual rooms with individual access through labyrinthed entranceways from low radiation areas. Fluid transfer facilities will be designed for easy flushing and internal cleaning. Double barriers (e.g., pipe inside a pipe) and shielding, when necessary, will be used for transfer lines passing through normal work areas. Radiation exposure and radioactive contamination will be controlled during the transfer of process materials or maintenance of equipment. Rough surfaces, square corners, cracks or crevices, and absorbent materials shall be avoided in the design of hoods, cells, gloveboxes, and process equipment to prevent crud buildup. To facilitate decontamination and control of atmospheric purity, selected cells containing radioactive equipment will be lined with gas tight stainless steel liners and equipped with rinse systems. The hot cell complex is well shielded and will house remotely operated equipment for the performance of nondestructive examination of core components and maintenance of reactor plant equipment. Other design features incorporated at the FFTF and intended to complement the "as low as is reasonably achievable" program include: (1) use of emergency lighting designed to provide adequate illumination after loss of preferred and emergency power system supplies; (2) routing of all piping carrying radioactive material so as to minimize personnel exposures; and (3) clear identification of radiation hazards using warning lights and/or authorized signs. These design features conform with those recommended in Regulatory Guide 8.8 and are acceptable.

Areas within the restricted area will be divided into five radiation zones. These zones will be established as a result of radiation surveys made during startup and operation of the facility. The dose rate criterion for each of these zones is derived from the zone's expected occupancy and access restrictions. These criteria will then be used as the basis for the radiation shielding design. This allows for

arrangements of radioactive equipment that are in accordance with the guidelines of Regulatory Guide 8.8. Health physics personnel will routinely monitor and post all radiation zones within the containment building as described in ERDA Manual WHAN-M-8, "Radiation Protection Procedures."

The radiation shielding will be designed to provide protection against radiation for operating personnel, both inside and outside the plant, and for the general public. Shielding will also be used where it is necessary to protect materials and equipment from excessive radiation damage. The primary heat transport system components are located in individual cells to maximize the maintenance and operation flexibility while minimizing doses to personnel. The cells in the containment building are arranged so that the reactor and high activity cells are at the center and the lower radiation and source-free cells are located around the perimeter of the building. This assures low radiation levels outside of the containment structure. Design features to minimize streaming will include: (1) use of stepped access penetrations through the closure head; (2) routing of secondary sodium piping into the containment building through dogleg penetrations; (3) shielding of annular spaces around electrical cables and coolant channels which pass through plugs in the reactor shield; and (4) use of a boron carbide shield ring in the annular gap around the reactor vessel.

The basic radiation transport analysis used for the Project's shield design is based on such codes as ANISN, DOT, 2DBS, and GRACE. Monte Carlo calculations were used for shielding problems involving more complex geometries. The concrete shielding in the plant is in compliance with ACI Standard 318-71, "Building Code Requirements for Reinforced Concrete," and ACI Standard 301-66, "Standard Specifications for Structural Concrete for Buildings." We find the shielding design and methodology presented in the Final Safety Analysis Report and various related documents acceptable based on the criteria of the Standard Review Plan.

The ventilation system at the FFTF will be designed to protect personnel and equipment from extreme thermal environmental conditions and ensure that plant personnel and the general public are not inadvertently exposed to airborne contaminant concentrations exceeding those given in ERDA Manual Chapter 0524. The Project intends to maintain personnel exposures as low as is reasonably achievable by: (1) maintaining pressure differentials among the zone areas having varying degrees of contamination potential to control the spread of airborne contaminants; (2) being able to air purge normally inerted cells to facilitate personnel entry for inspection and maintenance in an air atmosphere; and (3) controlling the release of airborne radioactivity to the outside environs. These design criteria are in accordance with those recommended in Regulatory Guide 8.8 and are acceptable. The ventilation system at the FFTF is designed to isolate the containment in the event of an accident with virtually no radioactivity release. The radioactive argon processing system collects and processes all gaseous radioactive products from the reactor and closed loop cover gas system. The effluent from all other cells and spaces subject to

potential contamination by radioactive gases is processed through the cell atmosphere processing system. To prevent leakage of radioactive gases from the system, high integrity seals will be used in the FFTF whenever welded joints or seal welds can not be used. High-efficiency particulate air filters installed for use in design basis accident situations will be mounted in accordance with Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Power Plants." Air filtration in the control room is designed to limit radiation exposure to control room personnel in accordance with the guidelines of ERDA Manual Chapter 0524.

The primary purpose of the area radiation monitoring system at the FFTF is to provide the radiation status at all plant locations and indications of any changes in radiation conditions. To accomplish this objective, the Project plans to use thirty-one area monitors. The criteria used for installing these area monitors throughout the facility will be: (1) the frequency and duration of occupancy; (2) the radiation sources and their proximity to personnel; (3) the flow of traffic; (4) the probability of significant increases in radiation levels; and (5) the anticipated radiation levels under normal and abnormal conditions. The area radiation monitoring system is equipped with local and remote audio and visual alarms and a facility for central recording. Both the containment building and the reactor service building are provided with a set of three high gamma evacuation monitors. Each set is wired in a two-out-of-three logic network to sound an evacuation alarm upon detecting high gamma activity from the reactor primary system or from the fuel handling or fuel storage facilities.

The primary design objectives of the Project's airborne radioactivity monitoring system are to (1) measure and control quantities of radionuclides in plant systems and (2) measure and control radioactivity in effluents or potential effluents before and/or during their release to the environment. The Project will install fixed airborne radioactivity monitors at all exhaust ports which could potentially contain radioactive effluents resulting from system malfunctions or accident conditions. These monitors will continually monitor the gaseous effluent releases and will alarm at the detection of one maximum permissible concentration-hour for beta-gamma particulates as listed in ERDA Manual Chapter 0524, Table I. The Project will provide mobile continuous air monitors to monitor for radioactive gas and particulate concentrations in occupied work areas. For inert cells or other normally inaccessible areas where contamination may exist, the Project will use portable air samplers to test the concentration of airborne radioactivity prior to entry. In areas where the airborne contamination can not be controlled by the ventilation system, the Project will take appropriate measures to reduce airborne radioactivity. Such measures may include draining or purging of radioactive materials from equipment in that area and removal of surface contamination from equipment and area surfaces. The objectives and location criteria of the FFTF's area and airborne radiation monitoring systems are in conformance with 10 CFR Parts 20 and 50 and

Regulatory Guide 8.2 "Administrations Practices in Radiation Monitoring," and Regulatory Guide 8.8.

12.4 Dose Assessment

The Project has based the estimate of annual man-rem exposure at the FFTF on plant specific projections as to occupancy and dose rate and on experience from currently operating reactor facilities. Equipment and equipment layout will be designed to minimize time spent by employees in high radiation fields. All piping, valves, pumps and other equipment containing primary coolant are located in shielded pipeways or cells immediately surrounding the reactor. The shielding at the FFTF was conservatively designed and the actual anticipated dose rates within shielded cells are expected to be less than the design dose rates.

The Project has performed an assessment of the doses that will be received by plant operators and craftsmen. This dose assessment is based on occupancy factors, expected dose rates, and estimates of the time and manpower necessary to perform the various tasks involved in plant operation. This dose assessment includes a breakdown of personnel by craft into operators, electricians, instrument technicians, millwrights, pipe fitters, welders, and servicemen. The Project estimates the total annual man-rems to plant operators and craftsmen to be 132 man-rem. This estimate is based on experience in operating other reactors and on studies to determine the amount of time operators and craftsmen would spend in various cells of the FFTF to operate and maintain equipment. Currently operating light water reactors average between 400 and 500 man-rems per unit annually. However, since the FFTF is a liquid metal breeder reactor, the expected operating doses can be expected to differ from those of light water reactors. The FFTF differs from most light water reactors by the fact that operations such as fuel handling and equipment maintenance are carried out in well shielded cells with remotely operated equipment. This greatly reduces the personnel exposure associated with these operations. In addition, the fast test reactor, primary sodium piping, and associated valves and equipment are well shielded in cells having inert atmospheres. Access to primary piping components is restricted for sufficient time following shutdown to allow for sodium decay. We find the basis for the FFTF's exposure estimate acceptable and consistent with the acceptance criteria in the Standard Review Plan.

Airborne radioactivity levels in normally or routinely occupied areas where a potential exists for high airborne activity will be controlled by appropriate ventilation system design. The Project will use fixed and portable airborne activity monitors and respiratory protection equipment, when needed, during maintenance, inspection, and refueling. All buildings having a potential for radioactive airborne contamination will be equipped with airborne radioactivity monitors both upstream and downstream of the filtration systems. Radioactive gas leakage into inert atmospheres will be processed directly through the cell atmosphere processing system. Therefore, there will be no radiation exposure in the containment work areas due to leakage of noble gases from the reactor. Estimates of leakage

into the air atmosphere, as well as estimated internal dose estimates, are presented in Section 12.1 of the Final Safety Analysis Report. The Project's dose estimates for noble gases are based on comparability with the assumptions given in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," and are well within the exposure limits for noble gases contained in 10 CFR Part 50, Appendix I. The total estimated dose due to maximum expected gaseous leakage at the FFTF is several orders of magnitude below the dose limit for individuals in unrestricted areas designated in ERDA Manual Chapter 0524.

12.5 Health Physics Program

The objectives of the Project's health physics program are to provide reasonable assurance that the limits of ERDA Manual Chapter 0524 are not exceeded and to ensure that occupational radiation exposures to employees, visitors, and others from Hanford operations are maintained as low as is reasonably achievable. The Manager of FFTF Health and Safety is in charge of this program and he is responsible for keeping the Manager of HEDL Plant Safety informed at all times of radiation hazards and conditions related to potential exposure. Duties of the Manager of FFTF Health and Safety include establishment and direction of an adequate health and safety program, providing general safety and radiological control indoctrinations of all site personnel, detailed training of personnel involved with work in radiation areas, and review and approval of nuclear and criticality safety procedures for the FFTF. The objectives of the health physics program and the ways in which it will be implemented are in accordance with the guidelines of Regulatory Guides 8.8, 8.2, 8.10, and 1.8 and are acceptable.

The radiation protection facilities at the FFTF will include a counting room, personnel decontamination room, health physics offices, control points, laundry, radioactive work tool room, and a radiation management laboratory. Nearby Hanford facilities will be available for ultra-low-level radiological analyses, bioassay and environmental sample analyses, personnel dosimeter processing, and instrument calibration. These facilities are sufficient to maintain occupational radiation exposures as low as is reasonably achievable and are consistent with the provisions of Regulatory Guide 8.8.

Equipment to be used for radiation protection purposes will be current state-of-the-art equipment. It will include portable radiation survey instruments, personnel dosimetry instruments, fixed and portable area and airborne radioactivity monitors, laboratory equipment, air samplers, respiratory equipment, hand and shoe counters, and protective clothing. The number and types of equipment to be used are adequate and provide reasonable assurance that the Project will be able to maintain occupational radiation exposures as low as is reasonably achievable. All permanent and temporary plant personnel who do not work with radioactive materials and who do not

require entry into radiation zones will be assigned a Hanford basic thermoluminescent dosimeter (gamma measurement only). This thermoluminescent dosimeter will be exchanged annually. All personnel whose work involves routine entry into radiation zones will be assigned a Hanford multipurpose thermoluminescent dosimeter (beta, gamma, and neutron measurement). This will be exchanged monthly for all employees whose annual dose is expected to exceed one rem, and quarterly for all others. All personnel will be assigned pocket ionization chambers when entering the controlled area. The readings from these ionization chambers will be used to keep a running total of an individual's dose prior to thermoluminescent dosimeter processing.

Employees who work with radioactive material in a dispensible form receive regularly scheduled internal dose evaluations. Employees will receive in-vivo examinations with their routine employee medical examinations. In addition, a contracted laboratory will collect and perform radiochemical and autoradiographic analysis of excretory specimens from specified Hanford employees.

Maintenance, repair, surveillance; and refueling procedures and methods used by the Project are approved by the health physics section to assure as low as is reasonably achievable exposures in accordance with Regulatory Guide 8.8. Procedures are also developed to assure that exposure limits are not exceeded by plant or visitor personnel on site; to administer and control conditions of radiation work permits; to post radiation areas; to establish radiation access control zones; to control all radioactive material entering or leaving the plant site; and to train plant and visitor personnel in radiation protection policies and procedures.

Based on the information presented in the Final Safety Analysis Report and the Project's responses to our questions, we conclude that the Project intends to implement a radiation protection program that will maintain in-plant radiation exposures within the applicable limits of ERDA Manual Chapter 0524 and will maintain occupational radiation exposures as low as is reasonably achievable.

13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Qualifications

The Fast Flux Test Facility operational activities are conducted under the onsite supervision of the FFTF Plant Manager. He reports to the Vice-President, FFTF Project, who in turn reports to the President, Westinghouse Hanford Company.

The Plant Manager is responsible for the safe and reliable operation of the site. Reporting to the Plant Manager is a staff of approximately 252 full-time employees. This includes: An FFTF Operations Manager, responsible for the safe operation of the FFTF plant and supporting facilities, with a staff of approximately 130 persons; an FFTF Maintenance Manager, responsible for all plant preventive and corrective maintenance activities, with a staff of approximately 93 persons; an FFTF Planning and Scheduling Manager, responsible for coordination of activities and scheduling, with a staff of approximately 5 persons; an FFTF Plant Engineering Manager, responsible for providing engineering services to other plant groups, with a staff of approximately 13 persons; and an FFTF Health and Safety Manager (reports to the FFTF Plant Manager for technical direction only), responsible for health physics functions and industrial safety, with a staff of approximately 11 persons.

Reporting to the FFTF Operations Manager are the plant operating shifts. The shift crew for plant operation will consist of 16 persons. Two of these will hold qualifications equivalent to a licensed operator. In addition, each shift will include 3 interim examination and maintenance cell technicians, several trainees, a shift maintenance crew, and a radiation monitor technician.

We have reviewed the qualification requirements for plant personnel described in Section 13.1.3 of the Final Safety Analysis Report and find them acceptable since they meet the qualification requirements described in Revision 1 to Regulatory Guide 1.8, "Personnel Selection and Training," The Project has stated that they will submit to us general personnel qualifications prior to reactor startup.

Technical support for the plant staff will be provided by the technical staff of the Hanford Engineering Development Laboratory.

We conclude that the organizational structure and qualification requirements of plant personnel meet Revision 1 to Regulatory Guide 1.8, and are satisfactory to provide an acceptable operating staff. We further conclude that the Project has the necessary resources to provide technical support for the operation of the plant.

13.2 Emergency Planning

The Hanford Engineering Development Laboratory has submitted, as part of the Final Safety Analysis Report, an Emergency Plan for the Fast Flux Test Facility. We have reviewed this Emergency Plan, dated March 1978, and find that it includes all of the elements of emergency planning required of commercial nuclear power reactors as detailed in 10 CFR 50, Appendix E, Emergency Plans for Production and Utilization Facilities. The plan follows the format of Annex A to Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," generally conforms to the contents specified in the guide, and provides interfaces with those offsite agencies who may be called upon to provide emergency services in the event of an accident at the site.

The State of Washington, Department of Emergency Services prepared, as a part of their general disaster preparedness activities, a document entitled, "Washington State Emergency Plan for Fixed Nuclear Facility Incidents." This plan, dated May 1976, was reviewed and concurred in by the Commission's Office of State Programs. The plan satisfactorily conforms to the guidance found in the Commission's Guide and Checklist for Development and Evaluation of State and Local Government Radiological Emergency Response Plans in Support of Fixed Nuclear Facilities, NUREG-75/111. The Fast Flux Test Facilty is one of several nuclear facilities in the State of Washington which are identified in this plan, and for which contingency planning has been developed by the State to ensure that existing capabilities are effectively utilized to minimize the effects of nuclear incidents.

13.3 Review and Audit

The Hanford Engineering Development Laboratory has described their proposed provisions for the review and audit of plant operations. They include the Operations Review Committee that will provide a continuing review of plant operations and the independent Safeguards Council that will provide an independent review of plant operations. We have reviewed the provisions for the review of plant operations and find them acceptable since they meet Section 4 of ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." Audits of plant activities are described in Section 16 of the Final Safety Analysis Report.

14.0 INITIAL TEST PROGRAMS

14.1 Summary

We have reviewed the information provided in the Final Safety Analysis Report (through Supplement 25) pertaining to the initial test program. This review included an evaluation of:

(1) The Project organization and staffing for the development, conduct, and evaluation of the test program.

(2) The qualifications and experience of the principal participants managing and supervising the test program.

(3) The administrative controls that will govern the development, conduct, and evaluation of the test program.

(4) The degree of participation of the plant operating and technical staff in the test program.

(5) The Project's requirements pertaining to the trial use of plant operating and emergency procedures during the test program.

(6) The schedule for conducting the test program.

(7) The sequence of testing to be followed.

(8) The methods for conducting individual tests and the acceptance criteria to be used in evaluating the test results for plant structures, systems, and components.

(9) The test program's conformance with applicable Regulatory Guides including 1.41 (March 1973), "Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proposed Load Group Assignments," and 1.80 (June 1974), "Preoperational Testing of Instrument Air Systems."

We have concluded that with the exception of the confirmatory tests planned for demonstrating natural circulation capability that is noted below, the information provided in the Final Safety Analysis Report shows that an acceptable initial test program will be conducted in accordance with the acceptance criteria in Section 14.2 of the Standard Review Plan and with the regulatory

positions of the regulatory guides listed in Item 9 above, or that acceptable justification has been provided for exceptions to the regulatory positions in these guides.

14.2. Natural Circulation Verification Test Plan

In order to verify that the FFTF heat removal system will adequately remove decay heat from the reactor core in the absence of primary and/or secondary heat transfer system forced circulation, the Project proposed to conduct a series of tests demonstrating the effectiveness of the natural circulation decay heat removal option. We and our consultants have reviewed the proposed tests as described in "Interim Summary of FFTF Natural Circulation Test Plans" (September 1977), which was submitted for review. The following evaluation of the proposed test program is based primarily on our review of that document and on independent analyses performed by our consultants at Brookhaven National Laboratory and the University of California at Los Angeles.

The natural circulation verification test program, as proposed, consists of three separate tests. The first, HTS Secondary Loop Natural Circulation Test T-51-3A018, has as its purpose the determination of the natural circulation performance of the secondary heat transport system. All testing will be performed on the one tornado missile-hardened secondary heat transport system loop. The pumps in the two remaining secondary loops and in all three primary heat transport system loops will operate and will provide, in conjunction with loop trace heaters and dump heat exchanger pre-heaters, the required heat input to the system. No nuclear heat will be used in this test. Stated objectives of this test are (1) to determine steady state natural circulation flow in a secondary loop as a function of loop temperature differences, (2) to verify that dump heat exchanger modules can be isolated without undue perturbation under natural circulation conditions, (3) to demonstrate natural circulation capability at refueling conditions with low decay heat power, (4) to evaluate further dump heat exchanger heat loss characteristics, and (5) to obtain instrumentation and plant operation experience under natural circulation conditions.

The second test is designed to follow the first and has been designated Steady State Natural Circulation Test T-51-4A009. The purpose of this test is to evaluate the performance of both the primary and secondary heat transport loops using nuclear heat for various values of the core axial temperature difference up to the initial core design value of 258°F. Stated objectives of this test are (1) to perform a transition from forced to natural circulation in the reactor and primary heat transport loops, (2) to determine power removal capability from the reactor through the primary heat transport loops under natural circulation conditions as a function of the temperature difference across the reactor, (3) to demonstrate natural circulation operation of the reactor and of both the

primary and secondary heat transport loops at the maximum power level attainable within initial core design temperature limits, and (4) to determine the response to a step change in power under natural circulation conditions.

The last of the proposed tests is designated as Transient Natural Circulation Test T-51-5A008. This test will verify that the FFTF will satisfactorily undergo transition to the natural circulation heat removal mode following scrams from steady-state forced circulation operation at lower levels ranging from 5 percent to 100 percent of initial core power, with a loss of electric power to all primary and secondary coolant pump motors and to the dump heat exchanger fans.

For portions of the above tests, certain modifications to the plant protection system are required in order to allow plant operation. For example, the plant protection system must be modified to allow critical reactor operation under pony motor forced circulation and under natural circulation conditions. For all of the tests, sodium temperatures must be monitored in order that any onset of sodium freezing or boiling can be averted by operator action.

In general, the plant protection system modification and temperature monitoring aspects of the proposed verification test program appear to be acceptable.

The Project has indicated, however, that some changes in the individual test specifications have been made which have not been submitted for our review at this time, and that still additional changes are currently being processed. The Project has agreed that all project-approved natural circulation test specifications will be submitted for our review and that all additional changes to these approved test specifications will be submitted as they become available.

There are a number of recommendations which we have presented to the Project regarding the test program. These recommendations are now discussed in detail.

(1) In addition to verifying natural circulation capability under specific operating conditions, it is our opinion that a secondary objective of the proposed test program is to verify the thermal-hydraulic computer codes (e.g., IANUS and FLODISC) which have been used to predict both the plant natural circulation behavior as well as the plant thermal-hydraulic behavior under other flow conditions. Such verification is necessary since these computer codes are relied on to predict natural circulation phenomena or transient conditions other than those explicitly considered in the verification test program. The staff and the Project have not agreed specifically regarding the extent to which code verification is desirable. It is our opinion that these analytical models should be verified to the extent possible in order to demonstrate the degree of conservatism in the FFTF nominal design and safety margins.

In order to accomplish these objectives, we strongly recommed that the analytical pre-test predictions of the natural circulation tests include: (1) "best estimate" predictions using realistic models, methods, and parameter value assumptions and (2) predictions using the models, methods, and parameter value assumptions of Chapter 15 of the Final Safety Analysis Report, but based on the nominal conditions of the test (e.g., a decay heat uncertainty of 25 percent would be applied to the nominal decay heat expected in the core at the time of the test). If the comparison of test data to the pre-test predictions does not demonstrate that the models as used in the Final Safety Analysis Report are conservative, reassessment of the Chapter 15 analyses is recommended.

(2) A natural circulation transient can, depending on its severity, introduce thermal transients in various heat transport system components, potentially affecting adversely the design lifetime of those components. Most of the FFTF components appear to be sufficiently instrumented to adequately determine the thermal transient, if any, a given system transient has caused in a component. It is our recommendation, in order to verify that component stresses remain sufficiently small throughout the design lifetime, that the Project predict prior to the test the thermal transients expected to occur in association with the more severe of the proposed natural circulation tests in major system components such as the intermediate heat exchanger, the dump heat exchanger, and reactor vessel. These predictions can then be compared with the measured thermal transients in the various components to determine the adequacy of the design duty cycle for each major component. If the measured data indicate stresses in excess of those for which the equipment was designed, we recommend modification of the proposed component duty cycle to prevent premature component failure.

(3) The impacts of a tornado, although a low probability event, can include both the loss of integrity of the two non-tornado missile-hardened secondary heat transport loops, and the loss of power supplies for the main coolant pump motors and pony motors in all three primary and secondary heat transport loops and for the dump heat exchanger fans. Decay heat removal from this system configuration under natural circulation conditions will be by Phase I cooling from the hardened loop and Phase II cooling from the unhardened loops. (Note: Phase I cooling utilizes the dump heat exchangers as the heat sink in a given loop. Phase II cooling utilizes the primary system piping as the heat sink, with heat losses through primary loop pipe walls after the associated secondary loop has frozen or otherwise ceased to function. Phase III cooling, which occurs later in the cooldown transient, consists of reactor vessel cooldown after all primary loops contain solidified sodium.)

This combination of system configuration and cooling modes is not explicitly tested in the proposed natural circulation test program. In light of the minimal opportunity in the proposed test program for detailed thermal-hydraulic code verification during unequal loop flow conditions such as this and because the potential for such unequal low velocity flows in various portions of the reactor can lead to unexpected non-uniform core cooling, we recommended that the proposed natural circulation verification test program be extended to include a system test which approximates the post-tornado system configuration. The recommended test included a reactor scram from power operation concurrent with a total loss of power, simultaneous with the manual closing of the dump heat exchanger isolation valves associated with the two unhardened loops. The heat transport system behavior during this transient test would be pre-test predicted in both a best estimate and a conservative manner using the appropriate thermal-hydraulic computer codes as a further means of code verification. In response to our recommendation, the Project has proposed instead to modify the test program to include an "unequal loop flow" test and has indicated that the revised test specification will be submitted to us. We will review the test specification when it is submitted.

(4) Throughout the planned life of the facility, it is probable that changes in pump coastdown rates, heat transfer surfaces, and coolant flow paths (i.e., crud buildup and rod bowing) will occur. Since natural circulation performance can be affected strongly by relatively small changes in these thermal-hydraulic characteristics and in light of the general lack of industry experience with natural circulation phenomena in facilities of this size and type, we recommend that (1) performance criteria be set for such plant parameters as core pressure drop and pump coastdown rates and that these parameters be measured throughout the facility lifetime and (2) a single test sequence from the proposed natural circulation verification test program be performed periodically throughout the facility lifetime (e.g., every two years) in order to determine the effects of aging and wear on system natural circulation performance. It is specifically recommended that this test consist of a scram from steady-state forced circulation operation at as high a power as feasible, with a subsequent transition to natural circulation heat removal. Similar to the natural circulation verification test program to be performed at plant startup, the system transient associated with this periodic test should be pre-test predicted by appropriate thermal-hydraulic computer codes and the test should be performed only insofar as the design lifetimes of various heat transport system and core components are not adversely impacted by the associated thermal transient.

(5) The proposed natural circulation verification test program specifies that certain tests be performed only after successful completion of other tests

in the program. However, the criteria proposed to determine what constitutes a "successful test" do not appear to be closely related either to the determination of adequate natural circulation or to the requirements for thermal-hydraulic computer code verification. Accordingly, we recommend that the Project clarify performance criteria by which it may be judged whether a particular test has adequately demonstrated the predicted natural circulation performance and whether the system behavior to this point is sufficiently understood that the test program can proceed. Special emphasis should be placed on criteria for model acceptance.

(6) Test T-51-3A018 Sequences A and B and Test T-51-4A009 Sequences B and C appear to be concerned predominately with steady-state system behavior following a transient response to a thermal perturbation. Very little attention appears to be focused on the transient response itself, in particular the regions of flow transition. We recommend that this transient behavior be included in the computer code pre-test predictions, that performance criteria be specified and justified for various system components and regions (e.g., intermediate heat exchanger, dump heat exchanger, core region, etc.) to determine "successful" transient behavior as well as steady-state behavior, and that the transient response be compared for acceptability with these criteria on both a component and a system basis. It is our opinion that the inclusion of such information in the test program will improve the understanding of system behavior and computer model predictive ability. The Project has indicated that the test specifications are being modified to include predictions and measurements of the transient related phenomena for a limited number of transitions in the secondary loop. We will review these test specifications as they are submitted.

(7) Test T-51-5A008 includes a system natural circulation transient following reactor scram from steady-state power operation at 5 percent of initial core power. This transient is to be followed by similar transients after scrams from power operation at 75 percent and 100 percent initial core power. We recommend that an additional transient sequence be performed following a scram from a steady-state power level of approximately 25 percent of initial core power and 75 percent flow. This additional sequence should be performed prior to the 75 percent power level test. Alternatively, the proposed 5 percent power level test may be performed at a somewhat higher power level. It is our opinion that the gap between the 5 percent and 75 percent power levels is too large, given the relatively limited experience of industry with such system transients, to perform the tests as proposed.

It is our opinion that the proposed natural circulation verification test program, with the indicated recommendations, has reasonable assurance of determining the adequacy of the as-built FFTF to respond to thermal gradients in the primary and secondary heat transport loops in such a manner that natural circulation can be relied upon for decay heat removal in the event of the loss of all pump motive power. We will review all modifications to the verification program as they become available and will comment on them as necessary in supplements to this report.

15.0 ACCIDENT ANALYSIS

15.1 General

We have reviewed the capability of the FFTF to withstand normal and abnormal transients including postulated accidents of a severe nature such as major sodium spills and gross power/flow mismatches with first or second level protective system action. We find that 10 CFR Part 100 criteria are met under all circumstances except possibly for a most severe challenge to containment resulting from the very low likelihood of vessel melt-through in the event of a core meltdown accident.

We conclude that the intent of General Design Criteria 10, 11, 12, 13, 14 and 15 of 10 CFR Part 50 have been satisfied in their application to the FFTF.

The Project's transient safety analysis has been reviewed considering the first and second level trip functions of the plant protection system including the inherent time delays between the trip functions. For an overpower transient, the first trip occurs at 115 percent power and the second at 125 percent power. For neither plant protection system function is the 0.7 percent strain criteria used for the design of fuel cladding exceeded. The 0.7 percent strain criteria corresponds to an emergency event using the ASME classification of transients. The FFTF has followed accepted codes and practices for the reactor and heat transport system design. The plant protection system trip functions and trip levels are shown in Table 15.1.1-10 of the FFTF Final Safety Analysis Report.[1] The design thermal transients for the reactor and heat transport system are given in Tables 5A.1-3 and 5B.1-1, respectively. We conclude that the events described envelop all the anticipated modes of operation of the facility through the range of operating, normal, upset and emergency events. The core disruptive accident is evaluated as a faulted event for evaluating the structural response of the reactor coolant system boundary.

In addition we and the Project have evaluated the containment margins for core melt events assuming loss of in-vessel cooling following a core disruptive accident. The radiological consequences of this event have also been analyzed. We have concluded that the risks associated with a low probability reactor vessel melt-through are acceptably low assuming that a reasonable degree of containment integrity is maintained.

The criterion adopted to assure that the reactor coolant system boundary is not being challenged is in principle as follows:

(1) Engineer and construct a highly reliable vessel and coolant system.

(2) Maintain this system's integrity and reliability throughout its life.

(3) Prevent fuel clad failure and loss of coolable core geometry.

(4) Prevent boiling in the core by assuring adequate cooling, including natural convection.

(5) Construct the system such that any coolant loss will be minimized in the event of system leak, and that natural convection will be facilitated by the geometric placement of components.

(6) Provide a dependable control and shutdown system to assure that design power and temperature limits are not exceeded.

(7) Protect the heat transport system and reactor from the hazard of phenomenological events and fire by placing components in steel-lined inerted cells.

The FFTF Project addressed piping integrity in Appendix B of the Final Safety Analysis Report and does not consider an instantaneous double-ended rupture of a portion of the heat transport system a credible event. The staff has conditionally agreed that a cold leg primary system rupture is unlikely to be a core disruptive accident initiator but has not excluded the possibility of other smaller leaks and addresses this in Appendix E of this report. Leaks including double-ended pipe failure in the FFTF, other than in the cold leg inlet downcomer, do not have the potential to initiate a core disruptive accident but they can lead to sodium burning and cell pressurization. The consequences of these events are discussed in Section 6.2.1 of this report. The consequences of loss of decay heat removal are assessed in Section 15.3. of this report.

15.1.1 Reactivity Insertion Events

The Project has analyzed a spectrum of reactivity insertion events of different severity and probability classifications in order to demonstrate the adequacy of the primary and secondary scram systems to prevent or limit pin failure in the FFTF reactor. The particular events are listed below together with their classification and corresponding cladding strain limits (see Table 15.1-1 and 15.1.1-9 of Ref. 1 for definition of event classification and corresponding clad limits):

(1) Continuous C/R withdrawal at low power (unlikely event with 0.7% strain limit).

(2) Continuous C/R withdrawal at power (unlikely event with 0.7% strain limit).

(3) Single C/R assembly meltdown (up to extremely unlikely event with "Preserve Coolable Reactor Core Geometry" limit).

(4) Loss of Hydraulic Balance to fuel assemblies (up to extremely unlikely event with "Preserve Coolable Reactor Core Geometry" limit).

(5) Radial Displacement of Core Assemblies (unlikely event with 0.7% strain limit).

(6) Fuel Assembly Sodium Voiding (up to extremely unlikely event with "Preserve Coolable Reactor Core Geometry" limit).

(7) Single Fuel Assembly Meltdown (extremely unlikely event with "Preserve Coolable Reactor Core Geometry" limit).

(8) Closed Loop Test Section Meltdown (extremely unlikely event with "Preserve Coolable Reactor Core Geometry" limit).

(9) Cold Sodium Insertion (extremely unlikely event with "Preserve Coolable Reactor Core Geometry" limit).

For all the above events, the Project claims that the actual consequences, in terms of cladding strain, are much less than the limits for the various event classifications. In fact, it is claimed that there will be no additional cladding strain in the hot pin for any of the above events with the exception of the "Single C/R Assembly Meltdown" (max < 0.32% strain) and the "Loss of Hydraulic Balance to Fuel Assemblies" (max < 0.5% strain).

The most severe test of the FFTF plant protection system for reactivity insertion events is, in the Project's view, a "hypothetical" 3$/sec transient, which scopes all of the above mechanistic scenarios. Therefore the Project performs a detailed analysis of this terminated transient overpower event with a comparison to both terminated and unterminated transient overpower TREAT tests. For this extreme case, no pin failure is predicted.

All the transient overpower events (including the 3$/sec event) were analyzed using the MELT-III computer code. Transient cladding stresses and strains due to cladding/fuel differential thermal expansion and fission gas pressure were computed based on the results of the MELT-III calculations. Thus a calculated strain history was obtained for the events considered.

The Project verified the analytic model by comparing predicted fuel pin failure times with those observed in unterminated transient overpower TREAT tests. The model consistently predicted fuel pin failure times (for these TREAT tests) that are earlier than observed. Thus the Project concludes that the predicted time by which the plant protection system must operate is earlier than the actual time and

that therefore the model is conservative. This approach to calculating cladding strain and failure time and to thereby judge the adequacy of the plant protection system is part of the "design procedure" described in Ref. 2 which is evaluated in Section 4.2 of this report. (The qualifications indicated in the evaluation of this design procedure, as indicated in Section 4.2, are applicable here also). If the plant protection system responds adequately, then, (in terms of accumulated cladding strain) to a calculated transient analyzed by this method, it is the Project's position that the actual plant protection system will be effective (more effective than calculated) in keeping the cladding within the design strain limits.

It is important to determine the total time required to insert the rods and the reactivity worth of the rods. These values are summarized in Table 15.1.2-3 of Ref. 1. All Project analyses assume that the most reactive of the three primary rods is stuck, resulting in primary system worth (for two rods) of 3.2% $\Delta k/k$. Also, system drag loads are assumed that are in excess of those calculated, by adding the worst-case tolerances and using the worst-case friction coefficient. It was also assumed that the secondary shutdown system rods were in the beginning-of-cycle position when scram occurred. A stuck rod was not assumed in the case of the secondary system. A Project evaluation was carried out which showed that the fuel pin performance is relatively insensitive to the available worth of the scram system with one stuck rod during all anticipated or unlikely reactivity transients. However, according to the Project evaluation, for certain extremely unlikely events, the fuel pin integrity limit was exceeded for some beginning-of-life conditions.

The concerns regarding the design procedure, given in Section 4.2, are not repeated here. It is our opinion, after taking account of these concerns, that the plant protection system is adequate to protect the fuel pins from failure and to keep them within the design strain limits.

15.1.2 Heat Removal Reduction Events

The Project has analyzed transients that are initiated by heat removal reduction events in the heat transport system. These events have been classified into the following two categories:

(1) Reductions in primary flow, which cause an immediate rise in core temperature because of a reduction in heat removal from the core, and

(2) Reductions in secondary sodium flow or dump heat exchanger air flow, which cause a reduction in heat removal at the intermediate heat exchanger or dump heat exchanger, respectively, and affect the core only through a delayed increase in the core inlet temperature.

The most severe of the heat removal reduction events which may be caused by severe natural phenomena (earthquakes and tornados) have been discussed previously in Section 5.5 of this report.

The Project has classified each heat removal reduction event according to the expected frequency of occurrence of the event itself or of other events of higher probability whose consequences are enveloped by the event. The cladding temperature limits previously discussed in Section 4.2 of this report were used by the Project as criteria to determine whether the fuel performance limits are met in the course of the heat removal reduction transients. In the transient analysis, the Project required that one plant protection system (either - primary or secondary function) meet the limits applicable to the event classification. A second backup function, in the other plant protection system shutdown system, was required to meet the cladding temperature limits applicable to the next event classification which is less probable.

The analyses of the thermal-hydraulic behavior of the FFTF were performed by the Project using the IANUS[3] digital computer code, which describes the thermal-hydraulic behavior of a loop-type liquid metal-cooled reactor plant. In these analyses, the following transients were evaluated.

(1) Loss of offsite electrical power
(2) Loss of all offsite and emergency AC power
(3) Loss of electrical power to one primary system coolant pump
(4) Continuous flow reduction by failure of control equipment
(5) Mechanical failure of one primary system coolant pump
(6) Loss of electrical power to one secondary system coolant pump
(7) Mechanical failure of one secondary system coolant pump
(8) Loss of air flow to dump heat exchanger modules
(9) Freezing of sodium in secondary system coolant loop.

In the evaluation of cladding temperatures only the maximum cladding temperature was considered. Both initial and rated conditions were analyzed by the Project for each event. The more severe results for each event were reported in Ref. 1. In all cases except one, the rated conditions generated the more severe transients because the steady-state cladding temperatures were nearer the cladding temperature limits. The sole exception was the case of sodium freezing in a secondary coolant system loop in which the initial conditions gave more severe results due to the smaller margin to freezing. Each heat removal reduction event was analyzed for three-loop 400 MW operation and two-loop 267 MW operation as well as for other intermediate power levels. The results for three-loop 400 MW and two-loop 267 MW operation are shown in Table 15.1-1 although the staff's scope of review was limited to three-loop operation only.

Table 15.1-1 summarizes the predicted maximum hot channel cladding temperatures for each event. For all of the events analyzed, it was predicted that the action of the plant protection system would hold the hot channel cladding strain below the 0.7% strain limit for unlikely events (as defined in Section 4.2 of this report) and the 0.3% strain limit for anticipated events. In the case of those transients

TABLE 15.1-1

MAXIMUM HOT CHANNEL CLADDING TEMPERATURE[a] FOR EACH FLOW REDUCTION INCIDENT

INCIDENT	MAXIMUM HOT CHANNEL CLADDING TEMPERATURE[a] (°F) AND REACTOR SHUTDOWN SYSTEM			
	400 MW, 3 LOOP OPERATION		267 MW, 2 LOOP OPERATION	
	NOMINAL DATA AND FIRST PPS TRIP	WORST CASE DATA AND BACKUP PPS TRIP	NOMINAL DATA AND FIRST PPS TRIP	WORST CASE DATA AND BACKUP PPS TRIP
Loss of Offsite Electrical Power	1442 Secondary	1594[d] Primary	1406 Secondary	1513 Primary
Loss of All Offsite and Diesel Electric Power	1442[b] Secondary	1594[d] Primary	1406[b] Secondary	1513 Primary
Loss of Electrical Power to One Primary Pump	1477 Secondary	1494 Primary	1394 Secondary	1475 Primary
Continuous Sodium Flow Reduction	1588 Secondary	1622[d] (1605 Midwall) Primary	1538 Primary	1632 (1634 Midwall) Secondary
Mechanical Failure of One Primary Pump	1479 Primary	1576 Secondary	1474 Primary	1641 (1634 Midwall) Secondary
Loss of Electrical Power to One Secondary Pump[c]	1408 Secondary	1408 Primary	1385 Secondary	1385 Primary
Mechanical Failure of One Secondary Pump[c]	1408 Secondary	1408 Primary	1385 Secondary	1385 Primary
Loss of Air Flow in DHX Modules	1408 Secondary	1408 Primary	1385 Secondary	1385 Primary
Sodium Freezing	1408 Secondary	1408 Primary	1385 Secondary	1385 Primary

[a] Temperatures are calculated at the top of the active core at the cladding inner surface.
[b] For expected values of fission product decay heat, reactor pressure drop, and pump locked rotor pressure drop.
[c] Maximum temperatures occur at steady-state conditions.
[d] Maximum temperature for this case occurs at N80% power and 80% flow.

involving reduction in air flow through a dump heat exchanger module or in secondary system sodium flow, the predicted core temperatures did not rise above their steady-state values due to the system transport time delays and the rapid response of the plant protection system. For the transients involving a reduction in primary system sodium flow, it was predicted that the first plant protection system trip would limit the hot channel temperature so that a margin existed between the predicted maximum temperatures and 1580°F for anticipated events and 1600°F for all other events.

As can be seen from the table, if the action of the first shutdown system is neglected in the analyses, it is predicted that the second shutdown system would act so as to hold the maximum temperatures below 1600°F, except for the cases of continuous flow reduction and mechanical failure of one primary system coolant pump. In the case of a postulated instantaneous stoppage of the shaft of one primary system coolant pump while on two-loop 267 MW operation, the predicted maximum midwall cladding temperature would be above 1600°F if the first plant protection system trip was assumed inoperative. Because of the short time involved, the predicted cumulative cladding strain would be within acceptable limits. The case of continuous flow reduction is similar.

For the cases analyzed which led to the loss of all offsite and emergency AC power, resulting in cooling by natural circulation alone, a second peak in core temperature was predicted approximately two minutes after scram. Although the magnitude of this peak was found to be insensitive to which shutdown system tripped the reactor, it was found to be quite sensitive to the number of operating loops, initial power level, fission product decay heat generation rate, and reactor and pump pressure drops at low flows.

Assuming upper bound values of decay heat generation and pressure drops, the Project reported in Section 15.1.3.11 of the Final Safety Analysis Report that the maximum hot channel cladding temperature for the worst case during natural circulation conditions would be 1451°F. For most cases, substantially lower maximum temperatures were predicted. The results of the analysis indicated that, for expected or upper bound values of input parameters, no significant cladding strain would occur during natural circulation conditions.

We have reviewed the results of these analyses and find them acceptable because the violation of the fuel design limits is not predicted in any of these transients. However, these analyses were performed with an unverified thermal-hydraulic model (IANUS). We have previously reported our evaluation of IANUS in Sections 4.4 and 14.2 of this report. Thus, our acceptance of these results is conditioned upon satisfactory verification of the thermal-hydraulic model used in the analyses.

15.1.3 Local Fuel Failure Events

Introduction

The analysis of local fuel failure events and the propensity for these failure events to lead to failure propagation is the subject of Section 15.1.9 and Appendix C of the Final Safety Analysis Report.[1] In addition to analysis and experiments directed specifically at the FFTF as found in the Final Safety Analysis Report, there has been a large generic effort and Clinch River Breeder Reactor-related effort in the areas of local fuel element failure and failure propagation. Because of the many similarities between the FFTF and the Clinch River Breeder Reactor in terms of core design and operational characteristics, studies of fuel element failure propagation for the Clinch River Breeder Reactor are relevant and applicable in many respects to fuel element failure propagation in the FFTF. It is not surprising therefore that our basic conclusions regarding the FFTF are similar to those for the Clinch River Breeder Reactor, namely, that fuel element failure propagation is unlikely and that if it should occur it would be locally confined and have benign consequences.

It is important to point out the unique features of the active core for the FFTF relative to the Clinch River Breeder Reactor which may have a bearing on fuel element failure propagation. The major difference has been the presence of test loop subassemblies in the FFTF, in particular the two closed loop subassemblies in rows 2 and 6. The Project has requested, however, that we not consider the presence of the closed-loop assemblies at the present time. For the first year of FFTF operation these closed loops will not be in operation. Driver fuel assemblies or open test assemblies will be inserted in these two positions open to the upper and lower plenums. The other significant differences are: (1) the higher enrichment for the FFTF and higher specific power (about 12 percent higher); (2) lower burnup limits and cycle times; and (3) small pin and subassembly dimension differences. Although the above differences do alter boundary and initial conditions in the analysis of fuel element failure propagation for the FFTF, the ultimate effects of the differences are minimal.

Discussion

For the initial (first year) operation of the FFTF, operation with failed fuel is not intended. Assuming that the pin failure detection capability for the FFTF is adequate (see Section 9.2.4 of this report), the potential problems associated with operation with failed fuel or with slow-acting failure propagation are thus not germane to the present discussion. The emphasis here then is on the question of rapid fuel element failure propagation.

The fuel element failure propagation accidents are usually divided into four initiator categories: (1) stochastic fuel pin failures due to, for example, manufacturing

defects; (2) incorrect loading of fuel enrichments resulting in local overpower via incorrect pellets in pins, incorrect pins in subassemblies, or incorrect subassemblies in core loadings; (3) small blockages forming in core subassembly coolant channels; and (4) large planar blockages covering a major portion of the subassembly cross-sectional area. The concern for all these initiators is whether when a pin fails or blockage occurs there is a propensity for these conditions to lead to further pin failures and/or more aggravated blockages. The ultimate concern is whether or not these initiators could lead to subassembly-to-subassembly failure propagation and whole core involvement.

Extensive analytical and experimental work has been performed, and is continuing, to verify that conditions which might arise during plant operations are unlikely to cause pin-to-pin failure propagation. Aspects that have been covered include the effects of dimensional changes, wire wrap failure, fission gas release from pin failures, molten fuel release from pin failures, and other similar conditions which could lead to local flow disturbances or mechanical loadings. The results of this work thus far indicate that there should not be a significant potential for failure propagation beyond a few fuel pins at worst under the anticipated operating conditions and limitations. Experimental and analytical work has also been conducted on the effects of blockages within a pin bundle. The results thus far indicate that substantial blockages at the non-fuel inlet or outlet regions do not cause overheating, that inert planar blockages covering a few coolant subchannels in the fuel region do not cause any significant overheating, and that small heat-producing (fuel material) blockages do not cause significant overheating of adjacent areas. There is therefore a substantial basis to anticipate that local faults affecting single or a few pins within a subassembly will not rapidly propagate to adjacent pins.

The design approach being utilized in the coolant inlet region of the FFTF core should prevent large sudden flow blockage, such as that which led to extensive damage to two subassemblies in the Enrico Fermi Reactor.

Evaluation

From our review of the FFTF design, we have concluded that fuel pin failures which might occur under various plant operating conditions, including design transients, are unlikely to create conditions under which significant fuel failure propagation within a subassembly would occur. The general world experience with liquid metal fast breeder reactors supports this view.

15.1.4 Closed Loop System Safety Analysis

The review of the closed loop system safety analysis has been postponed. The Project requested (letter from A. J. Rizzo to W. P. Gammill, dated June 13, 1978) that the review of the closed loop system plant protection system be postponed

because the design analysis has not been completed. In addition, closed loop system testing is not anticipated sooner than the seventh irradiation cycle.

Because the plant protection system impacts on the performance of the closed loop in-reactor assembly under anticipated and transient conditions, we have postponed the review of Section 15.1.7 of the Final Safety Analysis Report. It will be treated in a supplement to this report.

15.2 Ex-Reactor Safety Analysis

15.2.1 General

Section 15.2 of the Final Safety Analysis Report presented the analysis of various accidents occurring outside of the reactor vessel which were evaluated for their impact on the public health and safety. The accidents which the staff considered necessary to review are events associated with spent fuel handling and storage, radioactive waste processing, and sodium fires. Our evaluation of these events is presented in the following sections.

15.2.2 Ex-Reactor Fuel Handling

The refueling process for the FFTF commences approximately 24 hours after reactor shutdown due to the many mechanical operations required before the pressure boundary can be opened and fuel transfer can physically be made. It was further determined that the earliest practical need to move assemblies from the core to the in-vessel storage module locations would be about 24 hours.

The maximum power of a driver assembly during reactor operation is 7 MWt. This would decay to 30 KWt within 24 hours after reactor shutdown. The in-vessel handling machine may move immersed driver assemblies at 40 KWt.

The closed loop ex-vessel machine is used to remove all core components and test assemblies from the reactor and can safely handle a 10 KWt driver assembly contained in a sodium-filled core component pot. Therefore, a maximum driver assembly can be removed from the in-vessel storage module into the closed loop ex-vessel machine about eight days after reactor shutdown.

The intermediate decay storage cell can handle a driver assembly that has decayed to 10 KWt or less which is contained in a sodium-filled core component pot. The interim examination and maintenance cell can handle three bare fuel assemblies with a maximum decay heat of 7 KWt for any element and 18 KWt for all three elements.

The bottom loaded transfer cask is designed to handle a driver assembly that has decayed to 1.4 KWt or less. The selection of a fuel element located in the in-vessel storage module and subsequent removal by the closed loop ex-vessel machine for either storage in intermediate decay storage or examination in the interim examina-

tion and maintenance cell is a physical process which involves an analog computer drive handling control scheme. The control sequencing and any override operations are performed under administrative procedures. Therefore, it is possible that a fuel element which had been operating at 7 MWt could accidentally be loaded into the closed loop ex-vessel machine from the in-vessel storage module shortly after the refueling process commences.

It is postulated that this "7 MWt fuel element" is accidentally loaded into the closed loop ex-vessel machine with a coincident loss of offsite power and a failure of its backup battery pack power supply shortly after the refueling process commences, resulting in a loss of cooling. It is assumed that the cladding on all the fuel pins fails within a short time and the gap activity (10% of the total inventory for the halogens and noble gases) is released into the containment with no credit given for holdup in the closed loop ex-vessel machine (this is a conservative assumption since the halogens would more than likely combine with the sodium and be retained in the core component pot). The containment is assumed to isolate before a fission product release occurs and to leak at the rate of 0.1% per day (this leak rate is conservative since it is the leak rate for the designed peak pressure of the containment). This accident scenario is considered to result in a radiological release in excess of any other postulated credible fuel handling accident. The calculated 0-2 hour thyroid and whole body doses were small fractions of 10 CFR Part 100 dose guidelines.

Based on our review, we conclude that timely isolation of the containment can be effected to limit a radiological release in a refueling accident to acceptable levels. Thus such an event poses no hazard to the health and safety of the public.

15.2.3 Ex-Reactor Fuel Storage

We have reviewed events associated with interim decay storage of spent fuel, including the dropping of a spent fuel subassembly and the extended loss of forced cooling, which have the potential to lead to radioactive releases. We have assessed these events and determined that their consequences are less limiting than are the consequences of events analyzed associated with the reactor itself or with the ex-reactor fuel handling. Thus we have not performed any further radiological analysis for these events.

15.2.4 Radioactive Waste Processing

We have reviewed the FFTF radioactive waste processing systems and have determined that the most limiting event associated with these systems is a failure of the radioactive argon reprocessing system.

The radioactive argon processing system pressure boundary is designed to satisfy Seismic Category I and American Society of Mechanical Engineers Section III Class 1

criteria and is located in a tornado-hardened area. The system components are located in a series of gas-tight cells and the cells housing the associated cold box and surge tank are steel lined. Both the cold box and the surge tank are isolated from each other and from the remainder of the system so that an event, plus a single failure, may release only the inventory existing in the cold box and the surge tank. Therefore, the total failure of the surge tank and the cold box is an extremely unlikely event.

Our calculations indicate that doses for failure of the surge tank and subsequent release of activity in the cold box over a two-day period would be a small fraction of 10 CFR Part 100 guideline doses providing the inventories of the cold box and the surge tank are of the same order of magnitude as those specified in Tables 15.2.14 and 15.2.15 of the Final Safety Analysis Report.

We recommend that appropriate technical specifications be placed on the maximum activity that can be stored in any one tank at any time such that single failure of an active component will not result in radiological consequences that exceed small fractions (F10%) of the 10 CFR Part 100 guideline dose.

15.2.5 Sodium Fires

Section 6.2.1 of this report discusses our evaluation of the effects of primary sodium fires inside containment. Based on our conclusion that the containment building design pressure is adequate, the radiological consequences of such events have not been analyzed further.

Section 6.2.2 of this report discusses our evaluation of the effects of a secondary sodium fire. Since the secondary sodium is non-radioactive, there are no dose calculations to be made. However, the effect of such an event on the habitability of the control room is a matter of staff concern. Our recommendations in this regard are presented in Section 6.4.3 of this report.

15.3 Analysis and Evaluation of Core Meltdown and Disassembly Accidents

15.3.1 Transient Overpower Assessment

Introduction

The transient overpower excursion analyses address unprotected power increase accidents and the consequences of these accidents in terms of the mechanical and thermal loading of the FFTF reactor vessel. Together with the complementary loss-of-flow accident analyses, these analyses are intended to scope all unprotected transients and therefore serve as a basis for the evaluation of the containment. The Project summarizes its transient overpower accident analysis in Appendix A, Section 4, (page A.4-15) of the Final Safety Analysis Report:[1]

"..... the energetics associated with both the base and parametric events considered are substantially smaller than the comparable values for which the containment system has been evaluated. Hence, a transient overpower accident with failure to scram in the FFTF is believed to pose no serious threat to the public safety."

This conclusion is consistent with the conclusion drawn by the Project in the recently published update of the unprotected transient overpower for the FFTF.[2]

With this present Project position in mind, it is instructive to review the evolution of the transient overpower assessment over the past eight years. As part of the original Preliminary Safety Analysis Report[3] a "conservative" transient overpower accident* was analyzed by the Project which yielded the 150 MJ work-energy value. Subsequently, in 1972, a new Hanford Engineering Development Laboratory analysis[4] indicated that the original Project analysis was overly conservative and that failures high in the core leading to fuel sweepout would dominate the accident progression thus leading to a benign event, that is, a permanent neutronic shutdown with the core maintaining an in-place coolable geometry. It is this basic scenario which has been the most prominent in the Project's analysis of the transient overpower for the FFTF since 1972.

We performed some parametrics on the Preliminary Safety Analysis Report transient overpower analysis and concluded that a more appropriate conservative value of work-energy resulting from a transient overpower was 300-500 MJ as compared to the Project's 150 MJ value. The results of this analysis are reported in our Safety Evaluation Report[5] with a tabular summary in Figure 12-1.

In Supplement No. 1 to the Safety Evaluation Report,[6] the matter of work-energy yields from the transient overpower was reassessed by the staff. It was determined then that the balance of information developed since the Safety Evaluation Report[5] appeared favorable, i.e., most evidence tended to reduce the concerns regarding transient overpower energetics.

Thus, although a conservative transient overpower accident was used in order to establish containment design margin, both the Project and the staff had concluded that an adequately conservative transient overpower for the FFTF would have a work-energy release considerably below the 150-350 MJ value used in determining containment design margin. The paragraphs below summarize our present assessment of the transient overpower accident for the FFTF.

*This accident assumed a 50¢/sec reactivity insertion and midplane failure of fuel pins. Positive reactivity feedback from fuel motion and sodium voiding resulted in a large energy dump into the fuel. It was assumed that this thermal energy was transferred to the sodium via an energetic fuel coolant interaction (FCI). The resulting isentropic expansion of the sodium to 1 atmosphere yielded the 150 MJ of work-energy.

Transient Overpower Accident Sequence for FFTF

The base-case scenario and important variations on that scenario for the FFTF transient overpower accident are quite similar to the base-case scenario and variations analyzed for the Clinch River Breeder Reactor. For the Clinch River Breeder Reactor there was an extensive independent analysis effort conducted by us and our consultants. These analyses are detailed in "An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Accident Energetics,"[7] and its accompanying references. For similar initial conditions (e.g., reactivity insertion ramp rates, fuel pin failure location assumptions) the FFTF transients tend to be more benign, principally because of the smaller region of positive void worth in the FFTF (whole core voiding in the FFTF is negative while for the Clinch River Breeder Reactor it is positive). Thus Ref. 7 serves as a general conservative guide for transient overpower behavior in the FFTF.

The basic transient overpower scenario is as follows: it is assumed that, as the power rises in the reactor, the coolant flow is maintained at steady-state. The increase in power causes the fuel to heat up providing negative Doppler feedback which keeps the net reactivity below the ramp input value and usually below prompt critical. During this time the sodium and cladding starts heating up. Pre-failure boiling occurs for low ramp rates in beginning of life cores. When sufficient energy has been generated by the transient, cladding failure occurs and solid and/or molten fuel may contact the sodium. Cladding can fail by one of two mechanisms: thermal loading (clad melt) or mechanical loading (fission gas pressure loading and/or fuel-pellet clad loading), the latter being much more common.

At the initiation of clad failure, the modeling of the failure dynamics involves four crucial aspects: (1) the degree of failure coherence, (2) the axial location of failure, (3) fuel motion at failure and (4) channel fuel and coolant dynamics. Fuel motion in the pins, fuel motion in the channel (that is, fuel being swept out of the active core, or fuel plugging up channels by freezing on cold steel or by debris pile up), and coolant voiding by fuel coolant interactions or fission gas release all may influence the subsequent evolution of the accident scenario along a broad spectrum of possible paths. On the one hand, the most energetic path is one in which fuel motion within the pin is toward the core midplane and coolant voiding is preferentially at the core center. This will most probably result in a mechanical disassembly of the core. On the other hand the most benign path is one in which a small amount of fuel moves away from the core center (high pin failures), fragments due to a limited fuel coolant interaction, and is hydraulically swept from the core. The cases in between these extremes involve a competition between positive and negative effects, including the effects of fuel freezing and plugging.

Summary of Project Analysis

A summary of the Project's analysis can be given with the aid of Figure A.4-6 of the Final Safety Analysis report[1] reproduced here for convenience (Figure 15.3-1). This figure indicates the MELT-III[8] prediction of the power and reactivity history for the base case (50¢/sec transient in the BOC-4 core). The following quote from pages A.4-4,5 explains the figure.

> "As noted from the power and reactivity balance plots, the calculation was continued out to 20 sec for the postulated reactivity insertion. During the first 2 to 3 sec of the transient, the temperatures began to rise and resulting Doppler feedback maintained the net reactivity at a level of around 60¢ above delayed critical. Initial failure occurred in channel 3 at an axial position 14 inches above the core midplane at about 2.7 sec, and the motion of the molten fuel is noted to cause a negative reactivity feedback and reduce the net reactivity of the system. Failure in Channels 5 and 11 follow shortly thereafter, resulting in a further reduction in the net reactivity and a continuing reduction in the reactor power level.
>
> For the time scale shown, explicit identification of channel failure is difficult to sort out, but the overall effect is to reduce the net reactivity well below criticality within a time scale of approximately 100 msec. Although the reactor is subcritical at 2.75 sec into the transient, the continued reactivity insertion eventually brings the net reactivity back up and the power level begins to climb.
>
> After the reactivity insertion mechanism is terminated at 8 seconds, with a total reactivity insertion of 4$, the core begins to cool down. This cooling process causes a positive Doppler effect to occur, but the new power/flow ratio reached is less than the steady-state value. Hence it is clear that long term cooling capability can be established.
>
> Details of the accident sequence are contained in HEDL-TME 75-50. A close inspection of the conditions existing in the failed pins reveals that the fuel-to-sodium ratio within the coolant channels is quite low, i.e., the propensity for hydraulic fuel sweepout appears high. Even for the base case calculations in which a very large amount of reactivity was postulated to be externally inserted, the total fuel loss from the core is only approximately 3% of the original inventory. Hence, long term cooling of an essentially intact core operating at a power/flow ratio below its initial value appears reasonably assured."

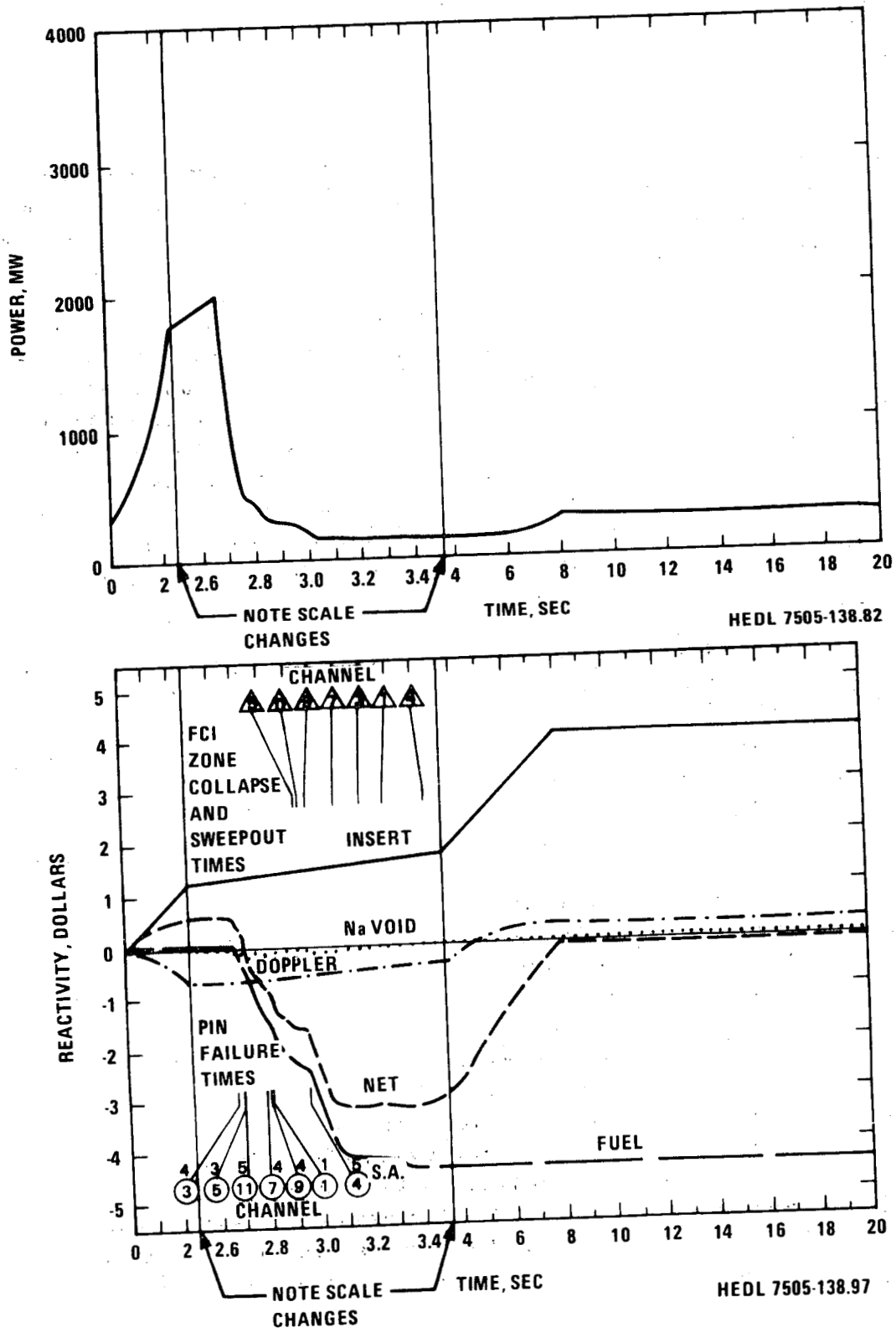


Figure 15.3-1 Overall Transient Response of 20-Channel BOC-4 Base Case (FSAR Figure A.4-6)

In addition to the base case assuming a beginning of cycle core, the base case is rerun in the Final Safety Analysis Report[1] for both an end-of-cycle core and a beginning-of-life core. The latter was assessed to behave differently. In particular, the beginning of life analysis concluded that sodium boiling and cladding dryout would provide the primary cladding failure mechanisms. As a result, the propensity for fuel freezing and blockage is enhanced. In spite of this tendency, the Project concludes that inherent incoherencies in the core together with predicted high failure locations assure long-term cooling capability.

Parametrics on both input conditions and phenomenology were performed for the base case. These included variations on initial power, inlet temperature, ramp rate, reactivity bound, Doppler coefficient, sodium void coefficient, fuel worth, fuel conditions, fuel failure, solid fuel motion and molten fuel motion. The conclusion of this study is that the general nature of the accident progression remains the same, thus indicating in-place coolable geometry for all cases.

The update of the above analysis[2] introduces some changes in the expected accident progession but, according to the Project, not in the basic conclusions regarding transient overpower. A modification of the damage parameter failure criterion (used in the Final Safety Analysis Report) referred to as the failure potential correlation yields predictions of pin failure approximately 500 msec later than damage parameter predictions for a 50¢/sec reactivity ramp. This allows for considerably more molten fuel to accumulate in the fuel pin and be ejected out into the coolant stream at failure, thereby increasing the propensity for fuel freezing and plugging from an increased local molten fuel/coolant ratio. Taking into account then the possibilities of freezing of fuel at the failure site, the Project proceeds to argue for long-term coolability based on arguments similar to those for the beginning of life case in the Final Safety Analysis Report,[1] namely, axially high failures coupled with strong intrasubassembly incoherence effects.

Staff Evaluation

As discussed in the Introduction, we and our technical assistance consultants (in this case consultants from Brookhaven National Laboratory, the University of California at Los Angeles, the University of Virginia and Pacific Northwest Laboratories) performed an extensive review of the transient overpower accident for the Clinch River Breeder Reactor including comprehensive independent analyses. Because of the similarities between the CRBR and the FFTF, the work performed earlier for the CRBR[7] provided some of the bases on which to assess the adequacy of the Project's evaluation of the transient overpower for the FFTF. It is our conclusion that a representative spectrum of transient overpower accidents results in "work-energy" consequences which are considerably less than the 350 MJ work-energy used to evaluate the containment design margin. This is consistent with the Project's view. However, we believe that there are sufficient uncertainties in the data, data interpretation and modeling associated with the transient overpower

analysis to preclude any firm conclusion at this time regarding the question of in-place coolable geometry following a transient overpower. The description of these uncertainties can be facilitated by dividing them into three subject categories: (1) low ramp rate transient overpower accident analysis; (2) the fuel-pin failure criterion used by the Project; and (3) the treatment of fuel freezing and plugging.

(1) In Appendix A, Section 2 of the Final Safety Analysis Report there is a discussion of the transient overpower accident initiators. An example is the uncontrolled withdrawal of the maximum-worth control rod which produces a ramp rate of about 3¢/sec and total worth of 5$. The base case for the transient overpower accident analysis was chosen to be a 50¢/sec ramp which is considered "conservative" by the Project because this rate of addition is about an order of magnitude greater than could be mechanistically envisioned due to control rod movement. Choosing an upper-limit ramp rate as "conservative" is consistent with past practice and is still considered a conservative approach by the staff. However, there has been considerable interest in low ramp rates, of the order of 1¢/sec. The accident progression for low ramp rates may be significantly different than that for 50¢/sec due to the possible presence of such phenomena as sodium-boiling before pin failure, intrapin fuel motion before pin failure, fuel swelling, and communication of transient-released fission gas with the fission-gas plenum. In addition to different phenomena coming into play there are problems with the ability of the present-day family of codes (SAS,[9] MELT-III,[8] HOPE[10]) to carry out low-ramp rate calculations over the full rod worth (for example, for FFTF, 5$ implies 500 sec).

    Although the dynamics of a low ramp rate accident sequence may be quite different than those of a high ramp rate sequence, there is presently no indication that it will lead to a mechanical disassembly; at worst there would be a greater tendency for core meltdown which should be adequately scoped by the loss-of-flow recriticality analyses. (See Section 15.3.3)

(2) The major new ingredient in the transient overpower update[2] by the Project was the replacement of the "Damage Parameter" fuel-pin failure correlation which was used in the Project analysis with the "Failure Potential" failure correlation. The reason for developing a new empirical model was that the damage parameter correlation (also empirical) failed to predict failure times for a number of tests involving lower-power microstructure fuel pins: HUT 3-5A, HUT 3-5B, HUT 5-5A and HUT 5-5B transient tested in TREAT. Thus the failure potential empirical model was developed and correlated to 12 TREAT tests including those above by adjusting four arbitrary parameters using linear regression analysis. As discussed in the Summary of Project Analysis section this new failure correlation led to later (than the Project analysis using damage parameter) failure times, thus yielding more molten fuel with an attendant increased

probability for plugging. Because of the tendency to fail later, it is more important to ensure that the pin failures occur high in the core. (If they failed close to the midplane there would be more molten fuel moving in the pin to the failure site thus adding reactivity and there would be tendencies to freeze similar to high failure events.) We believe that there will be a general tendency for pins to fail high in the core. (This opinion is independent of the results of this HEDL analysis.) However, due to the uncertainties associated with the statement that pins fail high in the core and specifically uncertainties associated with the failure potential correlation itself, it is important to list two relevant areas of concern:

(A) Since the failure potential correlation is empirical, the data on which it is based should be as prototypic as possible of the real reactor situation; that is, there should be only a very limited extrapolation of the correlation beyond the range of the original data base. This does not appear to be the case, due to the number of non-prototypic features of the EBR-II irradiations and TREAT transient testing. Among these are:

(a) no fully-developed and axially-varying restructuring in the 13 inch EBR-II pins used in TREAT
(b) irradiation at different linear powers than typical of the FFTF
(c) different fluence/burnup ratios in EBR-II compared to the FFTF
(d) no healing of cooldown cracks before TREAT transient testing
(e) fluence axial-dependence is not prototypic
(f) because of the soft TREAT neutron spectrum, radial temperature profiles are not prototypic.

A number of these non-prototypic features are used by the Project to explain why the MELT-III predicted cladding temperatures at failure for the FFTF (predicted using the failure potential correlation) fall completely outside the range of values found in the TREAT data base (note Figure 3.3 of Ref. 2). Regardless of the reason for this mismatch, the MELT-III analysis does seem to indicate that a considerable extrapolation of the correlation is necessary. Whether within the context of this particular mismatch or considering the non-prototypic features by themselves, these features make interpretations for the FFTF transient overpower behavior difficult.

(B) The second conclusion drawn by the Project (p 3-40 of Ref. 2) upon completing a statistical analysis of the failure potential correlation, namely, that "fuel pin failure near the axial center-line in the case of reactivity insertion ramps less than 1$/sec is essentially precluded," appears to be based on a statistical variation about a best-estimate failure location predicted by the failure potential correlation. Since this best estimate is determined from non-prototypic data, it is not clear that there is

sufficient confidence in the mean value to state that center-line failures have been "essentially precluded." (We agree however that fuel pin failures are not totally deterministic but will be partially stochastic with variations in time of failure and location of failure about the best estimate values. Thus the "popcorn popper" idea of incoherence appears to be appropriate (p 3-39 of Ref. 2)).

In conclusion, our position regarding transient overpower pin failure for the FFTF as predicted by the failure potential correlation is that predicted trends are correct and key ingredients accounted for. However, the application of the failure potential correlation to the FFTF means a considerable extrapolation from the EBR-II TREAT data base. Thus it is our opinion that the uncertainty in the best estimate failure location (as well as failure time) is larger than proposed and that failures near the center line are not precluded. This does not mean that we believe that the failures will dominate at the core midplane. It does mean that there is less certainty regarding the negative reactivity effects of fuel motion. Thus the possibility of added pin failures and more local fuel freezing increases. This is discussed further under (3) below.

(3) We agree with the Project that there is a real possibility that fuel will freeze and plug once it is injected into the coolant stream and that this possibility increases with the amount of molten fuel ejected. We also agree with the Project that there will be a certain amount of incoherence in failures (and therefore in initial freezing locations) across the subassembly due to the overcooling of the peripheral pins. Because of this incoherence and of stochastic failure effects, we believe that the total subassembly blockage by fuel freezing and plugging will not occur. However, partial plugging in the core may very well occur thereby presenting the question of coolability. Thus, our concern lies with the conclusion drawn by the Project that incoherence characteristics and partial sweepout together will assure long-term coolability. Our concerns in this area arise because of the following:

(A) Most of the TREAT transient overpower tests resulted in significant if not total plugging. These tests include the H-6 test which appears, from the preliminary data,[11] to have completely plugged. This test was to have been run under prototypic conditions of sodium flow, pressure drop, and power history (no overpowering) and thus was to have met the requirements to obtain valid post-failure data for a simulated FFTF environment. (Previous tests have been criticized for not being sufficiently prototypic.) Modification to the MARK-II loop met the first two conditions but evidently the test was overpowered (relative to an anticipated FFTF transient overpower power trace). Thus it is difficult to interpret the plugged condition in the H-6 TREAT test and its relevance for post-failure behavior in the FFTF.

In spite of the tendency to plug in the TREAT transient overpower tests, there is evidence of some sweepout of small fuel particles. In H-6 before the plugging took place, there is evidence that some sweepout of fuel from the initial failure occurred. Moreover, the R-12 test[12] indicated considerable sweepout. This test with fresh fuel pins, was successful in applying the "proper" power history (i.e., the anticipated FFTF power trace which drops off significantly after the first pin failure) to the test.

(B) The out-of-pile (CAMEL-2, HUMP 7 pin) tests[13] indicate sweepout for simulated transient overpower events, but the characterization of the injected material (temperature, void fraction, diaphragm behavior) and rate of injection involve uncertainties; moreover the degree to which the injected material characteristics, injection rates, and injected masses simulate prototypic transient overpower phenomena is not resolved at this time.

(C) The Project analysis does not consider the effect of clad melting of pins adjacent to failed pins and the subsequent tendency for failure propagation.

(D) It is not clear what the effect will be in the FFTF of possible large, delayed pressure pulses from fuel coolant interactions during the transient overpower sequence. Pressure pulses have been observed in TREAT (H-6 with 130 atmospheres[11]) and in out-of-pile tests. The observed pressure pulses have very little energy associated with them (i.e., the integral of the pressure pulse is low). If they remain small and localized in an FFTF environment, they could aid dispersing and sweepout.

In conclusion, we agree with the Project that the pin failures will tend to be above the reactor midplane, and consequently initial fuel motion in the fuel pin will tend to add negative reactivity and contribute to reducing the power. Certainly there will be an incoherence in failure time and position due to the stochastic nature of the failure events, the intrasubassembly incoherence due to overcooling of the peripheral pins and incoherence due to intersubassembly effects. We believe however that these ameliorating effects are not sufficient to ensure long term coolability after transient overpower events. Thus the possibility for entering a meltdown phase can not be eliminated.

Conclusion

Although we believe that there are significant uncertainties regarding the analyses and interpretation of experiments as presented by the Project, we basically agree with the Project conclusion, that transient overpower accident consequences are considerably less than the work-energy used to evaluate the reactor coolant system and containment design.

15.3.2 Loss-of-Flow Accident Assessment

Introduction

This review constitutes an updated assessment of an unprotected loss-of-flow accident for the FFTF. The earliest staff evaluation was documented in our Safety Evaluation Report of October 1972[1] and was based on information available in the Preliminary Safety Analysis Report[2] and information provided by the Project during the construction review stage.

Basically the Project's conclusion in the Preliminary Safety Analysis Report was that the reactor would eventually be shut down by the negative feedback from total sodium voiding, and that the accident would not have generated enough energy to disassemble the reactor. During the period of the Preliminary Safety Analysis Report review, Argonne National Laboratory (ANL) performed some additional analysis on the FFTF loss-of-flow accident[3] utilizing the ANL developed code SAS2A.[4] This analysis indicated that during the loss-of-flow accident sequence, subassembly slumping leads to reactivity insertion rates in the range of 15$/sec which would then drive some of the molten fuel into the sodium region near the top of the fuel subassemblies resulting in violent fuel/coolant interaction. The dynamics of this interaction process were evaluated and showed to be well within the structural design basis number of 150 MWsec. It should be pointed out that both the PSAR and the subsequent ANL analyses failed to carry the accident scenario to the point where the core would be in a permanently subcritical state thus providing the initial conditions for final disposition of the core material (i.e., in place cooling of the post-accident core debris or eventual penetration of the reactor vessel). We finally concluded in the Safety Evaluation Report that because of factors such as uncertainties in the phenomena involved and lack of experimental data, "it is prudent to retain substantial conservatism in the evaluation of postulated loss-of-flow accidents in the FFTF". Because of our continuing concerns as well as those of the ACRS regarding the evolution of the accident scenario following loss-of-flow accident in the FFTF, especially the potential for recriticality events and their magnitude, ANL pursued the analysis of the loss-of-flow accident for the FFTF and documented the results in two reports which covered both the initiating and core disruption phases.[5,6] From these studies ANL concluded that, for a liquid metal fast breeder reactor the size of the FFTF, the loss-of-flow sequence will involve a relatively gradual meltdown of the core during which mild nuclear excursions might occur. These excursions should be quite mild (< 20$/sec) and should not lead to high local pressurizations. It was also concluded then that, because of incomplete understanding of the dynamics of the transition phase (i.e., the phase that begins at the part of the analysis where subassembly integrity is starting to be lost; see Section 15.3.3 for our assessment of this phase of the accident), more energetic excursions in the range of 20 to 50$/sec cannot be precluded. These developments were taken into account in a supplement[7] to the Safety Evaluation Report where we concluded that even through it appears that maximum ramp rates

during core disruptive accidents in the FFTF may well be less than 100$/sec, the research and development effort should continue to reliably establish the accident scenario and its dynamics following the loss of subassembly integrity. The staff's present assessment of the initiating phase of a loss-of-flow accident in the FFTF follows next.

Present Evaluation

The present updating includes: consideration of new experimental data; the use of an improved version of SAS, i.e., SAS3A[8] in performing a limited number of parametric runs (intended mainly as sensitivity analysis on the parameters which are believed to be most important in the evolution of the transient); the latest information supplied by the Project;[9] and the staff work on the initiating phase of loss-of-flow accident for the Clinch River Breeder Reactor.[10]

The objectives of this evaluation are twofold; first, to assess the possibility of a loss-of-flow accident evolving directly into a hydrodynamic disassembly (and if so at what rate?) and second, to identify the core conditions at the end of the initiating phase which can be used as initial conditions in further assessment of the accident scenario, i.e., transition phase and potential for recriticality.

This review indicates that direct hydrodynamic disassembly is unlikely. At the end of the initiating phase, the core is expected to be neutronically shutdown and at a small fraction of the normal power. Positive reactivity insertion due to fuel slumping of the high power subassemblies aided by small amounts due to clad motion result in a benign excursion. Some fuel dispersal is expected to take place in the high power subassemblies due to fuel vapor pressure which is estimated to reach a few atmospheres. Clad motion in high power subassemblies will result in thin partial blockages at the top of the core and thicker ones at the bottom.

The anticipated sequence of events for the FFTF loss-of-flow accident includes: subassembly voiding, clad motion and plugging, fuel motion in voided subassemblies which results in a benign power transient and fuel dispersal in high power subassemblies, coolant voiding in low or intermediate power subassemblies and neutronic reactor shutdown. The major reactivity contributions come from coolant voiding, the Doppler effect, fuel axial expansion, clad motion and fuel motion. This scenario has been determined via SAS3A parametric calculations supplemented with a phenomenological understanding of the processes involved. After initiation of coastdown, the coolant temperature will rise and boiling commences in the central pins of the high power subassemblies. In this interval (which is of the order of 12-15 sec), the power level falls to about .75-.80 of the normal power level. The representation of coolant voiding in SAS3A is with a one dimensional multibubble slug model, where flow reversal occurs about .6 sec after bubble formation. In the subassembly, there is a temperature difference between the central and the outer pin rows of the

order of a few hundred degrees Centigrade; the staff with its consultants investigated the two dimensional aspects of voiding which arise from these temperature differences.[11]

This work indicated that the bubble will grow in a radial and axial direction and effective flow reversal will occur about 1.5 sec later. Some experimental verification of these predictions exists.[12] The effect of this delay on the loss-of-flow scenario for the Clinch River Breeder Reactor has been investigated[13] and does not seem to alter the basic event sequence. The phenomenology is equally applicable to the FFTF, i.e., the elongation of time to flow reversal does not affect the event sequence because, from the onset of boiling until clad motion initiation, the voiding reactivity feedback dominates the transient.

Following subassembly voiding and clad dryout the power level increases to about 90% of the normal power with fuel axial expansion and Doppler negative reactivities counteracting the positive reactivity of the high power subassembly voiding. Clad melting follows within one or two seconds and upward clad motion results due to coolant vapor streaming. The clad melting pattern will resemble the voiding pattern hence incoherent clad motion is to be expected. The modeling in SAS3A via the CLAZAS subroutine[14] does not account for this incoherency. This problem has been investigated experimentally with simulant materials as well as analytically.[15] The equivalent (as far as reactivity feedback is concerned for one dimensional clad relocation) velocity was calculated to be in the neighborhood of 80 cm/sec. A small positive reactivity insertion and a thin partial upper plug will result from the expected clad motion. Plug information will cause a decrease in vapor streaming and a reversal of cladding flow with a thicker plug expected to be formed in the bottom of the core. The FFTF Project based on simulant experiments[16] and quantitative considerations[17] concluded that there will be little or no upward motion of the molten cladding.

After a portion of the cladding has been removed, the fuel temperature will rise rapidly and fuel melting will take place in the central part of the pin progressing radially outward. As the temperature approaches the melting point, fission gases will be released. Of the available fuel-behavior TREAT tests, the most relevant is the F1 experiment in which the fuel underwent massive swelling and then slumped under gravity.[18] From the F1 and more recent work[19] we concluded that no fuel dispersal can be expected due to fission gas. Slumping of fuel or a fuel/steel mix results in significant reactivity feedback which makes the core supercritical to just below prompt critical. The power rises to about 50-70 times the normal power which raises fuel vapor pressure to few atmospheres (in few high power subassemblies). This disperses a small amount of fuel and renders the reactor subcritical. After the initiation of fuel melting the negative feedback due to axial expansion in that particular subassembly is discontinued. However from the initiation of fuel heating to the initiation of melting, we accounted for only 25 percent of the fuel axial expansion reactivity feedback (based on a volumetrically averaged

temperature). This is based on considerations of fuel swelling, cracking, and creep deformation. Some experimental evidence with in-pile measurements in support of the less than free expansion has been given by Cantley.[20] The scenario outlined above is roughly in agreement with the latest evaluation given by the Project.[8] The important conclusion is that the general outline of the scenario does not change with the variation of the parameters within the uncertainty range. Some of our consultants believe that even mild excursions will not take place.[21] The power excursion which results from fuel slumping is not autocatalytic in nature. Indeed another potential mechanism for negative reactivity insertion exists, i.e., voiding of the low power subassemblies which lie in the three outer rings and which have an overall negative voiding reactivity. This can be accomplished either by sodium boiling if the power continues around the normal level or by pin failures if the excursion is of the higher severity than what has been described above. In summary:

- The unprotected loss-of-flow transient in the FFTF is not anticipated to lead into direct energetic hydrodynamic disassembly, and

- At the end of the initiating phase the core is expected to be neutronically shut down due to limited fuel dispersal in the high power subassemblies (and/or voiding in the low power peripheral subassemblies). A few subassemblies are expected to be slumped with partial thin upper blockages and lower thick blockages.

The development of the core phenomenology and the entrance of the core into the transition phase will be discussed in the following section.

15.3.3 Recriticality

Introduction

After the issuance of the Safety Evaluation Report for the FFTF[1] in October, 1972, the Advisory Committee on Reactor Safeguards and the staff requested that the Project consider the possibility of recriticality for core disruptive accidents in the FFTF and the safety implications of recriticality. In response to this request, the Project had Argonne National Laboratory conduct a recriticality study.[2] This study, plus a complementary one on the loss-of-flow initiating phase, was reviewed by the staff and commented on in Supplement No. 1 to the Safety Evaluation Report.[4] Portions of the Supplement relating to the review of these two reports and to the issue of recriticality in general are quoted here for completeness. From page 18:

> "... It appears that HEDL and ANL believe that the LOF sequence in the FFTF will involve a relatively gradual meltdown of the core during which mild nuclear excursions might occur with reactivity insertion rates of less than 50 $/sec. Eventual whole core involvement in the meltdown was indicated to be

likely. According to these assessments a considerable amount of the molten fuel will be ejected into the upper plenum area above the core although detailed analyses of this phase of the sequence are not yet completed. ...Our major concern with the LOF accident in the FFTF has been the potential for recriticality involving large reactivity insertion rates. These can be postulated to occur as a result of coherent gravity recompaction of substantial amounts of material falling through substantial distances and/or pressure driven recompaction of lesser amounts of material. Sufficient reactivity must be inserted in either case to achieve prompt criticality...

From page 20:

... An aspect of the LOF analysis that requires considerably more study is the "Transition Phase." This phase begins at that part of the analysis where subassembly integrity is starting to be lost. In past analyses fuel slumping was assumed to lead to mild disassembly events (see ANL/RAS 71-29). Although mild excursions may still occur the new analyses give considerably more attention to details of fission product gas effects and clad and fuel motion including the effect of vapor streaming on liquid clad and fuel behavior. A new phenomenon has been modeled in this regard and is described as being potentially very important. This is the phenomenon of "flooding" or fluidization which can (according to ANL staff members) lead to a "dispersed flow regime." This is the configuration ANL believes will most likely exist during core meltdown and while the upper subassembly structure is being melted out to permit fuel explusion into the upper plenum volume. An important aspect of this modeling is the large degree of subcriticality implied by this (largely voided) configuration. This may make rapid recompaction events leading to prompt criticality even less likely. This modeling is quite recent and requires considerably more study and experimental support."

And finally, from page 21:

"Pending further evaluation of the transition phase analysis and pressure driven recompaction studies for LOF accidents... it appears to us that maximum ramp rates during CDA's in the FFTF may well be less than 100 $/sec. This would imply that the existing primary containment structure would not be breached during a CDA in the FFTF....."

It is the purpose of this section of the report to update the conclusions drawn in this Supplement based on: the Final Safety Analysis Report[5] evaluation, the "Updated Assessment" by ANL[6] and our own independent analyses. Before summarizing the Project's analysis and presenting our evaluation, a brief description of the recriticality is included.

Recriticality Description

The term "recriticality" is used here to refer to any and all cases of criticality which occur in the accident sequence following the initiating phase (for loss-of-flow the SAS[7] analysis phase and for transient overpower the MELT-III[8] analysis phase). This part of the accident is characterized by core meltdown and is not amenable to a mechanistic description with present-day computer codes.

The focus for an assessment of recriticality is the evaluation of fuel motion. The potential driving forces for fuel motion are fission-gas and volatile fission products; sodium vapor, steel vapor; fuel vapor, gravity and, for the case of a transient overpower sequence, pump pressure head. In addition to this central focus of fuel motion, it is important to consider the presence and motion of other materials which may make up the core internals, namely, steel, sodium, control material, blanket material and fission products. The motion of these materials can augment or mitigate reactivity insertion during an excursion while their presence can augment or mitigate the expansion work energy following the excursion.

In terms of actual recriticality sequences, the subject was divided into four problem areas in the ANL Report[2] and into five problem areas in our report on core disruptive accident's for the Clinch River Breeder Reactor[9]. Namely:

(1) Secondary and tertiary recriticality analyzed in conjunction with SAS results. In some instances, SAS can be used to calculate these recriticalitity events, while for others, SAS stops due to coding limitations (e.g., when fuel moves above subassemblies). For the latter cases, the recriticality events are assessed using initial conditions from SAS and by stand-alone calculations.

(2) Reentry of fuel which was initially dispersed by mild and localized disruptions. For these cases, the subassembly duct walls are assumed intact. Both gravity and pressure-driven recompaction are considered.

(3) Recriticality associated with the "transition to the transition phase." This phase covers the time from first loss of subassembly integrity to the possible establishment of a "sealed bottle" boiled-up pool. Of course, a sustained prompt critical excursion during this phase precludes entering the boiling pool phase.

(4) Transition Phase: Once in the sealed bottle configuration, collapse of boiling fuel/steel due to external pressures, pool instabilities, and/or increase in heat transfer rates.

(5) Reentry of material following moderate disassembly (gravity or pressure-driven recompaction) into the fuel/steel pool.

For simplicity here, areas 1, 2, and 3 are lumped together and called the "Transition-to-the-Transition Phase" and areas 4 and 5 are lumped together and called the "Transition Phase."

If the core does not go into immediate disassembly, it will proceed with a meltdown which now involves the subassembly duct walls, reflector subassemblies, test loop subassemblies, and control subassemblies, and will evolve into a "sealed bottle" boiled-up pool (commonly referred to as the transition phase) barring the occurring of significant recriticality in the interim. It is this interim which is referred to as the transition to the transition phase.

The description of this interim state of the core is much more difficult than when either all subassemblies are assumed intact or the core has evolved into a relatively homogeneous pool enclosed in a sealed system as in the transition phase. The evolution of the core will be highly dynamic with subassembly duct wall melting; intersubassembly sodium being forced out of the core; large portions of core solid structure and fuel shifting and dropping by gravity; core boundaries dramatically changing by freezing and plugging, melting and remelting; and fuel or fuel/steel mixtures moving relative to other core components by driving forces due to fission gas, fuel and steel vapor, and gravity. In assessing such a dynamic state of the disrupted core, it is important to understand the potential for recriticality.

The investigation of the transition phase is considerably simpler than that made in the transition-to-transition-phase, since the merging of, and gradual transitions between, more than one phase is absent. In order to assess the recriticality potential, two physical processes are key: namely, boiling pool hydrodynamics and pool boundary melting (heat losses).

Given a boiled-up transition phase pool, there are several ways by which, in principle, such pools can be made to achieve recriticality: (a) by pool collapse due to pressure sources at the boundary, (b) by pool collapse due to vapor condensation by heat losses or by passive additions of molten steel through the pool boundaries, or (c) by reentry of previously expelled fuel slugs formed in the upper blanket and fission gas plenum regions. The physical processes and how they relate to the three ways to achieve recriticality are the subject of the Project's evaluation and our assessment.

Summary of Project's Analysis

The Project's position on recriticality is summarized in Section A.3.2 of the Final Safety Analysis Report.[5] Only recriticality from the "Transition Phase" is taken into account. The Project's position for the transition phase is that

"...the probable mode of disassembly, if the core were to somehow proceed into a transition phase, is by upward blowdown into the outlet sodium plenum region. Mechanical work on the primary boundary would be low. The expected consequences of loss-of-flow with failure to scram thus do not threaten the primary coolant system integrity and have no subsequent safety implications. The dispersed fuel in the outlet plenum and that remaining in the core would be solidified and cooled as described in Section A.7 on Postaccident Heat Removal."

The main resource report for the Project's position on recriticality[2] covers various "transition-to-transition phase" areas in addition to the transition phase itself. Summaries of these analyses can be found (in addition to in the report itself) in Chapter 4 of Ref. 9 where the ANL FFTF analysis is compared to the Clinch River Breeder Reactor recriticality analysis. A summary in tabular form is given below where the recriticality sequences are broken down in five categories as described in the "Recriticality Description" Section above.

EVENTS ASSUMED FOR RECRITICALITY ANALYSIS

Recriticality Sequences	Previous Analysis[2] for FFTF
(1) Secondary and Tertiary Recriticality Analyzed in Conjunction with SAS3A	Maximum Range 5-20$/sec
(2a) Gravity-drive Reentry of Fuel Initially Dispersed by Mild Localized Disruptions (S/A Ducts Intact)	Maximum Range 30-40$/sec
(2b) Pressure-drive Recompaction (S/A Ducts Intact)	Maximum Range 50-100$/sec
(3) "Transition to Transition Phase"	No Specific Analysis
(4) Transition Phase (Collapse of Boiling Fuel-Steel Pool)	F 25$/sec
(5) Reentry of Material During Transition	F 100$/sec

The ANL analysis[2] concludes (p.74):

"In summary, it is our current judgement that the most probable path to termination is a boil-out of molten core materials into the above core sodium that does not involve an energetic disassembly excursion. The possibility of a disassembly excursion cannot currently be excluded, nor can its probability of occurrence be accurately assessed. Based on our current understanding we

judge that such excursions would not likely be initiated by reactivity insertions larger than about 100$/sec. It should be emphasized that this is an engineering judgement, however, and not the result of a rigorous analysis."

In a recent update assessment of the unprotected loss-of-flow accident[10] it is concluded that the previous ANL analyses are still valid and that (p.41),

"Even though the nominal path is expected to be a gradual meltout and ejection of core materials to the above-core sodium plenum, more energetic accident terminations were also considered in ANL/RAS 74-16. Cases were investigated in which secondary criticalities occurred which might lead to sufficient fuel vapor pressure to energetically and permanently disassemble the core. On the basis of these earlier studies, and the above evaluation of accident energetics phenomenology developed since that time, it is felt that a reasonable judgement can be reached that it is very unlikely that ramp rates as high as 100$/sec could be induced in FFTF as a result of the unprotected loss-of-flow accident."

Staff Evaluation

For core disruptive accidents in liquid metal fast breeder reactors, there are generic potential accident progression paths, and therefore, in a general way, FFTF recriticality is similar to the recriticality considerations for the Clinch River Breeder Reactor. Thus, the same type of considerations should be taken into account as were taken into account in Chapter 4 of Ref. 9. In addition, Chapter 4 lists recriticality ramp rates which are calculated for the FFTF Project.[2] In the recent update of the FFTF transition phase analysis,[10] a main conclusion is that the conclusions drawn in Ref. 2 remain valid. Thus, Chapter 4 of Ref. 9 is a good starting point in considering the FFTF recriticality potential in general and our evaluation in particular.

There are unique aspects to the FFTF (relative to the Clinch River Breeder Reactor) which have a bearing on core disruptive accident progressions, and in particular, core meltdown phase sequences:

(1) The FFTF has no axial or radial blankets. In place of these, there are Inconel-stainless steel reflectors (axial reflectors are 6" long at each end).

(2) The core volume and fissile inventory are both smaller than the Clinch River Breeder Reactor (volume ratio 0.41; fissile inventory ratio 0.46) while the FFTF enrichments are greater (inner core .224 vs. .177; outer core .274 vs. .256).

(3) In addition to the 9 control/safety rod subassemblies (Clinch River Breeder Reactor has 19 control and safety rod S/As), FFTF has a variety of special subassemblies (up to 9) which can be used for open and closed loops (two with

independent cooling systems) for experiments. The impact of the accident progression will be influenced by the characteristics of these special subassemblies which can, of course, change depending on the experiment in place.

(4) As a direct result of the smaller size of the FFTF, the whole core sodium-void reactivity coefficient is negative (as compared to positive for the Clinch River Breeder Reactor). This affects the initiating phase of the loss-of-flow accident, thus altering the initial conditions of the following "transition-to-transition" and transition phase.

With these differences in mind the present staff assessment of recriticality for the two key areas in the meltdown progression namely, the "transition-to-the-transition phase" and the transition phase is presented.

Transition-to-the-Transition Phase

There are no mechanistic analysis tools presently available with which to assess recriticality potential during this phase of the accident meltdown progression. In performing an assessment of recriticality then, we do not consider "mechanistic" scenarios but rather conservative recriticality sequences based on available reactivity. The approach to recriticality evaluation here can be divided into two parts: (1) an assessment of subassembly scale hydrodynamics and heat transfer; and (2) an assessment of the neutronic coupling of the separate, subassembly-scale events to the whole core response. First the subassembly-scale events are discussed, followed by the whole core response.

(1) Subassembly-scale Events

Two processes are key in understanding subassembly-scale events during the transition-to-the-transition phase, namely, "extended fuel motion" (hydrodynamics of the fuel/steel mixture) and "subassembly wall melting" (thermal interaction between the relatively cold subassembly wall and the high-temperature hydrodynamic region). These two processes are coupled through condensation-melting interaction. The key question to be asked is whether the fuel or fuel/steel mix within the subassembly slumps or collapses thus moving to a more reactive configuration. If this is indeed the case, then this "collapsing" phenomenon must be considered on a core-wide basis in order to address recriticality potential. It is our position that there will be a certain amount of slumping or collapsing. Slugs of fuel which have been driven up out of the active core region by, say, steel vapor pressure can reenter by gravity slumping or, possibly, by pressure-driven recompaction. These types of reentry phenomena have been considered by the Project and are assessed in Ref. 2. With the assumption that these fuel slugs are reentering a core very close to delayed critical, the conservative analysis indicates ramp rates through prompt critical of from 5$/sec to 100$/sec,

the latter being pressure-driven recompaction. A recent analysis[11] by one of our consultants indicates that it is more likely that a well-mixed fuel/steel hydrodynamic zone will exist as opposed to separate fuel slugs. In addition, the same analysis indicates that substantial upper and lower plugs will form prior to extended fuel motion, thus precluding sodium-vapor-pressure driven recompaction. On a subassembly scale then the chief concern is whether or not this boiled-up mix of steel and fuel can be produced and maintained thus keeping the fuel in the subassembly from collapsing into a neutronically more reactive configuration. A conclusion of this study is that, for the FFTF, this boiled-up pool will collapse either by an inadequate supply of new steel from the subassembly walls (since steel is considerably lighter than fuel, it will rise rapidly in the pool as a portion of it vaporizes) or by a rise in heat losses to subassembly walls or other heat sinks. Also, these collapses will tend to repeat thus possibly developing hydrodynamic oscillation.

(2) Whole Core Response

Since the collapse of these subassembly sized pools is probable, and since these pools will have formed in a considerable number of subassemblies, it is important to consider the implications for the whole core in terms of recriticality. At first the oscillations will be random and neutronically incoherent. If this dynamic system should pass through (delayed) critical, fission heat will be generated coherently throughout the critical system. This, however, is anticipated to be a sluggish recriticality which may create sufficient pressures to cause a blow-down or set the stage for further sluggish recriticality which in turn may cause a blow-down thus neutronically shutting down the core. The concern here is whether large sustained recriticalities can be achieved, thus generating considerable work energy. There is, of course, presently no analysis capability for mechanistically treating this dynamic process. However there are scoping reactivity calculations which support the opinion that these excursions will be mild and not sustained. For example, the sustained ramp rate through prompt critical (in $/sec) (necessary in order to have significant work energy) can be thought of as the product of the velocity of the collapsing or reentering fuel mass (cm/sec) and the worth gradient of that event ($/cm). For an entirely coherent collapse of the FFTF core the worth gradient is of the order of 1$/cm; for any less coherent event, the value would be less. Thus for the transition-to-transition-phase sequence described above, it would be considerably less than 1$/cm, at least initially. As the core dynamics become more coherent (if it does at all) the hydrodynamic collapse velocities would be limited to those of a boiled-up pool as described in the next section which discusses the transition phase namely, less than 30-40 cm/sec.

Thus it is our opinion that energetic recriticality is highly unlikely during the transition-to-the-transition-phase. That is not to say, however, that there will be no recriticality during this phase. Mild recriticality may induce a blow-down

which would shut the core down permanently thus precluding transition phase considerations.

Transition Phase

As noted in the last section questions may be raised about the possible existence of the transition phase. However at this time it cannot be excluded from the accident sequences that need consideration. In this section, therefore, we will assume the existence of the transition phase for the purpose of exploring the potential consequences. The physical processes are in a general sense akin to the transition-to-the-transition-phase, namely, boiling pool hydrodynamics and pool boundary melting (heat losses), but there are several important differences. In particular upward pressure relief paths are likely to develop which will influence the course of events.

Given then a boiled-up transition phase pool there are several ways by which, in principle, such pools can be made to achieve recriticality: (a) by pool collapse due to pressure sources at the boundary, (b) by pool collapse due to vapor condensation by heat losses or by massive additions of molten steel through the pool boundaries, or (c) by reentry of previously expelled fuel slugs formed in the upper blanket and fission gas plenum regions.

(a) Pool Collapse due to Pressurization

Potential pressurization mechanisms are the vapor pressures of the more volatile sodium and steel. It is difficult to envision circumstances in which liquid sodium would be entering the pool through its boundaries under sufficient pressure to produce a large enough volume entrainment and thus pressurization. However, the FFTF reactor is designed to contain two pressurized ($\cong$ 14 atm) sodium loops (CLIRAs) which will fail most likely during the latter times of the transition-to-the-transition phase or into the transition phase itself. (One estimation is that the CLIRA outer wall will melt in 3 seconds following adjacent subassembly meltdown.) As a result of this failure, sodium may be injected into the boiling fuel pool from within. Even if a vapor explosion (msec time scale) does not take place, rapid vaporization of sodium by radiation, condensation, and convection will take place. At these temperatures sodium vapor will act like inert gas, producing a highly subcooled pool and thus collapse. However the resulting sodium pressures would be most likely originating near the core center, and rather unlikely at the core boundaries. The initial pool response would further decrease the system reactivity until the vapor begins to separate by bubbling and frothing. Based on similar arguments to those presented in connection with separation of steel or steel vapor, ramp rates for this process in excess of $\sim$ 30-40$/sec cannot be visualized. It should be recognized, however, that this is a highly undesirable event.

Pool pressurization by steel vaporization is not envisioned except in the sense that a steel-fluidized pool is significantly below the boiling point of fuel and if an interruption in steel supply was to occur, collapse would take place. This problem is considered further under (b) below.

(b) Pool Collapse due to Vapor Condensation

Coherent and sustained pool collapse has not been shown to be realistically possible to date. Jackson et al.[2] arbitrarily postulated an instantaneous suppression of all vapor production in a boiled-up subcritical pool. In this case pool subsistence is controlled by the rate of the exiting vapor which was estimated, using a drift flux formulation and a churn-turbulent flow regime, at 23 $cm^3/sec/cm^2$. Pool surface drop at 23 cm/sec is thus implied, which together with the figure of 1$/cm at prompt critical for coherent compaction produces a ramp rate of 23$/sec. This rate is probably an overestimate since the rate of phase separation drastically reduces as the void fraction decreases and the flow regime changes to liquid continuous.

Since heat generation cannot physically be reduced below the small but very effective decay heat levels, our consultants searched experimentally for mechanistic means to produce pool collapse. Theofanous[11] determined that realistic pool collapse velocities can be bounded at about 13 cm/sec yielding ramp rates of $\leq$ 13$/sec.

(c) Fuel Re-entry

This is the most extensively studied recriticality problem. Ostensen and Henninger[12] found that a coherent 18 subassembly gravitational re-entry into the boiled-up Clinch River Breeder Reactor core will produce less than 30$/sec. These 18 subassemblies contain the equivalent of 9% of the core fuel and they drop 20 cm into the pool achieving a velocity of 200 cm/sec. Boudreau and Jackson[13] presented similar calculations for the FFTF. They considered a large number of variations and concluded that 100$/sec ramp is a reasonable "engineering judgement" figure. They did provide, however, several mitigating circumstances, incoherent re-entry being a major one.

No phenomenological indication exists that significant fuel masses can re-enter coherently from a plugged up upper core structure region that is sufficiently strong to contain a transition phase pool. Thus we view the existing re-entry calculations as useful studies for providing an understanding of the limits to recriticality for the whole-core pool.

In conclusion, we do not see any mechanistic routes to transition-phase recriticality with attendant large work-energy releases for the FFTF. All the conservative

scenarious considered had ramp rates at prompt critical less than 25-40$/sec which is well within the bound used to evaluate the reactor coolant system and containment design.

15.3.4 Mechanical Disruption/Structural Response

(a) Mechanical Disruption

In Sections 15.3.1, 15.3.2, and 15.3.3 the potential for sustained reactivity ramp rates was discussed for transient overpower, loss-of-flow, and recriticality, respectively. This section is presented in order to provide a comparison between estimates of the ramp rates and the reactivity ramp rates associated with large work energies (100-350 MJ) which are used to assess the reactor coolant system design.

The potential for structural damage to the reactor vessel from a core disruptive accident may be evaluated in a variety of ways. The traditional method has been to divide the accident into phases and use a separate computer program to evaluate each phase. The Project has analyzed the initiating phase of the accident with MELT III[1] for transient overpower and SAS2A[2,3] for loss-of-flow; the disassembly phases with VENUS-II[4,5]; and the resulting work potential with SOCOOL-II[6] when Na-vapor was assumed the working fluid, and with an isentropic expansion calculation when fuel-vapor was assumed the working fluid. The results of the analysis by the Project indicate that the range of energetic accidents can be covered by considering either a direct disassembly from a loss of flow or transient overpower accident or by a subsequent recriticality event. The Project has stated and we concur that no mechanism can be described which produces ramp rates beyond a few tens of dollars per second. Within this context, the Project has chosen the results of a SOCOOL-II calculation as a design objective.[7] The energy is found by an isentropic expansion of sodium vapor which is produced by an energetic fuel-coolant interaction. The energy available to do work from this expansion down to a final pressure of one atmosphere is 150 megawatt - seconds (or 150 megajoules (MJ)). This is the "reference case" (Figure 15.3-2). In addition to the 150 MJ case, the Project analyzed two additional accidents to evaluate the design margin.[8] These accidents were characterized by reactivity ramp rates of $100 and $110 per second, and the work was evaluated by an isentropic expansion of the fuel vapor at the average core temperature. The analysis also considered the potential for augmenting the work with sodium which might become entrained in the expanding bubble.[9] The pressure-volume curve for the 110$/sec case is shown in the figure.

There is no single number which can completely describe the potential for damage to the reactor vessel from a core disruptive accident. A number which is often used is the energy from an expansion of the working fluid (sodium, fuel, or both) to one atmosphere. Since the sodium slug will impact the reactor vessel head at a pressure much higher than one atmosphere, this number is of little value in describing the

PRESSURE VS. VOLUME DISASSEMBLY CALCULATIONS

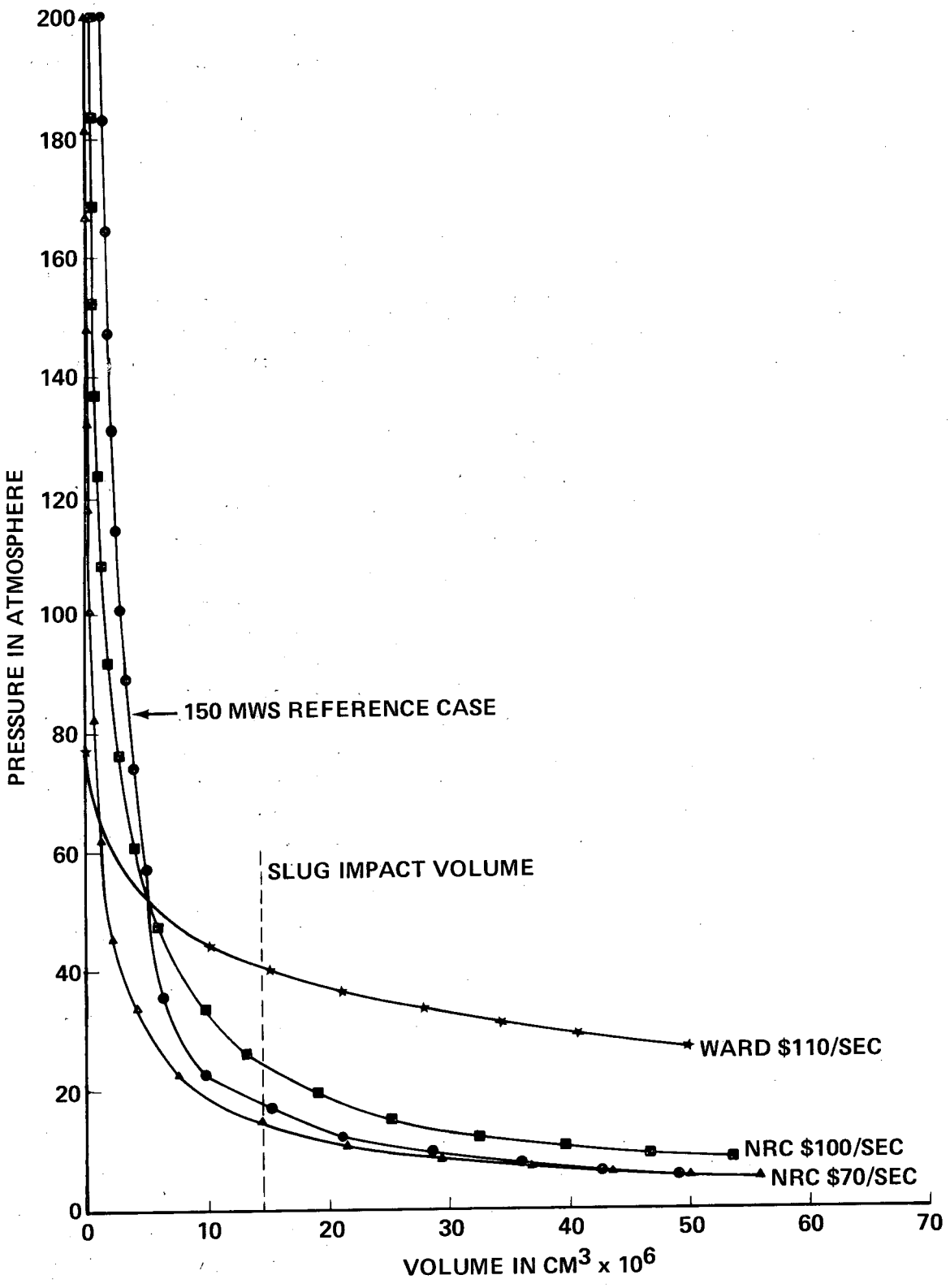


Figure 15.3-2

accident. A method which is somewhat more useful is measuring the work which could be done in expanding to a volume equivalent to the point where the sodium slug will impact the vessel head. This method, however, still does not account for the effect of the initial pressure or the pressure at the moment of slug impact. Figure 15.3-2 shows several pressure-volume curves for hypothetical core disruptive accidents. Each of these curves will produce different amounts of deformation to the reactor vessel when used as input to computer codes such as REXCO-HEP.[10] The various shapes of the curves in the figure result from the different methods used to do the isentropic expansion. The 150 MJ reference case uses sodium as a working fluid. This results in a high initial pressure and a fairly efficient expansion. The WARD 110$/sec case uses UO_2 as the working fluid but the expansion is based on the average core temperature and augmented by heat transfer to the entrained sodium. Although the initial pressure is lower than the reference case, the pressure at slug impact is much higher.

The curves for the two staff cases are from calculations using cell by cell expansion of the core at the termination of the VENUS-II calculation. Fuel vapor is the working fluid. The initial pressure is determined by the highest temperature in the core, so it is much higher than the pressure which corresponds to the average temperature. However, very few cells have these high temperatures and the pressure drops rapidly as the bubble grows.

Which of these curves is most appropriate as input to a REXCO-type code depends on the exact evolution of the bubble during the accident. We used the method of cell by cell isentropic expansion of the fuel for the Clinch River Breeder Reactor[11] and consider it appropriate for use here.

We performed a series of VENUS-II calculations for a representative FFTF model in order to determine the initial conditions for expansions discussed above. This model was for a loss of flow accident. All sodium was assumed to have boiled out of the core region and a significant fraction of the fuel was molten. The initial reactivity was set at $1.00 (just prompt critical) and the initial power was 100 times the nominal steady state power of 400 MW. With this basic model parametric calculations were done for three ramp rates. Table 15.3-1 shows the results of these calculations. The energy from expansion up to the slug impact volume from the 150 MJ reference case of the Project is about 80 MJ. This is roughly midway between our 70$/sec and 100$/sec cases. The energy at the same volume from the Project's 100$/sec calculation[9] is about 70 MJ but the final pressure is higher and calculated damage to important components was greater. This makes it approximately the same as our 100$/sec case.

Based on our calculations and our review of the Project calculations it is our opinion that the 150 MJ reference case places a reasonable bound on the consequences

TABLE 15.3-1

FFTF VENUS-II DISASSEMBLY CALCULATIONS

Initial Conditions

Power = 100 Po

AVERAGE TEMPERATURE=2711°K

Total Reactivity=$1.00

FINAL CONDITIONS

REACTIVITY RAMP RATE:	$40/SEC	$70/SEC	$100/SEC
TOTAL ENERGY	1500 MJ	2133 MJ	2598 MJ
ENERGY IN MOLTEN FUEL	1270 MJ	1850 MJ	2280 MJ
PEAK TEMPERATURE	5430°K	6023°K	6560°K
AVERAGE TEMPERATURE	3533°K	3947°K	4238°K
WORK @ 1 ATMOS.	48.4 MJ	147 MJ	262 MJ
%VAPOR @ 1 ATMOS.	5.3%	11.6%	17.4%
WORK @ SLUG IMPACT	20.2 MJ	56.6 MJ	101 MJ
% VAPOR @ SLUG IMPACT	2.7%	6.2%	9.8%
WORK BASED ON AVERAGE TEMPERATURE @ 1 ATMOS.	0.0 MJ	55 MJ	116 MJ

TOTAL FUEL MASS=2.85 x 10^6 gram

Po = 400 MW

of a core disruptive accident, and the Project's use of the 100$/sec and 110$/sec cases represents a reasonable basis for the assessment of design margin. Reactivity insertion rates in excess of $50-$100 per second are judged to be extremely unlikely. We find that although different methods have been used to compute the pressure-volume relationships which characterize these accidents, the Project's choices were appropriate and adequately model the expansion of the core within the known uncertainties.

(b) Structural Response

The Projects structural analysis is currently summarized in Ref. 12. We and our consultants have reviewed this report and compared the results contained therein to those presented by WARD[2] in 1974. The objective was to determine if the old and new analyses substantially agreed and if the latest results would affect our previous conclusions regarding the adequacy of the FFTF reactor coolant system design.

The results presented in Ref. 12 do not differ significantly from those originally presented[8] and those presented at the Project-staff meetings in the fall of 1974. For many components Ref. 12 gives more detailed results. In addition the results of several analyses performed since 1974 are reported.

The staff and consultant review resulted in the same basic conclusions that were reached in 1974 regarding the adequacy of the FFTF reactor coolant system design. The FFTF reactor coolant system should not grossly fail for the 150 MJ transient overpower core disruptive accident and for the 350 MJ loss-of-flow core disruptive accident. Movement of the reactor vessel and head are within available clearances; the operating deck should not be impacted or penetrated. The head-mounted components are expected to remain attached and missiles are not expected to be generated. The primary piping system should remain functional for fluid containment.

The additional components analyzed and the more detailed results reported in Ref. 12 indicate that the safety margins in the reactor coolant system design may be smaller than had been estimated in 1974, but sufficient.

We and our consultants at the Naval Surface Weapons Center also analyzed sodium leakage. For a conservative case, up to about 1000 lb (454 kg) of sodium could be expected to be ejected directly into the head access compartment. The major possible leak paths were found to spray sodium sideways. At most no more than about half of this sodium would spray into the upper portion of the head access compartment with the remainder forming a sodium pool on the floor of the head access compartment. Under these conditions, which include spray deflectors, sodium spray should not pose a threat to the containment building.

In conclusion, the FFTF reactor coolant system is expected to remain intact for the 150 MJ transient overpower core disruptive accident and for the 350 MJ loss-of-flow core disruptive accident. Possible sodium leakage of up to about 1000 pounds (454 kg) into the head access compartment should not pose a threat to the containment building from rapid overpressurization.

15.3.5 In-Vessel Post-Accident Heat Removal

The capability of the FFTF to contain a core meltdown event in-vessel within the steel dome of the lower core support structure was extensively addressed by the staff as part of its review in Supplement 2 to the Safety Evaluation Report, issued on March 7, 1975. At the time the staff concluded that "substantial capability to accommodate the effects of a core disruptive accident and maintain the integrity of the primary coolant boundary do exist" and that "in-vessel post-accident heat removal is reasonably assured following such an event." However, it was also noted in Supplement 2, as well as in Supplement 1 in which the staff addressed the issue of core disruptive energetics among other issues, that "the events leading to and following core disruption are subject to considerable uncertainty;" and that "the possibility of loss of vessel integrity cannot be completely discounted, if for no other reason than the uncertainties in defining the final disposition of the debris." These uncertainties and recommended areas for additional studies and research and development are described in Sections 5 and 4 of Supplement 2 and 1, respectively, as well as discussed in numerous meetings with the FFTS Project personnel and the

Advisory Committee on Reactor Safeguards. At the March 17, 1975 Denver ACRS meeting, for example, specific in-vessel post-accident heat removal concerns were discussed, such as (1) uncertainties in the model of fuel movement and pool formation mechanisms in the lower dome, (2) validity of the model for determing heat transfer directions (up, down and laterally) at the FFTF vessel lower dome, (3) non-quiescent vs. stratified pool heat transfer, and (4) the small scale supporting experimental program which is not truly prototypic. Based on the uncertainties discussed above, the staff concluded in Supplement 2 that "the residual risk which arises from certainties in in-vessel post-accident heat removal makes it prudent to provide for the consequences of this low likelihood event in the best possible manner," and specifically to "consider means to alleviate high doses associated with vessel melt-through."

The staff further advised that "emergency planning should be initiated for the FFTF pursuant to 10 CFR Part 50, Appendix E," and that "the time to melt-through or alternative measures could have considerable impact on the content of emergency plans." Finally, the staff advised that "RRD's contractors should determine the best method for reducing the radiological consequences of a melt-through consistent with appropriate emergency plans. More refined calculations may reduce the potentially high radiological doses currently computed for a melt-through or alternatives to increased delay time should be examined by RRD's contractors such as including scrubbers, filters and iodine traps in the building exhaust system. Possibly a combination of improved delay time and inclusion of some of these may be appropriate to control the radiological releases."

Since the staff's evaluation of the adequacy of in-vessel post-accident heat removal in Supplement 2, the Project has not provided any substantive information which would make the staff change its previous position as reflected above. Emergency planning procedures have been implemented at the FFTF and the radiological consequences are addressed in Section 15.3.7 of this report.

15.3.6 Containment Response to a Core Meltdown Event

1. Introduction

This section provides an assessment of containment margins available in the FFTF to mitigate the consequences of a postulated failure of in-vessel post-accident heat removal following a hypothetical core-disruptive accident.

Following failure of in-vessel post-accident heat removal, the Project relies on the inherent retention capability of the FFTF reactor containment building to safely contain a postulated core meltdown event. Based on analyses reported in Ref. 1, the Project maintains that the containment would remain intact and effective for long periods so that the radiological doses following containment

failure would be below 10 CFR Part 100 guideline values. Our review of the Project's analysis indicates a great deal of concern about the basis and justification of assumptions, analysis methods, and conclusions related to the adequacy of FFTF containment margins, especially as they affect the radiological consequences of a core melt-through event. The Project's approach involves predictions of complex material interactions and structural behavior of steel and concrete containment structures under very severe thermal conditions. Our review finds that there are presently large uncertainties associated with these evaluations. Following reactor vessel and guard vessel melt-through, the evaluations of the meltdown scenario and the resultant radiological doses are highly dependent on some key assumptions made in the Project's analysis, such as the fragmentation, self-leveling and coolability of the molten fuel/steel mixture entering the reactor cavity, the potential failure and the extent of failure of the reactor cavity liner, and the subsequent sodium and core material penetration into the exposed reactor cavity floor concrete. In the Project's analysis, for example, a sodium-concrete penetration rate of one-half inch per hour was used for four hours, whereas experimental evidence exists from both HEDL[2] and the recently performed large scale tests at Sandia,[3, 4] which indicate that this rate could be higher, perhaps in the range of one to three inches per hour even without the influence of molten fuel and steel. Other key assumptions are the rates and quantities of gas and water vapor generated from the interaction process, the natural recombination of hydrogen in the presence of air and sodium, and whether the venting system behind the reactor cavity steel liner is effective in removing water vapor and preventing reaction with sodium that has spilled onto the reactor cavity floor. The Project's calculations, as well as those performed by the staff, show that treatment of these parameters is crucial to the calculated impact on the containment structure in terms of exceeding the containment pressure and temperature limits, breaching the containment building basemat, and exceeding the lower limit hydrogen concentration for an explosive mixture with air (4% was selected as the limit and is regarded as sufficiently lower than the concentration of explosive mixtures with air).

2. Summary of Project Results

In Ref. 1, the Project postulated two general configurations resulting from failure of in-vessel post-accident heat removal. The first which is termed "in-vessel" assumes a complete loss of heat sink subsequent to a hypothetical core disruptive accident. It is also assumed that the core debris remains inside the vessel and that the decay heat is transferred to the stagnant in-vessel sodium. The in-vessel configuration represents a situation where the core debris remains coolable within the reactor vessel, but where heat dissipation through the main heat transport loops has been lost by core disruptive accident related structural damage. The second configuration which is termed "ex-vessel" assumes that the core debris has melted through the

reactor and guard vessels and, along with the reactor vessel sodium inventory, comes to rest on the floor of the reactor cavity. For both configurations, the Project assumes that the core debris remains coolable and in a particularized form within the sodium pool, so that all the decay heat would be transferred directly to the sodium. Hereinafter this core debris configuration will be referred to as the "coolable debris bed" configuration.

The Project has considered a range of conditions for both in-vessel and ex-vessel configurations. The conditions which were assessed include the effects of vent and purge of containment, reactor cavity liner failure, structural failures (collapse of reactor cavity floor and failure of the reactor support ledge), hydrogen-oxygen recombination, and engineered systems (reactor containment building coolers and exhaust filtration). For the in-vessel cases, the Project determined that the containment would need to be vented to preserve its integrity in the range of 82 to 123 hours depending on whether the reactor containment building pressure was relieved either by venting at 10 psig (design pressure) or by structural failure at 20 psig (assumed structural failure point). For the ex-vessel cases, the results indicate that containment would have to be vented in the range of 42 to 229 hours because of exceeding either the 10 psig reactor containment building design pressure limit or a 4 percent reactor containment building hydrogen concentration limit. If the pressure is allowed to increase to 20 psig (i.e., twice the design pressure) and relieved at that time due to structural failure, the Project's ex-vessel results indicate that blowdown would occur at about 960 hours. Based on radiological evaluations for the above range of containment atmosphere relief times, the Project concludes that the calculated site-boundary doses are below the guideline values of 10 CFR Part 100, even without the addition of engineered systems, such as reactor containment building coolers and exhaust filtration.

3. Summary of Staff Analysis Methods

We have performed a series of analyses, with the use of the CACECO code,[5] of the transient pressure and temperature response of the FFTF containment building following a hypothetical core melt-event and failure of in-vessel post-accident heat removal capability (i.e., loss of main heat transport loops to dissipate core debris decay heat). The purpose of these studies was to assess the time to challenge containment from either excessive hydrogen (H_2) concentrations or excessive containment pressure and/or temperature and to evaluate the radiological release at the site boundary following a core disruptive accident for up to thirty days using the containment leak rate and venting in parallel to control H_2 concentration or pressure whichever is limiting. We have also analyzed both in-vessel and ex-vessel core melt configurations.

For the ex-vessel cases, we and our consultants performed analyses assuming the core debris to be either in a coolable particulate debris bed form (as assumed by the Project) or in a molten pool form. We felt it necessary to analyze the latter configuration because the assumption made by the Project, that following melt-through of the reactor and guard vessels the molten fuel/steel mixture will immediately fragment, self-level, and attain a coolable debris-bed configuration, is not fully substantiated by experimental data. The Project's assumptions on coolable debris bed formation are based on very limited experimental data,[6] obtained with small quantities of molten materials (< 2 kg). In the absence of experimental data of a sufficiently large scale to demonstrate the validity of the Project's assumptions, we considered it prudent to also analyze the containment response and penetration into the reactor cavity floor assuming that the core debris remains in a molten pool form.

We and our consultants did not attempt to confirm all of the Project's containment margins cases presented in Ref. 1. Our objective was rather to identify those assumptions made by the Project which appear to most strongly influence the subsequent accident scenario. Once an area of concern was established, its impact on the subsequent accident progression was determined by sensitivity studies. Although the details of our containment margins sensitivity evaluations are presented in the following subsections, and in Appendix F and Ref. 7, a summary of the principal parameters which were found to significantly influence the containment transient response to a meltdown event are provided below:

(1) The in-vessel cases have revealed considerable sensitivity to:

   (a) type and size of the flow path that is assumed to exist between the reactor vessel and the containment building;

   (b) the quantity of sodium that is ejected from the reactor vessel during the core disruptive phase of the accident; and

   (c) whether there is an additional vent path from the reactor cavity to the reactor vessel besides the reactor vessel to containment building vent (a one-way flow valve which prevents oxygen from entering the reactor cavity from the containment building and reacting with sodium was found to significantly ameliorate this situation).

(2) The ex-vessel cases have revealed considerable sensitivity to:

   (a) whether or not the reactor cavity steel liner fails;

(b) the rate and extent of sodium-concrete penetration (if the liner fails);

(c) collapse time of reactor cavity floor (when collapse occurs sodium comes into contact with the unlined concrete walls of the subcavity (see Figure 15.3-3) resulting in a severe containment transient);

(d) sodium vaporization time (those parameters which most strongly affect sodium vaporization include the pool size, disposition of decay heat, and the chemical reactions which result from the contact of sodium and concrete); and

e) transport of sodium vapor from the reactor cavity to the upper containment building (the parameters which influence sodium vapor transport are the vent size and vent design options, such as "rupture disks" and one-way flow valves; however, the Project's containment margins evaluations are based on non-mechanistic venting or leaking from the reactor vessel through seals in the reactor head to the upper containment building).

For both the in-vessel and ex-vessel scenarios, the combustion of sodium vapor in the containment building is the largest contributor to the transient. Therefore, those parameters which facilitate sodium vaporization and transport from the reactor vessel to the containment building are the controlling factors in the accident sequence.

4. Summary of Staff In-Vessel Evaluations

The in-vessel scenario begins with the assumption that, following the hypothetical core disruptive accident the core debris is distributed in the reactor vessel as particulate beds in the same configuration as that given in HEDL's coolable in-vessel post-accident heat removal assessment.[8] The further assumption of no coolant circulation in the heat transport system loops allows the vessel sodium to heat up and boil out of the vessel into the reactor containment building. After the sodium is boiled out of the reactor vessel, the fuel debris is assumed to consolidate and melt through the reactor and guard vessels. The mixture of fuel and steel then falls onto the floor of the reactor cavity. The mixture melts into the concrete floor until only 6 inches of reinforced concrete remain, at which point the floor is assumed to collapse. The molten pool is then released into the subcavity where it continues spreading horizontally and vertically until heat dissipation and heat generation are balanced.

A summary of the staff's and its consultant's in-vessel containment margin calculations is presented in Table 15.3-2. This table presents the results

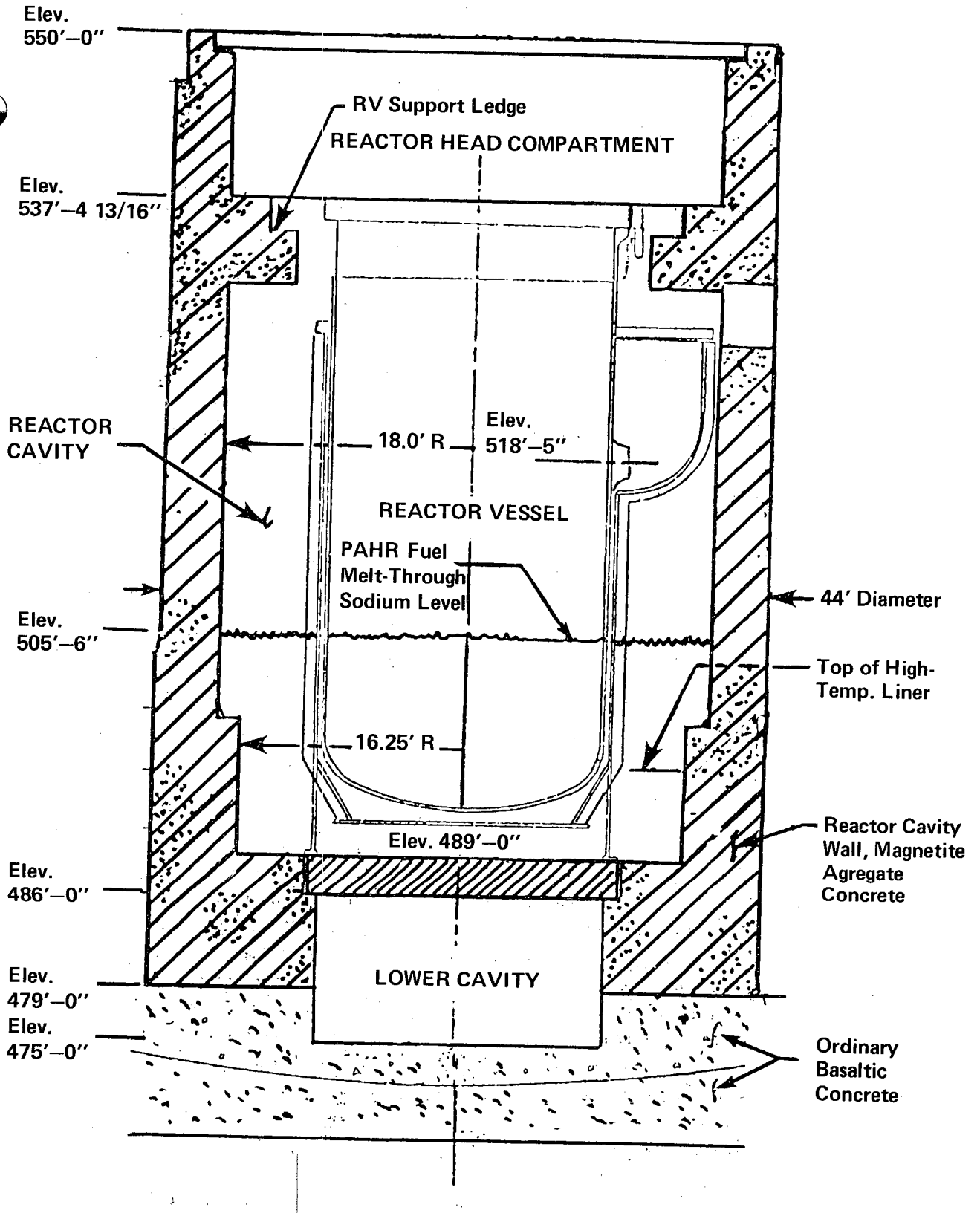


Figure 15.3-3 Sectional View of Reactor Vessel and Cavity Showing PAHR Fuel Melt-Through Level of Sodium Pool

for the times to reach either the containment design pressure (10 psig) or the lower limit hydrogen concentration (4%). The containment would have to be vented to maintain its integrity depending on which of these limits is reached first. Results are also presented for the containment building steel shell temperature which would determine whether the containment maintains its structural and leaktight integrity. The containment building steel shell should be able to maintain structural integrity at temperatures up to about 600°F. The containment seal penetrations are designed to maintain their leaktight integrity for 100 hours at 250°F. Generally, it was found that the containment would have to be vented before either of these limits is reached. Parameters which were considered in these evaluations include: size of the vent passage from the reactor vessel to the reactor containment building, rupture disc vs. continuous blowdown from the reactor cavity to the reactor containment building, two-way vs. one-way flow from the reactor cavity to the reactor containment building, additional vent path from the reactor cavity to the reactor vessel (normally the vent path for in-vessel cases is from the reactor vessel to the reactor containment building; however, if the omega seals should fail, an additional vent path would be created), the amount of sodium ejected from the reactor vessel, and liner intact or failure. A more complete description of these cases is presented in Ref. 7.

Comparison between these results and those presented in Ref. 1 indicates reasonable agreement where the input parameters were the same. For example, Case 654 of Ref. 1 indicates that 10 psig is reached in 82 hours, whereas the equivalent staff case (36 $in.^2$ vent area and no sodium ejection) shows that 10 psig is reached in 72 hours. As is evident from Table 15.3-2, the required FFTF containment vent times for in-vessel core melt scenarios range from 9 to 80 hours. Some of the cases presented in Table 15.3-2 are considered to represent upper bound or worst case conditions, such as the 42 percent sodium ejected case coupled with a total failed liner floor area (830 $ft.^2$). Considering the relatively benign energetics of a core disruptive accident in the FFTF, we believe that the vent flow area should be less than 30 $in.^2$ (reactor head seal area), and the amount of sodium ejected should be less than 21 percent. Considering the uncertainty and the sensitivity of the in-vessel results to the other parameters, our best judgment at this time leads us to conclude that between 50 and 80 hours would be available before the FFTF containment would have to be vented before exceeding either the 10 psig containment design pressure or the 4 percent hydrogen concentration limits.

5. Summary of Staff Ex-Vessel Evaluations

In the ex-vessel configuration, the reactor vessel and guard vessel are assumed to have been breached in three hours so that the entire sodium inventory, together with the entire active core, reside on the reactor cavity floor

TABLE 15.3-2

SUMMARY OF IN-VESSEL CASES

REACTOR VESSEL TO RCB VENT			REACTOR CAVITY TO VESSEL VENT	PERCENT SODIUM EJECTION FROM VESSEL	REACTOR CAVITY LINER CONDITION	TIME FOR RCB TO REACH 10 PSIG (HOURS)	TIME TO REACH 4% H_2 IN RCB (HOURS)	RCB SHELL TEMP. AT 10 PSIG (°F)	REMARKS
SIZE (IN^2)	ACTIVATION (PSID)	FLOW							
36	15	TWO-WAY	NO	0	INTACT	72	80	505	EFFECT OF SODIUM EJECTION, SEE FIGURES 2.5 THROUGH 2.7, Ref. 7
36	15	TWO-WAY	NO	21	INTACT	65	74	490	
36	15	TWO-WAY	NO	42	INTACT	49	58	494	
36	15	TWO-WAY	YES	42	INTACT	31	41	380	EFFECT OF LINER FAILURE, SEE FIGURES 2.8 THROUGH 2.10, Ref. 7
36	15	TWO-WAY	YES	42	FAILED	9	13	195	
36	15	TWO-WAY	NO	0	INTACT	72	80	505	EFFECT OF VESSEL TO RCB FLOW AREA, SEE FIGURES 2.2 THROUGH 2.4, Ref. 7
113	15	TWO-WAY	NO	0	INTACT	36	48	205	
36	CONTINUOUUS	TWO-WAY	YES	21	INTACT	35	25	380	EFFECT OF VESSEL TO RCB FLOW MODEL, SEE FIGURE 2.12, Ref. 7
36	15	TWO-WAY	YES	21	INTACT	39	49	444	

(refer to Figure 15.3-3). The sensitivity of the results to the assumed three hour melt-through time is addressed in Ref. 7. Two separate physical configurations were studied with regard to the ex-vessel cases. The first assumed that the core debris remained coolable and in a particulate debris bed form within the sodium pool, so that all the decay heat would be transferred directly to the sodium. In this configuration, only sodium attacks the concrete floor for cases where the protective steel liner is assumed to fail. The fuel/steel mixture is assumed not to interact with the reactor cavity cell concrete until all of the sodium has boiled off. In the second configuration, the debris was assumed to be molten and to occupy a layer between the reactor cavity floor below and the sodium pool above. In the second configuration, the molten pool caused a melting attack on the concrete floor after penetrating the cavity floor steel liner. The melt front penetration attack together with the associated reaction products and generated energy were calculated by the INTER code[9] and were supplied to the CACECO codes as functional input.

A summary of our ex-vessel containment margin calculations are presented in Tables 15.3-3 through 15.3-6. Results are presented for the times to reach either the containment design pressure (10 psig) or the 4 percent hydrogen concentration limit; it is assumed that the containment would have to be vented to preserve its integrity at either of these times depending on which is limiting. Some cases are also presented where H_2 is assumed to recombine with O_2 existing in the containment building provided the criteria presented in Ref. 10 are satisfied. The small-scale studies reported in Ref. 10 have identified various combinations of conditions under which hydrogen recombination was found to occur. Whether their criteria can be extrapolated to the large volume and possibly stratified conditions in the reactor containment building has not been established. As indicated by the analyses presented in Appendix F, incorporation of the H_2 recombination does not have a significant effect on the pressure transient. Although one mole of gas is effectively removed from every three moles of reactants, the heat of reaction is rather high and hence the pressure transient may be more severe when H_2 recombination is considered. The FFTF Project contractors also found the effects of recombination to be minimal (see Ref. 1, Cases 654 and 656).

5. (a) Ex-Vessel Coolable Debris Bed Results

Table 15.3-3 presents results assuming the formation of a coolable debris bed on the lower floor of the reactor cavity. The liner is assumed to fail over 346 ft^2 which represents the floor area directly under the guard vessel. Penetration into the concrete floor is by sodium-concrete reaction. Because of the uncertainty in the sodium-concrete penetration rate and duration, as discussed in Appendix C and Ref. 11, parameteric calculations were performed to examine the sensitivity of the ex-vessel

TABLE 15.3-3

SUMMARY OF EX-VESSEL COOLABLE DEBRIS BED CASES

Case(8)	Time P_{RCB} = 10 psig (hours)	Time RCB H_2 Con. = 4% (hours)	Max. RCB Atm. Temp. (°F)	Time T Max Occurred (hours)	Remarks
1	Pmax = 9 psig at 16 hrs.	13	2531	26	(1), (4), (5)
2	16	13	1474	26	(1), (4), (5)
3	16	20	1443	26	(3), (5)
4	22	14	1395	38	$\dot{P}$ = 2"/hr
5	28	14	1312	50	$\dot{P}$ = 1.5"/hr
6	40	15	1187	75	$\dot{P}$ = 1.0"/hr
7	76	15	448	100	$\dot{P}$ = 0.5"/hr
8	18	16	1302	30	(2), (5)
9	25	22	1197	44	P = 2"/hr
10	44	27	1082	60	P = 1.5"/hr
11	45	40	733	75	P = 1.0"/hr
12	80	42	252	100	P = 0.5"/hr
13	J100	45	150	75	(7)

(1) Cases 1 through 7 have H_2O vented directly to reactor cavity.

(2) Cases 8 through 13 have H_2O vented to H&V cooler room.

(3) Case 3 is with H_2 recombination; all other cases had no H_2 recombination.

(4) Cases 2 through 12 did not allow RCB venting or purging; Case 1 allowed RCB venting and purging for H_2.

(5) Cases 1 through 3 and 8 based on $\dot{P}$ = 3 in/hr.

(6) $\dot{P}$ = sodium-concrete penetration rate (continuous).

(7) Variable sodium-concrete penetration as proposed by the FFTF Project Contractor (see Appendix C, Equation 3).

(8) All cases assume a liner failure area of 346 ft^2 which is the reactor cavity floor area between the guard vessel support skirt.

containment margin results. In addition, the path that the water vapor (associated with the submerged concrete) takes is uncertain; therefore, cases are presented where the vapor is assumed to be either vented directly into the reactor cavity where it reacts with the spilled sodium to produce H_2, or is vented through the cavity vent pipes to the heating and ventilating cooler room, which is in direct communication with the upper containment building atmosphere. As is evident from the results, the time to reach the 4 percent hydrogen limit is not sensitive to the sodium-concrete penetration rate if the vapor is vented directly to the reactor cavity. However, if the water vapor is vented to the heating and ventilating cooler room, the time to reach the 4 percent hydrogen limit is sensitive to the sodium-concrete penetration rate. With respect to the time to reach the 10 psig containment design pressure, these results are shown to be sensitive to the sodium-concrete penetration rate, irrespective of what path the water vapor takes. Good agreement between staff and Project results were found for the case where the sodium concrete penetration rate and the water vapor path were the same (i.e., 1/2 inch per hour for 4 hours and water vapor vented to the heating and ventilation cooler room). A more complete description of these evaluations is presented in Appendix F.

To investigate the importance of the reactor cavity steel liner, Table 15.3-4 presents results for cases where either the liner was assumed to remain intact, or part of the cavity floor liner was assumed to fail, thereby resulting in a sodium-concrete reaction. As is evident, for cases where liner failure is assumed, the containment transient is significantly more severe. The energies associated with the sodium-concrete reactions and the sodium-water reaction dominate in the event that liner failure occurs. Results are also presented in Table 15.3-4 for the times to reach 20 psig (assumed reactor containment building structural failure point). As is evident, the containment margins increase by a factor of two for the liner intact case and less than this for the failed liner cases.

For the liner intact case, the Project predicted it would take 229 hours for the containment building to reach 10 psig, while our results indicate about 80 hours (see Table 15.3-4) would be required. This discrepancy appears to be based on whether or not sodium boils. The Project is predicting that sodium will not boil, while we are predicting sodium will reach saturation temperature in about 24 hours. Apparently, in the Project's case sodium never boils because an assumption is made that temperature equilibrium exists between the sodium and reactor cavity atmosphere. This temperature equilibrium assumption creates a fictiously high structural heat sink in the gas space. Our results, which include modeling the non-equilibrium conditions between the sodium pool and the

gas space, show that the structures in the gas space provide less of a heat sink because of the lower gas temperatures in this region. We consider the Project's intact liner results to be overly optimistic.

TABLE 15.3-4

SUMMARY OF THE EFFECT OF LINER FAILURE FOR EX-VESSEL COOLABLE DEBRIS BED CASES*

Reactor Cavity Liner Failure Area (ft^2)	Time for RCB to Reach 10 psig (Hours)	Time for RCB to Reach 20 psig (Hours)
0.0 (Liner Intact)	80	160.
300.	16	18
600.	12	15
900.	7	12

*Vent size = 36 in^2 and vent activation at 30 psid; for the reactor cavity liner failure cases, the sodium-concrete penetration rate is assumed to be 3.0 in/hr.

5. (b) Molten Pool Melt-Front Analysis

In addition to ex-vessel cases assuming formation of coolable debris beds, containment margin evaluations were made for the FFTF assuming the core debris forms a molten pool on the lower reactor cavity concrete floor. The assumption of a molten pool configuration implies that there will be an early melting attack on the concrete during heating and vaporization of the sodium. The consequences of an early melting attack may impact in the following ways:

(1) Early penetration of the fuel/steel pool into the concrete of the reactor cavity floor, resulting in floor collapse and release of the core debris and sodium into the unlined subcavity.

(2) More severe containment pressure-temperature transients due to the additional gases released from the melt front penetration and from the early collapse of the reactor cavity floor.

The CACECO code does not include a melt front penetration model so that is was necessary to use codes developed to analyze melt/concrete interactions in conjunction with CACECO to predict the containment response. Various initial molten pool configurations were assumed, and existing melt front penetration codes (GROWS,[12] and INTER[9]) were used to analyze the interaction of the fuel/steel mixture with the concrete floor. Accordingly, analysis of core debris configuration was carried out in two stages; namely, a melt front analysis and a containment response analysis.

The consequences of early molten fuel/concrete interactions during heating of the overlying sodium pool have been identified as possible early cavity floor collapse or more severe containment response due to the additional gases released from the decomposing concrete. The GROWS code basically models only pool growth, and it was therefore used as a comparison against more sophisticated INTER code, which considers gas release from the concrete, as well as oxidation processes. The calculations carried out with GROWS were, therefore, restricted to molten pool penetration into the reactor cavity floor. The more detailed INTER calculations were used to predict both penetration into the reactor cavity floor and, after floor collapse, the subsequent penetration into the subcavity concrete. In addition, gas and energy releases from the decomposing concrete, as determined by INTER, were used as input to CACECO to predict the containment response. It should be stressed that both of these models are in the early stages of development with respect to predicting the extent of penetration of core debris into concrete. However, the models are sufficiently well developed to provide a useful tool in performing sensitivity analyses to explore the importance of various parameters, and it is in precisely this manner that these codes were used by us and our consultants.

The details of our melt front penetration analysis is presented in Ref. 7. Only a summary will be presented here. Because of the uncertainties in some of the core meltdown phenomena, the melt front analysis was carried out in a parametric fashion with the use of the GROWS and INTER computer codes. Key parameters which are found to be important in determining the rate and extent of molten core debris penetration into concrete are upward/downward heat split, extent of spreading of the melt over the lower cavity concrete floor, oxide/metallic phase mass ratio and relative position, and the interfacial heat transfer coefficients (i.e., metal to concrete and oxide to concrete). Table 15.3-5 presents a summary of the results for the time to penetrate 30 inches of the reactor cavity floor (assumed failure point) as predicted by the GROWS and INTER codes. These calculations are for a 50/50 percent heat split for the amount of decay heat that goes into heating up overlying sodium

versus that fraction which goes downward to heating up the underlying concrete. Based on Ref. 13, it is felt that this is a reasonable assumption for core melt-concrete interactions. As is evident from Table 15.3-5, the time to penetrate the cavity floor is very sensitive to the extent to which the melt spreads over the cavity floor. We consider it reasonable to assume that the melt would spread over at least 50 percent of the area of the cavity floor directly between the guard vessel support skirt.

Accordingly, the GROWS code and the INTER code results indicate that penetration of the cavity floor would occur in about 30 to 50 hours respectively. The gases generated during penetration of the reactor cavity and subcavity concrete were needed as input to the CACECO code to allow their impact on the containment response to be evaluated. It should be emphasized based on imput of the quantity and associated energy of the generated gases from melt front penetration into the containment response evaluations that the gases released from the melt front had a minor effect on the overall containment pressurization when compared with failure of the cavity floor and the impact of the resulting interaction of sodium with the unlined subcavity concrete. Therefore, the time to penetrate the cavity floor is closely coupled to the time at which contain ment must either be vented or fail.

Appendix D presents the staff assessment of the state of technology related to experimental investigations of molten fuel/steel-concrete interactions.

TABLE 15.3-5

COMPARISON OF INTER AND GROWS CODE PREDICTIONS FOR MOLTEN POOL PENETRATION INTO REACTOR CAVITY CONCRETE FLOOR (ASSUMING a = 0.5

CODE USED	FRACTION OF FLOOR INSIDE SKIRT INITIALLY OCCUPIED BY POOL	CONDITIONS AT 85. cm (30 IN.) VERTICAL PENETRATION		
		TIME (HOURS)	DIAMETER POOL (CM)	MASS CONCRETE DECOMPOSED (x 103 KG)
INTER	0.04	15	401	37
GROWS	0.05	7.	315	21
INTER	0.5	52	762	107
GROWS	0.5	32	640	75
INTER	1.0	68	988	127
GROWS	1.0	64	838	127

a X Fraction of decay heat transferred to sodium

NOTE: INTER results are based on the original interfacial heat transfer coefficients given in Ref. 9. GROWS predictions are based on equal radial and downward heat fluxes.

5. (c) Ex-Vessel Molten Pool Results

The containment response to several ex-vessel core melt scenarios were evaluated assuming the core debris to be in a molten pool form. These calculations were performed by coupling the INTER and CACECO codes. The gas generation resulting from decomposed concrete, as predicted from INTER, was input to CACECO to determine this effect on the containment transient. As previously mentioned, the effect of the gas addition from the decomposed concrete on the containment transient was minimal. This is due primarily to the low CO_2 content of decomposed basaltic type FFTF concrete.

Due to the existence of an unlined subcavity situated below the reactor cavity in the FFTF, it was necessary to approach the problem in two stages. The first stage consisted of a melting attach on the floor of the reactor cavity. The assumption was made that the floor of the cavity would fail when the melt front arrived within 6 inches of the subcavity (30 inches into the cavity floor mat). This criterion was used to determine (from the appropriate INTER computer code runs) the point at which the first stage terminated and the second stage began. The second stage consisted of a melting attack of the molten pool on the subcavity floor together with a sodium chemical attack on the vertical concrete walls of the subcavity.

A parametric study was performed by assuming various fractions of the total decay heat to be transmitted downward in the melting attack. The previously discussed case of an intact liner was taken as the zero-downward-heat-flux case (i.e., all of the decay heat goes directly into heating up the sodium). In addition, two cases, one assuming 50 percent downward heat transfer and the other assuming 100 percent downward heat transfer, were run. The appropriate heat split was specified to the INTER code which calculated the melt front progression, the gas generation, and the secondary upward heat transport (via the ejected gases). It was determined that the gases generated from the decomposing concrete are independent of the growth pattern and dependent only on the upward/downward heat split. Therefore, the gas generation rate predicted by INTER could be related to a simplified version of the GROWS code. In the case where 100 percent of the heat was directed downward, the cavity floor was penetrated within 13 hours, and for the 50 percent case, it was penetrated in 32 hours; for both of these cases, the core melt debris was assumed to be spread over 50 percent of the floor area directly between the guard vessel support skirt, and the results are based on the GROWS code assuming that the radial and downward heat fluxes are equal.

A summary of the ex-vessel containment response results is presented in Table 15.3-6. The important input parameters are again included together with the times required to reach the reactor containment building limits. The first two cases shown in Table 15.3-6 assume, after melt-through, that the molten fuel/steel mixture fragments in the reactor cavity and forms a self-leveling, sodium-flooded, coolable debris bed. The first case assumes the liner to remain intact, whereas the second case assumes the liner to fail over an area of 346 ft^2. For the coolable debris bed, intact liner case, a best estimate of the time-to-sodium boiling is 24 hours, resulting in the reactor containment building reaching its design pressure in 80 hours. For the failed liner case, the containment pressure limit of 10 psig is reached in 45 hours for an effective sodium-concrete penetration rate of 1 in/hr (see Appendix C). The next two cases shown in Table 15.3-6 are included for completeness. The purpose of analyzing these cases was primarily to illustrate the effect of varying the differential pressure (blowout criterion) necessary to actuate communication between the reactor cavity and the reactor containment building. The results indicate that the times to reach the reactor containment building pressure and H_2 limits are relatively independent of the blowout criterion. The higher the differential pressure before the blowout, the more severe is the pressure transient in the reactor containment building at blowout; however, the longer-term effect is apparently minimal. The last two cases shown in Table 15.3-6 assume the core debris to remain in a molten pool form after melt-through. The molten pool/concrete interactions were analyzed using the GROWS code, which predicted failure of the reactor cavity floor after 32 hours if 50 percent of the decay heat is assumed to be transferred to the sodium (a = 0.5) and after 13 hours if there is negligible heat transfer to the sodium (a = 0). These penetration rates were based on an initial pool geometry where the melt was spread over 50 percent of the floor area inside the guard vessel support skirt. It should be noted that for the two molten pool configurations considered, the reactor containment building hydrogen concentration limit was reached before the pressure limit. This is due primarily to the large quantity of water released from the decomposing cocrete in the reactor cavity. Apparently, the rapidity with which the cavity floor mat is penetrated in the most significant consideration in the containment transient. This is due to the overwhelming effect of the sodium-concrete reaction on the sodium heating and vaporization rates. When the cavity floor concrete is penetrated, sodium is allowed to contact the bare concrete in the subcavity resulting in an acceleration of the sodium heating rate. The greater the downward heat flux the sooner will this penetration occur and the sooner will the sodium-concrete reaction come into play. The times at which the containment pressure reaches 10 psig for the 0 percent

TABLE 15.3-6

SUMMARY OF EX-VESSEL CASES

REACTOR VESSEL TO RCB VENT SIZE (INCHES)	ACTIVATION (PSID)	FRACTION OF DECAY HEAT TRANSFERRED TO SODIUM (a)	REACTOR CAVITY LINER CONDITION	MOLTEN POOL PENETRATION OF CAVITY FLOOR (HOURS)	TIME FOR RCB TO REACH 10 PSIG (HOURS)	TIME TO REACH 4% H_2 IN RCB (HOURS)	RCB SHELL TEMPERATURE AT RCB PRESSURE LIMIT (°F)	REMARKS
36	30	1.0	INTACT	COOLABLE DEBRIS BED	80	80	308	EFFECT OF FAILED LINER, SEE FIGURES 3.9 THROUGH 3.11, Ref. 7
36	0	1.0	FAILED (346 ft^2)		45	40	190	
36	15	1.0	INTACT	COOLABLE DEBRIS BED	65	53	294	EFFECT OF CAVITY TO RCB VENT ACTIVATION, SEE FIGURES 3.12 THROUGH 3.14 Ref. 7
36	0	1.0	INTACT		65	53	294	
36	30	0.5	MELT	32	40.	9.	300	EFFECT OF POOL HEAT TRANSFER, SEE FIGURES 3.15 THROUGH 3.17, Ref. 7
36	30	0.0	ATTACK CONCRETE*	13	25	3	170	

*Melt assumed to spread over 50% of the reactor cavity floor between the guard vessel support skirt.

down, 50 percent down and 100 percent cases were 80, 40, and 25 hours, respectively. It was found that it takes about 10 hours after the reactor cavity floor is penetrated for the containment pressure to reach 10 psig.

6. Staff Summary of FFTF Containment Margins

In comparing the ex-vessel and in-vessel containment margin results, the ex-vessel accident conditions are usually more severe or limiting. Therefore, the radiological consequences and conclusions on the adequacy of FFTF containment margins will be based on the ex-vessel core melt scenarios.

The ex-vessel FFTF containment margins evaluations discussed briefly above and reported in more detail in Appendix F and Ref. 7 are considered to envelop the range of outcomes of possible core melt scenarios. Clearly, some of the cases are considered to represent upper and lower bound estimates. By taking into account the uncertainties that exist in ex-vessel core melt scenarios involving either coolable debris beds coupled with sodium-concrete interactions (see Appendix C), or interactions between molten fuel/steel and concrete (see Appendix D), and by selecting reasonable parameters and assumptions for the previously described cases, a summary of our conclusions of FFTF containment margins is presented in Table 15.3-7. We conclude that between 30 to 60 hours should be available in the FFTF before containment would have to be vented to prevent failure by overpressurization. To prevent containment failure by a hydrogen explosion, steps would have to be taken to decrease the H_2 concentration (such as purging with an inert gas) in about 15 to 45 hours provided the core debris exists in a coolable debris bed configuration where only sodium is attacking the reactor cavity floor concrete. This time interval includes consideration of the uncertainty associated with the path (i.e., either to heating and ventilation cooler room or to reactor cavity) which the water vapor (associated with the submerged concrete) takes, as discussed above under subsection 5(a) and depicted in Table 15.3-3. If the core debris exists in a molten pool configuration so that the concrete is being decomposed by a melt-front attack, the production of H_2 is greater than for the coolable debris bed configuration. For the molten pool configuration, we concluded that steps would have to be taken to decrease the H_2 concentration in the time iterval of 10 to 20 hours to preclude a hydrogen explosion. The radiological consequences of having to vent containment in the above time intervals are discussed in the following section.

The Commission is supporting a confirmatory research program directed toward understanding the materials interaction phenomena associated with sodium-concrete reactions and interactions between molten fuel/steel and concrete. In the near future, both small and large scale tests will be performed utilizing the FFTF types of concrete, namely basalt and magnetite. Following the performance of these tests, we will reconsider our position in regard to the penetration rate of

sodium and/or molten fuel/steel into the FFTF reactor cavity concrete and assess its impact on the resulting containment margins. Should further experimental work define a need for augmenting containment margins by providing additional protective provisions, the FFTF containment could be backfitted to provide such provisions.

TABLE 15.3-7

FFTF CONTAINMENT MARGINS SUMMARY

	CORE DEBRIS CONFIGURATION	TIME INTERVAL TO VENT RCB TO PRECLUDE OVER PRESSURIZATION (Hrs)	TIME INTERVAL TO DECREASE RCB H_2 CONCENTRATION TO PRECLUDE H_2 EXPLOSION (Hrs)
1.	Coolable Debris Bed	30 to 60	15 to 45
2.	Molten Pool	30 to 60	10 to 20

15.3.7 Radiological Consequences of Core Disruptive Accidents

The radiological consequences of energetic core disruptive accidents were calculated in our previous Safety Evaluation Report and its supplements. In view of the present design features, physics, and accident calculations, it has been concluded that calculated energetics from core disruptive accidents are within the capability of the containment system. It is expected that in-vessel cooling of the core damaged from these core disruptive accidents will be sufficient to prevent the failure of the primary coolant boundary or melting of the vessel. Natural convection is the mechanism finally relied upon to provide decay heat removal. Although the calculations supporting the natural convection mechanism have been extensive, there remains a degree of uncertainty about the reliability of this mechanism, and there is no apparent way to completely resolve this uncertainty. Accordingly, the possible consequences of a meltdown accident in which in-vessel cooling was assumed to be insufficient to prevent the failure of the bottom of the reactor vessel have also been considered. In the later stages of this accident, venting or purging of the containment building may be necessary to preserve its integrity as a structure.

The FFTF design has some capability for venting and purging, but the capability should be reviewed by the Project to determine if it is suitable for the ex-vessel core melt accident conditions. Instrumentation is available to the operators on the need for such action should this low probability situation occur. (However, see item (9) in Section 7.6 of this report on the desirability of upgrading this instrumentation to safety-grade.)

The previous sections have reviewed the evidence leading to the conclusion that early catastrophic failure of the containment is an unlikely outcome of these low probability accidents. This section will focus on the radiological consequences of venting or purging measures required assuming that vessel melt-through occurs.

Initial scoping calculations of the consequences of venting of the containment building assumed an instantaneous puff release of the entire inventory of radioactive suspended aerosol existing at that moment in the building. This was the basis for earlier estimates that, if venting took place before about 60 to 100 hours after an accident, significant radiological consequences could result (using the dose guidelines in 10 CFR Part 100 as a guideline).

Present calculations have examined the effect of a more gradual venting or purging process that can take advantage of simultaneous fallout of part of the radioactive aerosol within the building. This has led to lower dose estimates than the previous puff releases. The reason for this is related to the relative rates of decline of the inventory of radioactive aerosol due to fallout and due to venting. Venting at rates sufficient to control the hydrogen concentration or to prevent excessive pressure buildup involves the continuous release of the containment building atmosphere at a typical rate corresponding to 400% per day. The fallout of aerosol material from the dense cloud following a violent sodium fire is calculated to correspond to 2000% to 4000% per day, and is the predominant mechanism for reducing the radioactive content suspended in the containment when the vent rates are in the 400% per day range. In our judgment, a reasonably conservative calculation of the radiological effects of venting or purging would model the venting at 400% per day at the time when the inventory of radioactive aerosol is at its maximum, since it is close to this time that the pressure or hydrogen concentration would require that some corrective measures be taken. Under this modeling, fallout will reduce the radiological consequences below those of the early puff releases modeled previously.

The radiological effects of meltdown accidents have been estimated from a series of CACECO,[1] HAARM 3[2] and TACT 4[3] calculations performed independently by the staff. Guided by the release fractions in the Reactor Safety Study, WASH-1400, and using mechanistic assumptions where justifiable, we have made the calculations somewhat realistic, although still on the conservative side. CACECO calculates the pressure, temperatures, and atmospheric composition in various compartments of the reactor building based upon assumed reaction rates between sodium and concrete and other chemical reactions among the basic reactor materials and their products. HAARM 3 calculates the behavior (agglomeration and fallout) of aerosols in the reactor containment building. The principal component of these aerosols is a mixture of oxides of sodium. With reasonable assumptions about the behavior of various classes of fission products, the aerosol behavior described by HAARM 3 can predict the behavior of radioactive species suspended with the aerosol. TACT 4 uses ORIGEN[4] generated fission product inventories and other activation inventories, together with fallout rates generated by HAARM 3, to predict the actual release of radioactive species to the environment. TACT 4 also uses meteorological data and dose conversion factors to determine the actual doses at the site boundary, low population zone, or beyond, with no credit for fallout or decay outside the containment building. In this connection. it should be noted that recent experimental

tests of burning sodium in air indicate more than 90% of the material was deposited within 100 meters of the source.[5] Although these results should not be applied quantitatively to the FFTF situation, they suggest a fallout factor of significant magnitude.

The CACECO studies have been based upon scenarios that begin with a reactor excursion and meltdown. The energetics of the excursion are such that little or no core materials escape in an upward path through the head seals to the reactor containment building at the time of the excursion. The core geometry is destroyed, however, and core materials generating intense decay heat fall to the bottom of the vessel. The in-vessel coolability of this bed of core debris is uncertain, but it is reasonable to assume that the decay heat results in failure of the vessels after about three hours. The entire inventory of primary sodium and core debris then falls to the floor of the reactor cavity. This cavity is steel-lined and filled with an inert atmosphere. The core debris decay heat or thermal shock could then be sufficient to fail the steel liners over a substantial area, exposing the underlying concrete to sodium and core materials.

A series of reactions then commences between the sodium, the concrete and the various cell and building atmospheres. The heart of the CACECO code is its calculation of the energy and material balances among the cells. The energy and gases released are sufficient to open a pathway in a short time to the reactor containment building, allowing sodium vapors that escape from the cavity to be oxidized in the air atmosphere in to air-suspended particulates (aerosols) of oxides of sodium. Water released from the concrete reacts with sodium and sodium oxides to form hydrogen and sodium hydroxide. The penetration of sodium into the concrete thus results in a significant but manageable buildup of pressure and hydrogen concentration in the reactor containment building. In these radiological studies, which are intended to be realistic, it is regarded that pressures in the range of 10 to 20 psig, or hydrogen concentrations beyond the range of 4 to 6 volume percent, would represent a serious challenge to containment. If the sodium penetration proceeds far enough to cause the 36-inch thick floor of the reactor cavity to fail, the sodium and core debris then enter the unlined subcavity. Here the concrete surface available for reaction is so great that the rate of reactions increases markedly, and the containment building pressure or hydrogen concentration may increase rapidly to the point that the containment may fail by overpressure (unless the containment is vented). Data indicate that the buildup of hydrogen may control itself temporarily through the slow reaction of hydrogen with oxygen in the presence of sodium, but this is not certain enough yet to be relied upon.

Clearly, the ability to maintain containment integrity for a significant length of time under such a sencario will be dependent upon the rate of reaction between sodium and concrete. The CACECO calculations indicate that at penetration rates of one-half inch per hour, as observed in some tests at HEDL, the containment would not be challenged for at least 45 hours. If the penetration rate is three

inches per hour, as observed in some Sandia tests[6] and in the F-1 test at HEDL,[7] the containment building would be threatened either by the buildup of pressure or hydrogen concentration in 10 to 30 hours.

The CACECO calculations for the radiological estimates discussed below used a concrete penetration rate of one inch per hour. We believe, in view of the data now available from Sandia and HEDL, that a penetration rate of sodium into concrete averaging one inch per hour would be a conservative value from among the spectrum of experimentally measured rates.

The CACECO calculations furnish a rate of boil-off of sodium from the pool as a result of the decay heat and the heat of the various chemical reactions. A substantial part of the vapor boiled off is introduced into the reactor containment building where it becomes an oxidized aerosol whose behavior can be calculated by HAARM 3. At the rate of boil-off corresponding to the 1 inch per hour reaction rate, approximately 7% of the sodium pool is boiled off in 40 hours, at a fairly uniform rate of 130 gm per second. The production of aerosol at this rate results in the achievement of a steady-state situation after about 7 hours, such that the rate of fallout of aerosol is equal to the rate at which the aerosol is introduced. The mass of suspended aerosol then remains constant, equivalent to about 0.4% of the original sodium pool. This equilibrium is maintained until the molten pool melts through the cavity floor into the subcavity at about 40 hours. The increased reaction surface area presented by the subcavity increases the rate of reaction about tenfold, so that sodium is then boiled off at a rate of about 1300 gm/sec. A new secular equilibrium is reached in about three more hours, so that the mass of sodium oxide in suspension at any moment then represents about 1.5% of the original sodium pool. Sixty hours after the accident, all the oxygen in the building is used up. At this point it is not clear what will be the effect of the further boil-off of sodium.

We have assumed that continued evaporation of sodium into the inert atmosphere would produce an aerosol of condensed sodium droplets behaving much the same as the sodium oxide. In this case, the containment would have been threatened by the buildup of hydrogen approaching 6% at about 40 hours (if hydrogen did not recombine) and by the buildup of pressure to 20 psig at about 60 hours. After all the sodium has evaporated, the fallout of the remaining aerosol is calculated to be quite rapid, about a factor of 100 in three hours. As the evaporation of the final half of the sodium pool takes place, between 66 and 96 hours after the accident, some of the remaining fission products and fuel may be released, possibly by mechanical action and turbulence at the pool surface.

In considering the radiological consequences of these meltdown events, it has been convenient to classify the fission products into major groupings based upon their volatility and chemical activity (Reactor Safety Study, Appendix VII). The first group, the noble gases xenon and krypton, would be released either at the time of

the excursion or at the time of vessel melt-through, three hours later. Since they are not subject to chemical combination or fallout, the time of release makes little difference, except for very short-lived isotopes. The next group would comprise those fission products that are soluble in sodium and more volatile than sodium. Cesium and rubidium are in this category. They can be expected to be released at a rate faster than the rate of boil-off of sodium. For practical purposes, one can assume that by the time 10% of the sodium has boiled away, essentially all of the cesium and rubidium will have been released to the containment atmosphere, where they will combine with aerosol particles which govern their subsequent fallout and leakage behavior.

The halogens are considered to form soluble compounds (NaI, for example) which are less volatile than sodium. There may be some of the volatile free iodine released at the time of the excursion, but the balance would form iodine salts in the sodium pool. The free iodine and the sodium halides would be subject to fallout, like other aerosol materials. Other less volatile groups of fission products are not expected to contribute significantly to the doses from airborne materials because of the difficulty in getting them volatilized or otherwise airborne. One percent of the nonvolatiles were considered to be carried upward with the sodium vapor. Two methods for estimating release of fuel from sodium pools have been suggested by different authors. Cybulskis[8] has regarded it as a molten material, subject to sparging by the boiling sodium vapors to the extent of 1%. The FFTF Project contractors have regarded it as a fractured solid material but with only a small percent of its mass in the form of particles less than a few microns in diameter.[9] These small particles would be sparged from the pool by the boiling sodium vapor, but subject to a cleanup factor of the order of 1000, as determined by Jordan and Ozawa.[10] The staff has preferred to use the particle size distribution as determined by Armstrong[11] which places less than 10 percent of the plutonium in the dispersible size particles (F 30mm diameter), but to reduce to 100 the possible cleanup factor of Jordan and Ozawa, since it is not known how this cleanup mechanism behaves at the end of the boilup phase. This reduction in the Jordan-Ozawa cleanup factor is also expected to provide for the possible formation of soluble or volatile sodium-plutonium compounds. Thus, in the staff calculations, 0.1% of the core plutonium was released to the aerosol in the containment building at the end of the sodium boilup phase. This material was subject to fallout with the sodium oxide aerosol.

Inspection of the CACECO and HAARM calculations led to the conclusion that the radioactive consequences of these scenarios would depend upon the leakage and the timing and amount of venting necessary. If no ventng is required, as would be the case with 1 inch per hour sodium-concrete penetration rate with hydrogen recombination and no subcavity penetration, the doses at the low population zone are low, and are of the same order of magnitude as that calculated for the most severe design basis accident. If venting is assumed in the above model, and if venting is arranged to occur at times when the suspended concentration of radioactivity is

not high, the consequences are still not severe relative to other core melt and loss of containment analyses, as in WASH-1400. It is believed that such a venting schedule should be investigated.

The pressure in the reactor containment building could be controlled temporarily, and at least partial control over fission product releases could be maintained. As the sodium boil-off rate increased due to the increased reaction surface in the subcavity, substantial amounts of radioactivity would have to be released with additional dose consequences. If containment could be maintained until a few hours after all the sodium has boiled off, the airborne activity would be greatly reduced by fallout within the containment, and subsequent venting could be accomplished without excessive releases.

Based upon this summary of conclusions drawn from inspection of the CACECO and HAARM calculations, a model was developed for detailed radiological calculations with the TACT code. It was apparent that the greatest release of radioactve material to the containment building would take place toward the end of the sodium boil-off phase. It was also apparent that venting or purging of the containment might be required at about the same time that the concentration of radioactive material in the containment building atmosphere was very high. It appeared therefore that the synchronization of the venting with the completion of the sodium boil-off would have a greater bearing on the radioactive consequences than the actual time of venting itself. What was not quite so obvious was the effect of continuing fallout of the aerosol within containment during the purging or venting.

The TACT code was therefore set up to include the rate of fallout of the aerosol in containment as a depletion mechanism proceeding at the same time as leakage or purging from the containment to the environment. A vent rate of 4000 cfm was chosen, as this is enough to relieve the pressure, and, with subsequent purging at 4000 cfm, to maintain an inert atmosphere. After an initial period (24 hours) of increasing boiling, during which noble gases and volatiles were released from the pool with small amounts of less volatile materials, 1% of the inventory of radioactive materials, including 0.1% of the core plutonium which is less volatile than other fission product solids, was suddenly released to the containment building atmosphere. At the same time, the leak rate from the building increased from 0.1% per day to 400 volume percent per day, corresponding to a 4000 cfm purge or vent.

In view of the fact that under the circumstances of this model the halogens would have ample opportunity to combine with sodium aerosol products, a fallout factor of 0.2 has been included in the iodine doses at the low population zone (7200 m) to account for the fact that no more than this fraction of the aerosol would escape fallout before arriving at the receptor. The corresponding factor at the exclusion radius (2400 m) was 0.5.

The radioactive dose results for an individual at the exclusion radius or at the low population zone radius are given in Table 15.3-8. Although the calculation is believed to cover an extremely unlikely combination of circumstances associated with an accident at FFTF, the consequences are not as severe as might have been expected. This is because the fallout within containment is a dominating effect during the time when the suspended radioactivity is at its highest. Eighty percent of the large doses from plutonium and solids results from the first half-hour after the major release from the pool at 24 hours. This suggests that rather temporary interdiction measures could be very effective in protecting downwind populations. Table 15.3-8 also shows the low-level doses that would be produced if no venting were required, or if venting could be delayed until 16 hours after the major release of fission products from the pool (40 hours from the start of the accident).

The FFTF Project has calculated a similar level of doses in various cases using some of the assumptions suggested by the staff in comparison with variants that they have proposed as being more probable.

The release fractions used here are comparable to those used in the analysis of light water reactors for similar low probability accidents, and the fission product inventories are lower for FFTF than for power reactors by a factor of 8. The mass of plutonium in FFTF is comparable to that in a light water moderated power reactor at the end of a cycle.

We conclude that the consequences of an FFTF core melt accident can be held to a level that is not out of line with those of other reactors from similar low probability accidents, provided that attention is paid to the management of releases from the containment, and preparations for these accidents are kept in readiness.

More recent CACECO calculations have indicated that, under the conditions chosen, venting could probably be delayed until about 36 hours after the start of the accident. Except for a slight decrease in dose due to radioactive decay of some isotopes, the doses listed in Table 15.3-8 would be applicable to this case also, as long as venting was limited at 4000 cfm at the time when the radioactivity in the containment building atmosphere was at its maximum.

The radiological analyses to this point have assumed that the core debris would take the form of a coolable bed of particulate matter. In the previous section a study was made also of the course of events if the core debris became a molten pool. The results of this study indicated that less sodium vapor was transferred to the reactor containment building in the molten pool case than in the debris bed case. This is apparently because of the effectiveness of the concrete in extracting heat from the molten pool, leaving less heat for sodium vaporization. Since the carryover of fuel and fission products to the reactor containment building depends upon the sodium transmitted to the reactor containment building, it is believed

TABLE 15.3.8

FFTF MELTDOWN ACCIDENT DOSES, REM

	No Venting		Vent After 24 Hours	Vent After 40 Hours
	2 hr Excl. Rad	30 d LPZ*	4000 cfm 30 d LPZ*	10,000 cfm 30 d LPZ*
I - thyroid	10	3	180	3
Solids - bone	negligible	.03	27	.11
- lung	negligible	.04	44	.17
Pu - bone	negligible	.03	45.0	.16
- lung	negligible	.001	1.8	.006
Noble gases - WB	.06	.04	2.6	1.7
- skin	.13	.12	12	8.8

Assumptions:

100 percent of noble gases released to RCB at time zero.

25 percent of iodines and 1 percent of solid fission products released to RCB at time zero.

Leak rate is 0.1 percent per day, until vent time.

At 24 hours, 1 percent of the remaining fission products and 0.1 percent of plutonium released to RCB.

At vent time, leak rate is increased to 400 percent per day (4000 cfm) or larger.

All solids (including halogens) undergo fallout within the RCB, along with Na oxides. No fallout outside containment.

TACT calculations show that:

80 percent of the doses associated with the 24-hour venting result from the material vented in the first half-hour of venting.

*LPZ = outer boundary of Low Population Zone.

that the doses from the molten pool case would be less than from the debris bed case calculated here.

It has been calculated that the molten core debris might lead to a more rapid buildup of hydrogen in the reactor containment building than if the debris were coolable as a bed of solid fragments. This could lead to a requirement for early venting of the containment building. If one assumes that the entire core forms a molten pool, venting could be required at 10 to 20 hours after the start of the accident. The suspensions of the more volatile fission products cesium and rubidium could be at their maximum at this time, about 5 to 10 percent of their total inventory. If the hydrogen ignited, requiring venting at this time, the cesium bone doses at the boundary of the low-population zone are calculated to be 20 to 40 rem. The doses from plutonium and other solids would be negligible in this case because these materials do not become significantly suspended as aerosols until much later in the boil-off sequence.

The staff concludes that the risks associated wth a low probability reactor vessel melt-through are acceptably low.

References for Section 15.1

1. Fast Flux Test Facility, Final Safety Analysis Report, Westinghouse Hanford Co., December 1975.

2. FFTF Fuel Pin Final Design Support Document, Issued December 10, 1971; revised February 28, 1972, R. Sim and A. Veca.

3. HEDL-TC-556, "Simulation of the Overall FFTF Plant Performance (IANUS-Westinghouse Proprietary Code), SL Addition, T. B. McCall and C. F. Wolfe, December 1975.

References for Section 15.3.1

1. Fast Flux Test Facility, Final Safety Analysis Report, Westinghouse Hanford Company, December 1975.

2. Wilburn, N. P. et al, "An Updated Assessment of the Unprotected Transient Overpower Accident in the FTR," HEDL-TI 7585, October 1977.

3. Fast Flux Test Facility, Preliminary Safety Analysis Report, WADCO Corporation, September 1970.

4. Culley, G. E. et al, "Fast Reactor Safety Implications of Recent Assessments of Fuel Pin Transient Behavior, "International Conference on the Engineering of Fast Reactors for Safe and Reliable Operation, Kernforschungszentrum Karlsruhe, October 9-13, 1972, (pp. 626-641).

5. "Safety Evaluation of the Fast Flux Test Facility," USAEC, Directorate of Licensing, Washington D. C. October 31, 1972.

6. Supplement No. 1 to the "Safety Evaluation of the Fast Flux Test Facility," USAEC, Directorate of Licensing, Washington D. C. December 13, 1974.

7. Meyer, J. F. et al, "An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Accident Energetics," NUREG-0122, NRC, Washington, D. C. March 1977.

8. Waltar, A. E. et al, "Melt-III, A Neutronics, Thermal Hydraulics Computer Program for Fast Reactor Sfaety Analysis," HELD-TME 74-47, December 1974.

9. Dunn, F. E. et al, "The SAS3A LMFBR Accident Analysis Computer Code," ANL/RAS 75-17, April 1975.

10. Rumble, E. T., "A Hypothetical Overpower Excursion Model for LMFBR's," Ph. D. Dissertation in Engineering, UCLA, 1974.

11. Page, R. J. et al, "Preliminary Data Report: TREAT H 6," ANL/RAS 77-55, December 1977.

12. Koopman, R. N. et al, "TREAT Transient Overpower Experiment R12," Transactions ANS, Vol. 28, June 1978 (pp. 482).

13. Spencer, B. W. et al, "Camel TOP/Fuel Sweepout Single-Pin Test C2," ANL/RAS 77-22, July 1977.

References for Section 15.3.2

1. "Safety Evaluation of the Fast Flux Test Facility," USAEC, Directorate of Licensing, Washington, D. C., October 31, 1972.

2. Fast Flux Test Facility, Preliminary Safety Analysis Report, September 1970.

3. Bohl, W. et al, "A Preliminary Study of the FFTF Hypothetical Flow Coast Down Accident," ANL/RAS 71-39, Revision 1, April 1971.

4. Dunn, F. E. et al, "The SAS2A LMFBR Accident Analysis Code," Proc. Conf. on New Developments in Reactor Mathematics and Applications, CONF-710302, March 1971.

5. Stevenson, M. G. et al, "Report on the Analysis of the Initiating Phase of a Loss-of-Flow (Without Scram) Accident in the FTR," ANL/RAS 74-24, Argonne National Laboratory, September 1974.

6. Jackson, J. F. et al., "Report on the Core Disruption Phase of an Unprotected Flow-Coast-down in the FTR," ANL/RAS 74-16 Argonne National Laboratory, August 1974.

7. Supplement No. 1 to the Safety Evaluation of the Fast Flux Test Facility," USAEC, Directorate of Licensing, Washington, D. C., December 13, 1974.

8. Dunn, F. E. et al, "The SAS3A LMFBR Accident Analysis Computer Code," ANL/RAS 75-17, April 1975.

9. Fauske, H. K. and Ferguson, D. R., "An Updated Assessment of the Unprotected Loss-of-Flow Accident in the FTR," ANL/RAS 77-42, October 1977.

10. Meyer, J. F. et al, "An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Energetics," NUREG-0122, Nuclear Regulatory Commission, Washington, D. C., March 1977.

11. Miao, C. and T. G. Theofanous, "Transient Two-Dimensional Sodium Boiling in LMFBR Subassemblies: The HEV-2D Code," PNE 78-130, Purdue University (1978).

12. Kikuchi, Y. and K. Hoga, "Transient Boiling of Sodium in a 19-Pin Bundle Under Loss-of-Flow Conditions," JAPFNR-262, June 1977.

13. Theofanous, T. G. et al, "Boiling Incoherency Effects in Unprotected Loss-of-Flow Accidents," ANS Trans. 27, 537 (1977).

14. Bohl, W. R. and T. J. Heames, "CLAZAS: The SAS3A Clad Motion Model," ANL/RAS 74-15, Argonne National Laboratory, August 1974.

15. Theofanous, T. G. et al, "Incoherency Effects in Clad Relocation Dynamics for LMFBR CDA Analyses," Nuclear Eng. Design, Vol. 36, No. 1, January 1976.

16. Henry, R. E. et al, "28-Pin Cladding Relocation Experiments," ANS Trans. 27, 498 (1977).

17. Fauske, H. K, "Some Comments on Cladding and Early Fuel Motion," (Interim Report), ANL/RAS 76-11, Argonne National Laboratory, March 1976.

18. Palm, R. G. et al, "F1 Phenomenological Test on Fuel Motion," (Interim Report), ANL/RAS 76-11, Argonne National Laboratory, March 1976.

19. Deitrich, W. L. and R. W. Ostensen, "An Assessment of Fission-Gas-Driven Fuel Disruption and Dispersal in a Hypothetical LMFBR Loss-of-Flow Accident," ANL/RAS 77-4, February 1977.

20. Cantley, D. A. et al, "Axial Expansion at Mixed Oxide Fuel Measured In-Pile," GEAP-10403 (1971).

21. Theofanous, T. G., "Multiphase Transients with Coolant and Core Materials in LMFBR Core Disruptive Accident Energetics Evaluation," NUREG/CR-0224, USNRC, July 1978.

References for Section 15.3.3

1. "Safety Evaluation of the Fast Flux Test Facility," USAEC, Directorate of Licensing, Washington, D. C., October 31, 1972.

2. Jackson, J. F. et al, "Report on the Core Disruption Phase of an Unprotected Flow-Coastdown in the FTR," ANL/RAS 74-16, Argonne National Laboratory, August 1974.

3. Stevenson, M. G. et al, "Report on the Analysis of the Initiating Phase of a Loss-of-Flow (Without Scram) Accident in the FTR," ANL/RAS 74-24, Argonne National Laboratory, September 1974.

4. Supplement No. 1 to the "Safety Evaluation of the Fast Flux Test Facility," USAEC, Directorate of Licensing, Washington, D. C., December 13, 1974.

5. Fast Flux Test Facility, Final Safety Analysis Report, Westinghouse Hanford Co., December 1975.

6. Fauske, H. K. and Ferguson, D. R., "An Updated Assessment of the Unprotected Loss-of-Flow Accident in the FTR," ANL/RAS 77-42, October 1977.

7. Dunn, F. E. et al, "The SAS3A LMFBR Accident Analysis Computer Code," ANL/RAS 75-17, April 1975.

8. Walter, A. E. et al, "MELT-III, "A Neutronics, Thermal Hydraulics Computer Program for Fast Reactor Safety Analysis," HEDL-TME 74-47, December 1974.

9. Meyer, J. F. et al, "An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Energetics," NUREG-0122, Nuclear Regulatory Commission, Washington, D. C., March 1977.

10. Fauske, H. K. and D. R. Ferguson, "An Updated Assessment of the Unprotected Loss-of-Flow Accident in the FTR," ANL/RAS 77-42, October 1977.

11. Theofanous, T. G., "Multiphase Transients with Coolant and Core Materials in LMFBR Core Disruptive Accident Energetics Evaluation," NUREG/CR-0224, USNRC, July 1978.

12. Ostensen, R. W. and R. J. Henninger, "Recriticality in the Transition Phase in CMFBR Accidents," ANS Trans. 24 346 (1975).

13. Boudreau, J. E. and J. F. Jackson, "Recriticality Considerations in LMFBR Accidents," CONF-740401, 1265 (1974).

References for Section 15.3.4

1. Waltar, A. E. et al, "MELT-III, A Neutronics, Thermal-Hydraulics Computer Program for Fast Reactor Safety Analysis," HEDL-TME 74-47, December 1974.

2. Dunn, F. E. et al, "The SAS2A LMFBR Accident-Analysis Computer Code," ANL-8138 October 1974.

3. Stevenson, M. G. et al, "Report on the Analysis of the Initiating Phase of a Loss-of-Flow (Without Scram) Accident in the FTR," ANL/RAS 74-24, December 1974.

4. Jackson, J. F. and R. B. Nicholson, "VENUS-II, An LMFBR Disassembly Program," ANL-7951, September 1972.

5. Jackson, J. F. et al, "Report on the Core Disruptive Phase of an Unprotected Flow-Coastdown Accident in the FTR," ANL/RAS 74-16, August 1974.

6. Padilla, A., "Transient Analysis of Fuel-Sodium Interactions," Trans. Am. Nucl. Soc. 13 375 (1970).

7. Florence, A. L. and G. R. Abrahamson, "Simulation of a Hypothetical Core Disruptive Accident in a Fast Flux Test Facility," HEDL-SR-1, May 1973.

8. Stepnewski, D. D. and C. R. Adkins, "Final Report of Task Force for Assessment of Additional FFTF Design Margin," WARD-317, August 8, 1974.

9. Cho, D. H. and M. Epstein, "Work Potential Resulting from a Mechanical Disassembly of the Voided FFTF Core," ANL/RAS 74-17, August 1974.

10. Change, Y. W. and J. Guildys, "REXCO-HEP" A Two-Dimensional Computer Code for Calculating the Primary System Response in Fast Reactors," ANL-75-19, June 1975.

11. Meyer, J. F. et al, "An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Accident Energetics," NUREG-0122, March 1977.

12. Graham, J., "HCDA Requirements and Loadings for FFTF Design Evaluation," FS-257, Revision 1, July 1976.

References for Section 15.3.6

1. "Containment Margins in FFTF for Postulated Failure of In-Vessel Post Accident Heat Removal," HEDL-TME 77-18, April 1977.

2. "FFTF Secondary Sodium Fire Protection System Test FI," HEDL-TME 73-48, April 1973.

3. Memorandum to T. P. Speis, LMFBR Branch, NRR, from M. Silberberg, Chief, Experimental Fast Reactor Safety Research Branch, RES, dated August 5, 1977, "Quick-Look Report on Sandia Preliminary Sodium-Concrete Interaction Tests 1 and 2."

4. Memorandum to T. P. Speis, Chief, LMFBR Branch, NRR, from M. Silberberg, Chief, Experimental Fast Reactor Safety Research Branch, RES, October 31, 1977, "Quick Look Report on Sandia Preliminary Sodium-Concrete Interaction Test 3."

5. Peak, R. D. and D. D. Stepnewski, "Computational Containment Analysis Code," ANS Trans., Vol. 21, p. 274, June 1975.

6. Baker, L. Jr. et al, "Post-Accident Heat Removal Technology," ANL/RAS 77-2, dated January 1, 1977.

7. Gasser, R. D. and W. T. Pratt, "Containment Response to Postulated Core Meltdown Accidents in the Fast Flux Test Facility," BNL-NUREG 24141-R, Revised August 1978, Published March 1978.

8. "Post-Accident Heat Removal Assessment for the FFTF," HEDL-TME 75-60, November 1974.

9. Murfin, W. A., "A Preliminary Model for Core/Concrete Interaction," Sandia Laboratories, SAND 77-0370, August 1977.

10. "Experimental Study of Hydrogen Formation and Recombination Under Postulated LMFBR Accident Conditions," HEDL-TC-730, December 1976.

11. "Intermediate Scale Sodium-Concrete Reaction Tests," HEDL-TME-77-99, August 1977.

12. For a description of the GROWS Code see Chapter V of: R. Kumas, L. Baker, Jr., and M. G. Chasanov, "Ex-Vessel Considerations in Post-Accident Heat Removal," ANL/RAS 74-29, October 1974.

13. Powers, D. A. et al, "Exploratory Study of Molten Core Material/Concrete Interactions July 1975 - March 1977," Sandia Laboratories, SAND 77-2042, February 1978.

References for Section 15.3.7

1. Stepnewski, D. and R. Peak, CACECO Code Modifications, in HEDL Quarterly Technical Report, October-December 1974, HEDL-TME 74-4, Vol. 3 (1974). See also R. D. Peak and D. Stepnewski, Trans. Am. Nucl. Soc. 21 p. 274 (1975).

2. Gieseke, J. A., K. W. Lee, and L. D. Reed, HAARM 3 Users Manual, BMI-NUREG-1991 January 5, 1978.

3. USNRC In-House Calculations of Radiological Effects of Accidents Using TACT-4, Unpublished.

4. Bell, M. J., ORIGEN, the ORNL Isotope Generation and Depletion Code, ORNL-4628, (1973).

5. Memorandum to L. G. Hulman, Chief, Hydrology-Meteorology Branch, DSE, from R. Jubach, "Trip Report to INEL for Sodium Combustion Tests," June 28, 1978.

6. Memoranda to T. P. Speis, Chief, LMFBR Branch, NRR, from M. Silberberg, Chief, Experimental Fast Reactor Safety Research Branch, RES, "Quick Look Reports on Sandia Preliminary Sodium-Concrete Interaction Tests," 1 and 2, August 5, 1977, and 3, October 31, 1977.

7. FFTF Secondary Sodium Fire Protection System Test F-1, HELD-TME 73-48, April 1973.

8. Cybulskis, P., "Effect of Engineered Safety Features on the Risk of Hypothetical LMFBR Accidents," Nuclear Safety 19, 2, p. 190, March-April 1978.

9. Containment Margins in FFTF for Postulated Failure of In-Vessel Post-Accident Heat Removal, HEDL-TME 77-18, Appendix D, April 1977.

10. Jordan, S. and Y. Ozawa, "Fuel Particle and Fission Product Release from LMFBR Core Catcher," CONF-761001, October 1976.

11. Armstrong, D. R., F. J. Testa and D. Raridon, "Molten UO_2-Sodium Dropping Experiment," Trans. Am. Nucl. Soc. 13, p. 660, November 1970.

16.0 TECHNICAL SPECIFICATIONS

Current light water reactor practice is to incorporate technical specifications as a condition of licensing and to use, insofar as possible, standard technical specifications for pressurized water reactors and boiling water reactors.

The Fast Flux Test Facility is unique in that no previous operating experience exists to rely upon for setting safety limits, control system setpoints, limiting conditions for operation, surveillance requirements or to establish an appropriate degree of reliance on design features and administrative controls. The design parameters used in the Final Safety Analysis Report, Sections 4.2, Fuel System Design; 4.3, Nuclear Design; 4.4, Thermal Hydraulic Design; Section 7.0, Instrumentation and Controls; and Section 15.0, Accident Analyses, considered steady-state conditions but the transition to natural convection is derived from a basic safety concept of not allowing sodium to boil in the core. The staff interprets this concept in a reasonably conservative manner to correspond to a maximum allowable sodium saturation temperature of 1670°F at the exit of the core.

Many of the reactor safety computations in Chapter 15.0 of the Final Safety Analysis Report are derived from codes such as IANUS and FLODISC. Until these codes are verified by testing, including natural convection, they represent best estimate values with some degree of conservatism. The setpoints for the plant protection system should be modified as experience is gained with the reactor.

We believe that the proposed safety setpoints for the plant control and protection systems of the reactor and main heat transfer system, as described in Section 17.2 of the Final Safety Analysis Report, Table 17.2.2-1, items 1 through 7, will provide adequate protection for initial operation of the reactor which is necessary to verify natural convection and the analytical codes used in the thermal-hydraulic design of the FFTF.

We concur in part with the limitations on these preliminary technical specifications as stated in the preface to Section 17.0 of the Final Safety Anlaysis Report. "These technical specifications have been developed solely on the basis of three loop operation of the FFTF at initial core conditions (600°F coolant inlet) with cores 1 and 2."

We believe however that the demonstration of natural convection as a safe reliable decay heat removal method is essential to the overall safety of the FFTF and would not advise continuous sustained operation of the reactor at full power until this has been done.

We advise the FFTF Project to reassess the thermal-hydraulic and time constants of the reactor and heat transport system after having performed these tests and modify the control and protection system setpoints accordingly.

17.0 QUALITY ASSURANCE

17.1 General

The description of the quality assurance program for the operations phase of the Fast Flux Test Facility is contained in Chapter 16.0 of the Final Safety Analysis Report through Amendment 14 and Supplements 5 and 15. Our evaluation of this quality assurance program is based on a detailed review of this information and discussions with Project representatives from the Department of Energy and the Westinghouse Hanford Company, the operating contractor for the Hanford Engineering Development Laboratory (HEDL). We assessed HEDL's quality assurance program for the operations phase to see if it complies with the requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and the applicable regulatory guidance listed in Table 17.1-1.

17.2 Organization

The organizational structure responsible for the operation of the FFTF and for the establishment and execution of the operations phase quality assurance program is shown in Figure 17.2-1. The President of the Westinghouse Hanford Company (who is also the Director of HEDL) has the responsibility for establishing corporate quality assurance policies, goals, and objectives. He also establishes the responsibilities and authority of the Quality Assurance Department and other HEDL organizations whose work affects quality. A master audit plan matrix, which is used for internal audits, identifies the quality-related responsibilities of each organization.

The Manager of Quality Assurance reports directly to the Director of HEDL and has the responsibility for establishment, effective implementation, and control of the quality assurance program. The Quality Assurance Department, which is shown in Figure 17.2-2, has the authority to identify quality problems; to initiate, recommend, or provide solutions through designated channels; to verify implementation of solutions; and to stop unsatisfactory work. The Manager of Quality Assurance is responsible for assuring that HEDL manuals and instructions provide for complete and adequate quality assurance requirements with sufficient reviews, inspections, and audits by quality assurance personnel to verify the effective implementation of the entire quality assurance program. The resolution of disputes on quality assurance program requirements arising between quality assurance personnel and other department personnel are escalated to appropriate levels of management designated in Figure 17.2-1 for resolution.

TABLE 17.1-1

REGULATORY GUIDANCE FOR QUALITY ASSURANCE

1. Regulatory Guide 1.8, "Personnel Selection and Training," (3/10/71).

2. Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electrical Equipment," (8/11/72).

3. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," (3/16/73).

4. Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants," (3/16/73).

5. Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants," (3/16/73).

6. Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," (6/73).

7. Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel," (8/73).

8. Regulatory Guide 1.64, Rev. 2, "Quality Assurance Requirements for the Design of Nuclear Power Plants," (6/76).

9. Regulatory Guide 1.74, "Quality Assurance Terms and Definitions," (2/74).

10. Regulatory Guide 1.88, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records," (8/74).

11. Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," (4/75).

12. ANSI Standard N45.2.8-1975, "Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants."

13. ANSI Standard N45.2.12, "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants," Draft 3, Rev. 4 (2/22/74).

14. ANSI Standard N45.2.13, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants," Draft 2, Rev. 4, (4/74).

15. ANSI Standard N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants."

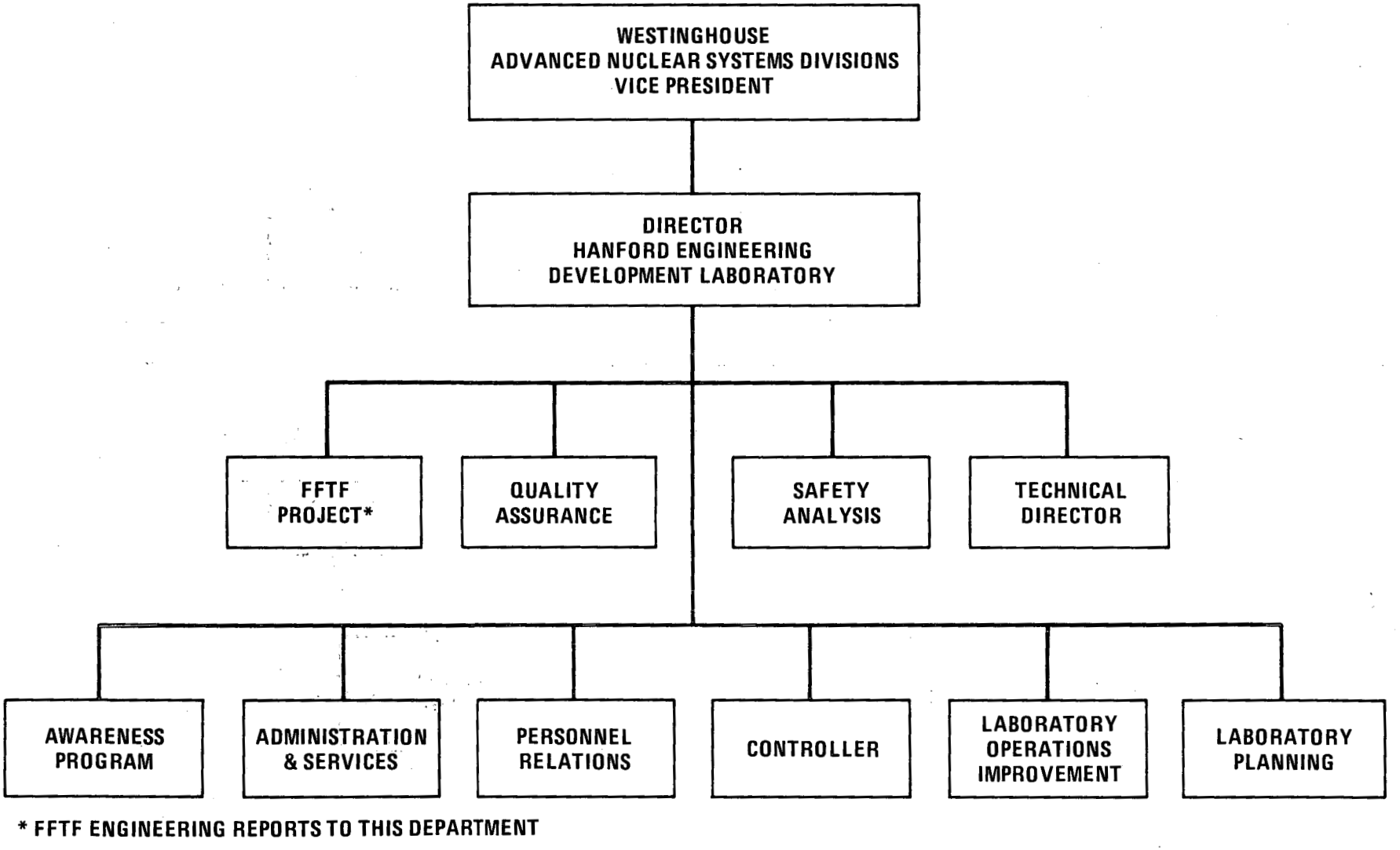


Figure 17.2-1 Hanford Engineering Development Laboratory Organization

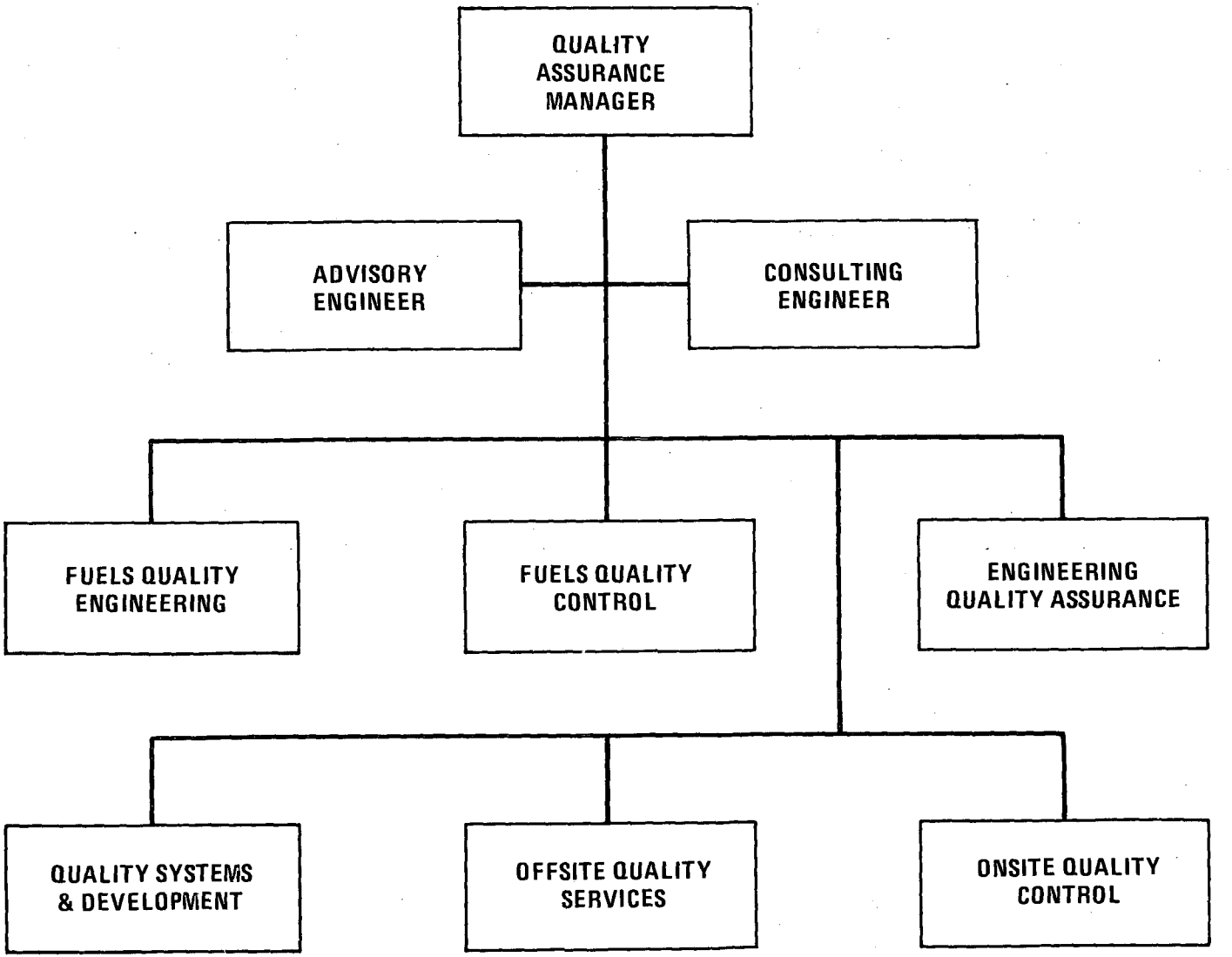


Figure 17.2-2 Hanford Engineering Development Laboratory Quality Assurance Department

The Quality Assurance Department is given responsibility for: (1) training and indoctrinating technical, management, and quality assurance personnel in quality assurance requirements; (2) reviewing and approving quality-related documents (e.g., instructions, procedures, drawings, specifications, and test plans); (3) assuring that personnel qualifications are current and applicable to the work being performed; (4) assuring that procurement documents include applicable quality assurance requirements; (5) performing pre-award evaluation of suppliers and surveillance and inspection at the suppliers' facilities; (6) approving the disposition of nonconforming items; (7) providing or approving receiving inspection plans; (8) assuring that corrective actions are effective and accomplished in a timely manner; and (9) auditing and surveillance of maintenance, modification, and operation activities performed by other organizations.

17.3 Quality Assurance Program

The Hanford Engineering Development Laboratory has committed to comply with the regulatory guidance provided by the Commission in Table 17.1-1, augmented by acceptable alternatives contained in Supplements 5 and 15 of the Final Safety Analysis Report." The quality assurance program for the FFTF also satisfies the requirements contained in applicable Reactor Development Technology Standards including F2-2, "Quality Assurance Program Requirements." This standard along with F5-1T, "Cleaning and Cleanliness Requirements for Nuclear Components," and F7-2T, "Packaging, Packing, and Marking of Components for Shipment and Storage," were evaluated and found to contain equivalent controls to those contained in the corresponding ANSI Standards (i.e., N45.2, N45.2.1 and N45.2.2) for light water reactors.

The policies, procedures, and instructions by which the quality assurance program for the FFTF is implemented are contained in the HEDL Quality Assurance Manual and other controlled documents. The quality assurance program requires that implementing documents encompass detailed controls for: translating codes, standards, regulatory requirements, technical specifications, engineering and process requirements into drawings, specifications, procedures, and instructions; developing, reviewing, and approving procurement documents, including changes; prescribing all quality-related activities by documented instructions, procedures, drawings, and specifications; issuing and distributing approved documents; purchasing items and services; identifying materials, parts, and components; performing special processes, inspecting and/or testing materials, equipment, processes or services; calibrating and maintaining measuring and test equipment; handling, storing, and shipping of items; identifying the inspection, test, and operating status of items; identifying and dispositioning nonconforming items; correcting conditions adverse to quality; preparing and maintaining quality assurance records; and auditing activities which affect quality. Quality is verified through checking, review, surveillance, inspection, testing, and audit of quality-related activities. The quality assurance program requires that quality verification be performed by individuals who are not directly responsible for performing the quality-related activities.

Inspections are performed by qualified personnel in accordance with procedures, instructions, and drawings approved by the Quality Assurance Department. Nondestructive examination personnel are certified in accordance with SNT-TC-1A.

The Hanford Engineering Development Laboratory has established an indoctrination and training program to assure that personnel performing activities affecting quality are knowledgeable in quality assurance/quality control requirements, implementing procedures and instructions; and demonstrate a high level of competence and skill in the performance of their quality-related activities.

The Quality Assurance Department is responsible for the establishment and implementation of the audit program. Audits are performed in accordance with preestablished written checklists by appropriately trained personnel not having direct responsibilities in the areas being audited. The audit function, which is conducted at scheduled intervals and/or on a random unscheduled basis, includes an objective evaluation of the adequacy of and compliance with quality assurance policies, practices, procedures, and instructions; the adequacy of work areas, activities, processes, items, and records; the effectiveness of implementation of the quality assurance program; and product compliance with applicable engineering drawings and specifications. The quality assurance program requires documentation of audit results and review by management having responsibility in the area audited to determine and take corrective action needed, if any.

Followup audits are performed to determine that nonconformances are effectively corrected and that the corrective action precludes repetitive occurrences. Audit reports, which indicate performance trends and the effectiveness of the quality assurance program, are prepared and issued to responsible management for review and assessment.

17.4 Conclusions

Our review of the quality assurance program description for the operations phase of the FFTF has verified that the criteria of Appendix B to 10 CFR Part 50 have been adequately addressed.

Based on our detailed review and evaluation of the quality assurance program description contained in Chapter 16.0 of the Final Safety Analysis Report through Amendment 14 and Supplements 5 and 15 to the Final Safety Analysis Report for the FFTF, we conclude that:

(1) The Quality Assurance Department is provided sufficient independence from cost and schedule (when opposed to safety considerations), sufficient authority to effectively carry out the quality assurance program, and sufficient access to management at a level necessary to perform their quality assurance functions.

(2) The quality assurance program description contains adequate quality assurance requirements and a comprehensive system of planned and systematic controls which address each of the criterion of Appendix B to 10 CFR Part 50 in an acceptable manner.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In its letter of July 15, 1975, the Advisory Committee on Reactor Safeguards indicated that certain matters would require further attention and resolution during the construction of the Fast Flux Test Facility.

Certain of these items have been resolved and others are addressed and discussed in this report as indicated in the following sections:

Section 1.7(6)	-	Liner Vents
Section 1.7(7)	-	Head Cavity
Section 1.7(8)	-	Subcavity
Section 13.2	-	Emergency Plans
Section 15.0	-	Containment Filters
Section 15.0	-	Post-Accident Evaluation

19.0 CONCLUSIONS AND ADVICE

19.1 Conclusions

Based on our evaluation of the FFTF Final Safety Analysis Report, including Supplements and Amendments through Number 27, and assuming favorable resolution of unresolved matters identified in this report, we conclude that:

(1) The design features important to safety provided for the FFTF comply with the intent of existing rules and design criteria for licensed reactors. Where the FFTF design has deviated from licensed light water reactor practice due to unique or special characteristics of a test reactor or liquid metal system, the safety significance of these differences has been evaluated to ensure that an adequate level of safety has been maintained.

(2) The FFTF can be operated as a test reactor under its initial conditions of operation in a safe manner with reasonable assurance of not endangering the health and safety of the public.

(3) The Department of Energy contractors, Hanford Engineering Development Laboratory, have an adequately trained and technically qualified staff of management and operating personnel who can operate the facility in a safe manner.

19.2 Staff Review

A summary of the staff's review of FFTF, including some comments on items not reviewed at the Department of Energy's request, is provided below:

(1) The staff review of the FFTF design was based on a plutonium-uranium mixed oxide core limited to an average burnup of 45,000 MWd/kg and operating temperatures of 600°F and 858°F sodium inlet and outlet respectively. The review was limited to three loop operation and did not include the effect of the closed loops in the core. The review did not include details of the higher operating temperatures associated with rated conditions nor the use of newer fuels which may be contemplated for future use.

(2) The staff did not review the FFTF design for operating basis earthquake events greater than the 0.05g since the plant will be shut down for these larger events and re-evaluated for structural damage prior to startup.

(3) The staff review is limited to safety evaluation of the design presented in the Final Safety Analysis Report. The staff did not perform any onsite review to audit conformance of construction with design requirements, as would be done for a licensed facility. This limitation was in accordance with the request of the Department of Energy.

(4) Also in accordance with the Department of Energy's request the staff did not review the FFTF safeguards and security provisions. Therefore the staff did not reach any conclusions regarding their effectiveness compared to those of licensed facilities or regarding the completeness or adequacy of the security and safeguards evaluations performed by the Department of Energy and its contractors.

(5) The Commission will not test or license the FFTF operators. This is a Department of Energy responsibility.

19.3 Advice and Recommendations

A. Natural Circulation Decay Heat Rejection

The Project has decided to rely on passive operational circulation in the heat transport systems for decay heat removal as a backup to forced circulation and has agreed to demonstrate the adequacy of the natural circulation option in the primary and secondary heat transport systems and in the dump heat exchanger modules by in-plant testing. The decision to rely on natural circulation has resulted in the Project choosing not to provide onsite electrical components and systems for decay heat removal which satisfy our requirements for safety-related systems. Since the performance level of natural circulation and the associated plant thermal-hydraulic conditions have not been well established and cannot be adequately established without the conduct of a detailed test program to determine FFTF system behavior, we recommend that the Project:

(1) Continue to pursue vigorously the development and implementation of the natural circulation verification test program, with emphasis on obtaining both a very wide range of data and data which closely resemble the predicted conditions of decay heat removal following full power operation.

(2) Define acceptance criteria for the tests so that it is clear what constitutes acceptable performance, with special emphasis on model acceptance. It should be made clear that the computer models used for these analyses need to be verified both in the natural circulation regime and over the full range of reactor operation to substantiate the safety analysis in Chapter 15.0 of the Final Safety Analysis Report.

(3) Identify the plant modifications to be implemented if natural circulation is not adequately established (for example, additional onsite power sources which satisfy staff requirements and forced circulation with pony motors).

We believe natural circulation to be a highly important safety matter which is fundamental to the safety approach on the FFTF. Failure to demonstrate adequate natural circulation in the FFTF may result in major costly plant modifications such as adding redundant diesel generators and safety grade electrical equipment to permit the operation of the pony motors during all shutdown modes.

In line with the above recommendations, we have requested that the proposed Natural Circulation Verification Test Program to be initiated during the initial reactor start-up phase of operation include verification of the IANUS and FLODISC codes for plant conditions expected throughout the operating range of the reactor, thereby confirming the adequacy of the safety analysis in Chapter 15.0 with respect to thermal-hydraulic transients. We have requested that we be provided with the details of the test plan and its schedule, the pretest transient and steady state predictions and, on a timely basis, the experimental test results. We do not recommend unrestricted reactor operation until the natural circulation capability has been verified and the safety analysis relating to core thermal-hydraulics confirmed. Detailed staff recommendations are presented in Section 14.2 of this report.

B. Containment Margins for Core Melt Events

As discussed in Supplement 2 to the previous Safety Evaluation Report and Section 15.3.6 of this report, we believe that there exists some uncertainty in the reliability of the mechanisms to prevent vessel melt-through in the event of a core melt accident. Although the rates and extent of penetration of sodium and core debris into concrete are not fully resolved, calculations have shown that over a wide variety of postulated accident conditions, the release of radioactivity would not result in doses to the public that are excessively above acceptable guidelines, provided that there are means to institute a controlled release of the containment atmosphere and that a reasonable degree of containment integrity is maintained.

We have determined that the generation of hydrogen resulting from sodium and core debris interactions with concrete is the principal challenge to containment which has not been fully resolved. Fully applicable data on concrete attack rates and hydrogen production have not been forthcoming, so that an assessment of the containment response requires consideration of the uncertainties associated with those processes. In the event of a core melt-through, the quantity of hydrogen generated could create potentially explosive or highly energetic flammable mixtures in the containment building atmosphere or portions of the building which could constitute a significant early threat to the integrity of the containment structure.

We have considered various means to alleviate the buildup of hydrogen in the containment building. A brief discussion of each is provided:

(1) Sacrificial Materials - High-temperature sacrificial materials which do not generate hydrogen and which protect the liner and concrete from sodium and core debris could be added (1) in the reactor cavity to protect the steel liner and (2) in the containment subcavity to protect the bare concrete floor and side walls. Because the reactor cavity and subcavity have been sealed, installation of additional materials in these regions would be difficult and expensive; however, it is not impossible prior to reactor startup.

(2) Spontaneous Recombination - Based on very small-scale tests reported in Ref. 10 of Section 15.3.6 of this report, the Project concludes that hydrogen will recombine spontaneously with oxygen provided the criteria of Ref. 10 are satisfied. However, the recombination of hydrogen and oxygen in the presence of sodium aerosols has not been demonstrated for the large containment volume and the possible stratified conditions which may exist there. Moreover, as discussed in Section 15.3.6 and Appendix F of this report, the influence of spontaneous recombination on the containment transient response was found not to be dominant.

(3) Catalytic Recombination - Recombining hydrogen catalytically at low rates of energy release is not considered to be feasible in the presence of an atmosphere that may be heavily loaded with sodium reaction products.

(4) Inert Gas - Dilution of the containment building atmosphere with an inert gas during a core melt accident is feasible but somewhat impractical because of the large containment volume.

(5) Artificial Combustion - Inducing combustion of hydrogen artifically while it is at low enough concentration to avoid a serious detonation may alleviate this problem; however, the Project has not provided documentation justifying this option.

(6) Containment Subcavity - Eliminating the containment subcavity void by filling it with either concrete or, as mentioned earlier, a sacrificial material would mitigate and delay the significant containment response transient which occurs after the cavity floor is penetrated and the sodium and core debris come in direct contact with the large bare concrete surfaces of the subcavity. As has already been mentioned, installation of additional material in the subcavity would be difficult and expensive, although it is feasible.

In view of the well known industrial hazard associated with accumulations of hydrogen, we believe that this subject should be further addressed by the Project. Specific plans should be made either for the monitoring and control of the hydrogen concentration in the course of a melt-through, or for the prevention or limitation of its formation in the containment building.

The Nuclear Regulatory Commission is supporting large-scale materials interaction experiments directed toward determining the rates and extent of sodium and core-debris penetration into concrete (see Section 15.3.6 of this report). Prior to the initial criticality of the FFTF, these experiments may provide significant additional understanding of the concrete attack phenomena and permit a different conclusion from that presented here. The Department of Energy is also urged to pursue its own large-scale experimental program in this area. Final confirmation of the soundness of present conclusions and the validity of any provisions to cope with the hydrogen problem could be significantly influenced by the results of these tests. Nevertheless, the alternatives for coping with the hydrogen generated should be carefully studied in more detail so that a suitable approach can be implemented upon completion of current experimental programs. The staff desires to be kept informed of the progress of this effort.

The staff has suggested that the emergency plan for the facility take into account the timing of the events of a core melt-through accident (Safety Evaluation Report Supplement 2, Section 7.0). Emergency plans for the facility are discussed in the Final Safety Analysis Report, Chapter 13.0. These plans have been reviewed by the staff and found to be generally acceptable. In the particular case of a core melt-through accident accompanied by limited venting or purging of the containment atmosphere, we have also concluded that the emergency plan is acceptable in both time-scale and scope to provide for the implementation of suitable procedures, including evacuation if necessary.

In accordance with the above discussion and evaluations described in Section 15.3.6 of this report, our conclusions and recommendations are:

(1) Hydrogen remains a serious potential problem according to available data and calculations.

(2) The Project should provide the necessary means to monitor and control the hydrogen concentration in the containment under those conditions in which the hydrogen concentration could threaten containment integrity.

(3) Large-scale experiments need to be pursued more aggressively to obtain better data on the generation and disposition of hydrogen under the conditions of interest.

(4) The Project should provide the means to vent the containment building in a controlled manner and prevent containment failure by overpressurization. The

staff has calculated that a controlled venting rate of about 4000 cubic feet per minute will be required when the building pressure is at or near 10 psig. This rate may be affected by additional information from the Nuclear Regulatory Commission's confirmatory large-scale materials interaction experiments and the Department of Energy's large-scale experimental program.

C. Piping Integrity

The staff's recommendation in its July 11, 1976 letter and in subsequent discussions with the Project regarding piping integrity, leak detection, inservice inspection and material surveillance includes a baseline preservice volumetric examination of highly stressed welds (hot crossover piping), periodic inservice inspection of these welds, inclusion of a sensitive reliable aerosol leak detection system and initiation of a material surveillance program. We have also recommended placement of an aerosol leak detector probe near each of the reactor inlet pipes in the reactor guard vessel and placement of a tube in each of the three hot crossover piping sections as a means for aerosol communication around pipe supports. The Project did initiate research and development efforts to develop an ultrasonic detection device for performing this type of inservice inspection at 400°F but since has stated that this device may not be sufficiently developed in time to be used before reactor startup. The staff recommends that the Department of Energy vigorously pursue this effort and the implementation of the inservice inspection program for the FFTF at the earliest practical date. The staff recommends also the continuation of the material surveillance program applicable to the hot leg piping and its implementation in the FFTF at the earliest practical date. The staff recommends that the originally proposed sections of removable insulation be retained and provisions be included to allow for tracks to permit remote manual inspection at a later date. If these programs are not ready to be implemented coincident with the present FFTF startup schedule, the effort nevertheless should continue and the implementation completed as soon as possible so that the system can be used for the balance of the FFTF operating life. An alternative is to perform periodic inservice inspection loop by loop in a cold drained condition using existing conventional methods as is done on light water reactors. We recommend a baseline ultrasonic examination of these welds cold and after wetted with sodium at 400°F. The staff's assessment of other aspects related to piping integrity is further discussed in Appendix E to this report.

As previously stated, the staff believes that sodium leak detectors perform an important safety function, (i.e., knowledge of the status of the reactor coolant system following a safe shutdown earthquake) and has recommended to the Project that the sodium aerosol leak detectors be seismically qualified and powered by Class IE power such that they may be relied upon following a safe shutdown earthquake. We therefore encourage the Department of Energy to implement these recommendations. We will evaluate the Project's response in a supplement to this report.

D. Cell Liner Test Plan

The staff considers that cell liners have an important safety function because they prevent large surface areas of concrete from coming in contact with significant quantities of hot sodium from postulated sodium spills or leaks.

The liners considered important are those which may encounter hot sodium and include those located in the reactor cavity, the heat transport cells and the sodium overflow tank cell. The LT-1 tests and subsequent small-scale failed liner test results were encouraging. The tests indicated that FFTF-type cell liners were not likely to fail when subjected to a hot sodium spill and that, if relatively small areas in the liner were open exposing concrete to sodium, the sodium-concrete reaction appeared to be self-limiting. This is an important consideration to the reactor cavity and heat transfer system cells since these areas, particularly the reactor cavity, will become relatively inaccessible and the reactor cavity liners will not be repairable once the reactor is placed in operation.

The Project's response to the staff request to propose a test plan demonstrating the continued functional capability of the accessible cell liners over the life of the facility has not been received, but we understand that it is in preparation and it will be submitted to us. We recommend that experimental testing with both intact and failed liners, including failed liner welds, be continued to show that if failure in the liners occur, the plant will not have to be shut down. We will evaluate additional test data as it becomes available.

E. Loose Parts Monitoring

It is the staff's opinion that the FFTF should be equipped with a loose parts monitoring system and urges continuing support of the program at Argonne National Laboratory for developing reliable under-sodium ultrasonic detectors. When this or a comparable device is developed for loose parts monitoring, we recommend its installation in the FFTF unless it creates an unanticipated safety problem. The Project has agreed to continue support of this activity.

F. Maximum Fuel Channel Exit Temperature

Although the staff review is limited to the proposed conditions of initial operation, e.g., reduced temperature, mixed oxide core, and low burn up, we have looked at some proposed rated operating conditions. On the basis of information currently available, i.e., unverified thermal-hydraulic codes and uncertainty in hot channel factor values, and in order to assure no boiling in the core, we recommend initially limiting the maximum fuel channel exit temperature to 1670°F rather than 1705°F until the latter number can be more fully supported with operational data. The Project has accepted our recommendation.

G. Fuel System

The FFTF as a test reactor, in contrast to a power reactor, has the facilities and capability for removal of fuel, for disassembly of subassemblies, for examination of pins, for reassembly of the pins into subassemblies, and for insertion back into the reactor. Refueling is scheduled for about 100 day intervals. We believe that a strong fuel surveillance program in FFTF should be supported to obtain both prototypic steady-state and transient fuel performance data to provide confirmatory support for the FFTF fuel design and to some degree the methodology used for the design of the fuel. We strongly urge support of this program. The Project has stated that such a plan exists and has agreed to its implementation. We request that a copy of this plan be submitted for our review when it becomes available.

H. Tornado Conditions

Since in the event of a tornado both the loss of all offsite and onsite AC power and the loss of two main secondary loops can be postulated, all decay heat from full power would have to be removed through one tornado missile-hardened secondary loop. Within a relatively short period of time a plant operator would have to go to the hardened dump heat exchanger to adjust outlet louvers and start isolating the modules. Recovery from such an unplanned shutdown resulting from a tornado will be hampered by debris which may obstruct the access of plant operators and prevent certain operations from being carried out in a timely manner. We therefore recommend that the Project implement its proposal to bring the reactor to a shutdown status whenever meteorological conditions in the vicinity of the site are conducive to formation of a tornado. The Project has agreed to this recommendation. Also, the staff does not recommend operation of the plant in the event of erratic behavior of the check valve of the tornado missile-hardened heat transfer circuit.

19.4 Miscellaneous Unresolved Issues

The items discussed in this section comprise a list of incomplete or unresolved issues. They represent review areas in which a recommendation or a request for information has been made to the Project and for which the Project's response or the staff's documentation of review remains incomplete or not resolved at this time. These issues will be closed out or addressed in a supplement to this report. They are not expected to interfere with the initial sodium fill startup phase of the FFTF. Since the preparation of this report, the Project has agreed to shut down the reactor when weather conditions conducive to a tornado exist in the vicinity of the site. Because of this agreement the staff's concerns regarding a Class IE power supply for containment isolation valve position indication have been resolved. We conclude that the control power proposed for the purpose of valve position indication is acceptable providing assurance can be given that false position indications will not occur.

(1) The review of the closed loops has been deferred since their design is incomplete. The Project has advised the staff that these loops will not be installed until the seventh fuel cycle. We will address this subject in a supplement to this report.

(2) The staff has not finished its review of conventional fire hazards. These will be addressed in a supplement. The existing conventional fire protection is believed adequate for the initial sodium fill prior to startup of the FFTF. Sodium fires have been addressed in this report.

(3) If natural circulation is not adequately demonstrated, the main heat transfer system pumps may have to be qualified to operate at pony motor speeds and without seal oil lubrication or the seal oil pumps will have to satisfy the requirements for safety-related equipment, depending on how the issue is resolved. The Project is aware of this potential problem.

(4) We recommend upgrading the reserve air system at the tornado missile-hardened dump heat exchanger from Quality Group D to Quality Group C for emergency operation of valves and louvers, which is a safety-related function. The Project has agreed to upgrading the existing system insofar as possible without replacing it.

(5) We have not completed our review of the typical seismic and environmental qualification procedures used by the Project for safety-related instruments. The Project has transmitted these procedures to the staff. We will address these concerns in a supplement.

(6) We had requested the Project to provide an alarm in the event of malfunction of the heating and ventilating system for situations where malfunctions of this equipment could cause degradation of safety-related equipment. The Project assured the staff that the safety-related equipment was qualified for the most severe environmental conditions. Documentation has been provided recently but has not been reviewed. This subject will be addressed in a supplement.

(7) The staff has not completed its review of protective system instrumentation exposed to high levels of radiation and will address this concern in a supplement.

(8) We recommend, in lieu of removal of safety-related sensors for time response testing due to all-welded construction, that output parameters of other sensors be compared during transient events such as scram to demonstrate that no degradation of sensor time response has occurred. The Project is in accord with this procedure and resolution of this issue will be addressed in a supplement.

(9) We recommend that the Project identify those circuits and equipment in the containment isolation system which have not been qualified to MIL HDBK 217 criteria and show that the overall reliability of the system is not affected.

The Project has agreed to supply additional information which we will address in a supplement.

(10) The Project has committed to providing a shutdown status display for the post-accident monitoring of the reactor. We request documentation of how this will be done and will address it in a supplement.

(11) We recommend bellows-sealed pressure transducers for measuring containment pressure rather than the spin orifice plate used by the Project. The Project does not consider this necessary. This will be addressed in a supplement.

(12) We recommend 18 valves as part of containment isolaton be given type C tests. The Project has committed to do this and will provide the documentation. We will address this in a supplement.

(13) The staff has not reviewed the environmental qualifications for Class IE electrical equipment and terminal blocks. The Project has agreed to supply additional information. This will be addressed in a supplement.

(14) We have recommended independent and redundant valve position indication on primary and secondary heat transport system sodium valves. The Project has stated that valve position indication is displayed at all times. We believe this concern can be satisfactorily resolved. The resolution of this item will be reported in a supplement.

(15) We recommend that electrical penetrations through containment be provided with breakers rather than fuses. The Project has responded that the use of fuses is in compliance with Regulatory Guide 1.68 (1973) and it would not be cost beneficial to replace them with breakers. We will reconsider our recommendation in a supplement to this report.

(16) The equipment air lock escape hatch indicator is not on Class IE power. The Project states this display is on control power. We find this acceptable.

APPENDIX A

CHRONOLOGY OF RADIOLOGICAL REVIEW
FAST FLUX TEST FACILITY

NOTE: Documents referenced in this chronology are available for public inspection and copying for a fee at the NRC Public Document Room 1717 H Street, N.W., Washington, DC.

March 25, 1976	Submittal of the Final Safety Analysis Report (FSAR) for the Fast Flux Test Facility (FFTF) by the U.S. Energy Research and Development Administration (ERDA)
March 31, 1976	Submittal of the Technical Specifications for the FFTF
May 11, 1976	Meeting in Bethesda with ERDA staff to discuss the quality assurance program
June 11, 1976	Letter to ERDA (T. Speis to R. Ferguson) transmitting a staff evaluation report of Appendix B of the FSAR, Primary Piping Integrity, and requesting additional information in this area
July 20-21, 1976	Meeting in Bethesda with ERDA staff to discuss natural convection
August 13, 1976	Letter to ERDA (R. Boyd to R. Ferguson) advising that acceptance review of the FSAR has been completed; that based on understanding reached during 7/21/76 meeting the FSAR is accepted for detailed review; and requesting additional information resulting from acceptance review
August 20, 1976	Letter from ERDA (R. Ferguson to R. Denise) providing clarification regarding the scope of NRC review that is being requested
September 3, 1976	Letter to ERDA (R. Denise to R. Ferguson) advising that NRC will proceed with review as proposed in 8/20/76 letter, noting several points that should be given further consideration, and transmitting a review schedule
September 13, 1976	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information regarding containment

September 15, 1976	Submittal of Supplement 2 to FSAR, consisting of responses to letter of 6/11/76 in the area of piping integrity
September 17, 1976	Site visit and meeting with ERDA staff in Richland to discuss the electrical system, piping integrity, reactor cavity liner, natural convection studies and pump problems
October 18, 1976	Submittal of Supplement 1 to FSAR, consisting of responses to letter dated 8/13/76
October 28, 1976	Meeting in Bethesda with ERDA staff to discuss selection of hot channel factors, their combination and application and update on natural convection analyses
November 17, 1976	Submittal of Supplement 3 to FSAR, consisting of further responses to letter dated 8/13/76
November 24, 1976	Submittal of Supplement 4 to FSAR, consisting of cross references correlating FSAR and SDDs to NRC Standard Format and Content of Safety Analysis Reports, LMFBR Edition
January 7, 1977	Submittal of Supplement 5 to FSAR, consisting of responses to letter dated 9/13/76
January 7, 1977	Submittal of Amendment 2 to FSAR, consisting of quality assurance revisions
January 14, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information in the areas of mechanical and structural engineering, reactor analysis and core performance
January 19, 1977	Submittal of Amendment 1 to FSAR, consisting of revisions to the plant protection and building electrical systems
January 28, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information on accident analyses, radiological assessment and meteorology
February 8, 1977	Submittal of Amendment 3 to FSAR, consisting of revisions to safety features, auxiliary systems and emergency planning
February 15, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information concerning hot channel factors

February 16, 1977	Letter from ERDA (R. Ferguson to R. Denise) requesting that review schedule not be delayed and, if possible, completed by August 1978
March 18, 1977	Submittal of Amendment 4 to FSAR, consisting of revisions to safety features, instrumentation and controls, radiation protection and quality assurance
March 21, 1977	Letter from ERDA (R. Ferguson to R. Denise) advising of revisions to initial operation conditions
March 29, 1977	Letter to ERDA (T. Speis to R. Ferguson) advising that due to late receipt of some material and references and problems being encountered, the review schedule may slip to December 1978 or January 1979
April 4, 1977	Submittal of Supplement 6 to FSAR, consisting of responses to letter dated 1/14/77
April 20, 1977	Submittal of Supplement 7 to FSAR, consisting of responses to letter dated 2/15/77
April 22, 1977	Submittal of Supplement 8 to FSAR, consisting of responses to letter dated 1/28/77
May 2, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information concerning power systems
May 4, 1977	Submittal of Amendment 5 to FSAR, consisting of revisions made in response to letter dated 1/14/77
May 9, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information concerning reactor analysis
May 23, 1977	Submittal of Amendment 7 to FSAR, consisting of revisions made in response to letter dated 1/14/77
May 26, 1977	Meeting in Bethesda with ERDA staff to discuss the review schedule and progress of construction
June 9, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information concerning fuel performance
June 15, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting an evaluation on the status of diffuse material relating to fires and a comprehensive fire hazards analysis report

June 17, 1977	Meeting in Bethesda with ERDA staff to discuss the seismic design of building structures
June 30, 1977	Meeting in Bethesda with ERDA staff to discuss core thermal-hydraulic analyses, hot channel factors and power level control
July 13, 1977	Letter from ERDA (R. Ferguson to R. Denise) providing information on comprehensive fire hazards evaluation
July 14, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information on quality assurance
July 14, 1977	Letter from ERDA (R. Ferguson to R. Denise) advising that it is not necessary for NRC review to include the areas of safeguards and security
July 18, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information on reactor systems
July 22, 1977	Submittal of Amendment 6 to FSAR, consisting of revisions and safety assessment report which reflects change from Mobil-therm to water/glycol mixture coolant in certain portions of in-containment inert gas and air cooling systems
July 27, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information on quality assurance, hydrology and meteorology
August 8, 1977	Letter from ERDA (R. Ferguson to R. Denise) concerning the establishment of magnitude of current seismic design margins
August 9, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information on cell liners
August 17, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information in the area of mechanical engineering and accident analysis
August 17, 1977	Letter from ERDA (R. Ferguson to R. Denise) transmitting plans for inservice testing of safety-related pumps and valves
August 19, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information on instrumentation and quality control

August 29, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information on the containment systems
August 31, 1977	Letter from ERDA (R. Ferguson to R. Denise) providing responses to questions on hot channel factors
September 2, 1977	Letter to ERDA (T. Speis to R. Ferguson) requesting additional information on materials
September 9, 1977	Submittal of Supplement 10 to FSAR, consisting of responses to questions in the electrical area
September 9, 1977	Submittal of Supplement 11 to FSAR, consisting of responses to requests for additional information
September 20-22, 1977	Site visit and meeting in Richland with ERDA staff to discuss the current project schedule
September 27, 1977	Meeting in Bethesda with ERDA staff to discuss the fire hazards analysis report
October 5, 1977	Submittal of Amendment 10 to FSAR, consisting of revisions to hot channel factors
October 14, 1977	Submittal of Supplement 16, Part I to FSAR, consisting of responses to questions on meteorology
October 18, 1977	Letter to Department of Energy (DOE-formerly ERDA) (T. Speis to R. Ferguson) requesting additional information on power systems and possible error in SPRAY code
October 19, 1977	Meeting in Bethesda with DOE staff to discuss the reactor cavity liner
October 21, 1977	Submittal of Supplement 15 and Amendment 14 to FSAR, consisting of quality assurance revisions
October 26, 1977	Submittal of Supplement 12, Part II to FSAR, consisting of responses to remaining questions in the area of core performance
November 1, 1977	Submittal of Supplement 14 to FSAR, consisting of responses to questions on conduct of operations
November 11, 1977	Letter to DOE (T. Speis to R. Ferguson) requesting additional information on containment margins for post-accident heat removal

November 16, 1977	Submittal of Supplement 13, Part I to FSAR, consisting of responses to letter dated 11/18/77
November 25, 1977	Letter to DOE (T. Speis to R. Ferguson) requesting additional information on reactor systems
December 7, 1977	Submittal of Amendment 13 to FSAR, consisting of revised information in response to letter dated 6/9/77
December 12, 1977	Submittal of Amendment 11 to FSAR, consisting of revised information in response to letter dated 7/18/77
December 12, 1977	Submittal of Amendment 15 to FSAR, consisting of radiation protection revisions
December 13, 1977	Submittal of Supplement 16, Part II to FSAR, consisting of responses to letter dated 8/17/77
December 13, 1977	Submittal of Supplement 17 to FSAR, consisting of responses to questions on cell liners
December 13, 1977	Submittal of Supplement 21 to FSAR, consisting of responses to letter dated 9/2/77
December 13, 1977	Submittal of Supplement 24 to FSAR, consisting of responses to questions concerning SPRAY computer code
December 16, 1977	Letter to DOE (T. Speis to R. Ferguson) requesting additional information on auxiliary systems
December 16, 1977	Letter to DOE (T. Speis to R. Ferguson) requesting additional information on fire protection
December 16, 1977	Submittal of Supplement 19 to FSAR, consisting of responses to letter dated 2/15/77
December 16, 1977	Submittal of Supplement 22 to FSAR, consisting of responses to letter dated 10/18/77
December 16, 1977	Submittal of Supplement 23 to FSAR, consisting of responses to letter dated 11/11/77
December 20, 1977	Submittal of Supplement 20 to FSAR, consisting of responses to letter dated 8/29/77

December 20, 1977	Submittal of Amendment 16 to FSAR, consisting of revised information in response to letter dated 8/29/77
December 31, 1977	Submittal of Supplement 25 to FSAR, consisting of responses to letter dated 12/16/77
January 9, 1978	Submittal of Amendment 12 to FSAR, consisting of revised information in response to letter dated 2/15/77
January 9, 1978	Submittal of Amendment 18 to FSAR, consisting of revised information in response to letter dated 10/18/77
January 9, 1978	Submittal of Amendment 19 to FSAR, consisting of revised information in response to letter dated 2/15/77
January 9, 1978	Submittal of Amendment 20 to FSAR, consisting of revised information in response to letter dated 12/16/77
January 17, 1978	Letter to DOE (R. Denise to R. Ferguson) advising that if NRC review does not include areas of safeguards and security, it will not be fully comparable to a licensed facility review and, therefore, no conclusions can be reached in these areas
January 23, 1978	Letter to DOE (T. Speis to R. Ferguson) requesting additional information on reactor systems and quality assurance operations
January 30, 1978	Submittal of Amendment 21 to FSAR, consisting of revised information on hot leg pipe rupture
January 31, 1978	Letter from DOE (A. Rizzo to R. Denise) transmitting responses to letter dated 11/11/77
February 7, 1978	Submittal of Supplement 26 to FSAR, consisting of responses to letter dated 11/25/77
February 13, 1978	Letter to DOE (T. Speis to R. Ferguson) advising that staff is still concerned about the adequacy of containment margins and requesting two base case studies using parameters provided
February 16, 1978	Submittal of Supplement 28 to FSAR, consisting of responses to questions on piping systems
February 17, 1978	Submittal of Amendment 22 to FSAR, consisting of revised information in response to letter dated 9/2/77

February 24, 1978	Submittal of Amendment 24 to FSAR, consisting of revised information in response to letter dated 7/18/77
March 8, 1978	Submittal of Supplement 20 to FSAR, consisting of responses to requests for additional information
March 9, 1978	Submittal of Amendment 23 to FSAR, consisting of revised information
March 9, 1978	Submittal of Amendment 26 to FSAR, consisting of revised information
March 17, 1978	Submittal of Amendment 9 to FSAR, consisting of a revised Emergency Plan
March 27, 1978	Letter from DOE (A. Rizzo to R. Denise) responding to outstanding requests for information
March 29, 1978	Meeting in Bethesda with DOE staff to discuss the containment margins
April 4-7, 1978	Site visit and meeting with DOE staff in Richland to discuss instrumentation and control topics
April 10, 1978	Submittal of Supplement 27 to FSAR, consisting of responses to letter dated 1/23/78
April 18, 1978	Submittal of Supplement 24 to FSAR, consisting of updated information on modifications being made to SPRAY code
April 26, 1978	Letter from DOE (A. Rizzo to R. Denise) transmitting additional information on seismic qualification of instrumentation and control equipment
April 28, 1978	Meeting in Bethesda with DOE staff to discuss the proposed natural circulation verification test program
May 4, 1978	Letter from ERDA (A. Rizzo to R. Denise) responding to NRC concerns regarding containment margins
May 16, 1978	Letter to DOE (T. Speis to R. Ferguson) requesting necessary information dealing with electrical power and containment leakage testing

May 31, 1978	Letter from DOE (R. Ferguson to R. Denise) in response to 4/26/78 letter, requesting that comparison against 1971 LWR criteria not be included in overall review scope
June 2, 1978	Letter from DOE (A. Rizzo to R. Denise) transmitting responses to NRC position on electrical power
June 13, 1978	Letter from DOE (A. Rizzo to W. Gammill) requesting that the closed loop plant protection system design be excluded from scope of review currently in progress
June 13, 1978	Submittal of Amendment 27 to FSAR, consisting of information regarding environmental qualification of safety-related equipment
July 12, 1978	ACRS Subcommittee meeting

APPENDIX B

REFERENCE LETTERS

UNITED STATES
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
DIVISION OF REACTOR RESEARCH AND DEVELOPMENT
FAST FLUX TEST FACILITY PROJECT OFFICE
P. O. BOX 550
RICHLAND, WASHINGTON 99352

NOV 13 1975

Nuclear Regulatory Commission
ATTN: Mr. R. P. Denise, Assistant Director
for Advanced Reactors
Washington, D. C. 20555

Gentlemen:

References: (1) Letter dated August 7, 1975, R. P. Denise, NRC, to T. A. Nemzek, RRD-HQ, regarding the FFTF FSAR Review

(2) Letter dated August 12, 1975, R. P. Denise, NRC, to R. L. Ferguson, FFTFPO, regarding additional documentation of the consequences of a postulated loss of in-vessel postaccident heat removal

In your August 7, 1975, letter to Mr. T. A. Nemzek [reference (1)], you requested confirmation of our plans for NRC review of the FFTF Final Safety Analysis Report (FSAR). As indicated in our November 5, 1975, discussions, the forecast date for submittal of the FFTF FSAR for NRC review is March, 1976. It is requested that the NRC review be targeted for completion approximately two years after submittal. This provides the same review time that you have indicated is normally required for review of licensed plants.

Regarding the scope of the NRC assessment, we request that—based on review of appropriate design documentation and safety evaluations—NRC provide advice regarding the adequacy of the FFTF design to ensure safe operation of the plant. The NRC reviews need not include assessments of "as built" configurations, construction audits, evaluation of acceptance test program results, etc., to verify that the plant is constructed in accordance with design criteria and documentation. These functions are accomplished within the ERDA organization as required by ERDA Manual Chapter 0540. Enclosure 1 is a brief summary of the safety reviews provided for FFTF to adequately assure protection of public health and safety during all phases of plant construction and operation.

It is also our intent to solicit NRC advice regarding the adequacy of the FFTF Technical Specifications. The Technical Specifications will be provided with the FSAR in March, 1976. Beyond the initial review of

the Technical Specifications, we anticipate that NRC involvement in final approval and review of future modifications would be requested on a case-by-case basis.

In addition as you suggested, we proposed to continue to address major safety issues through separate meetings with the NRC staff with the goal of resolution of these issues. In this regard, we have enclosed (Enclosure 2) a proposed schedule for resolving the decay heat removal by natural convection and the piping integrity issues.

Your letter also requested information about the format that has been utilized for the FSAR. The FSAR package is composed of three parts: (a) the FSAR, (b) the FFTF System Design Descriptions (SDD), and (c) supplementary documentation. Although we have not followed the NRC guidelines precisely, the FSAR portion of the package is a coherent document which contains a summary plant description and a complete plant safety evaluation. A list of major topic headings is enclosed (Enclosure 3). The SDDs are included as a part of the package to supply additional detail regarding the design requirements and description and are referenced as appropriate to supply back-up information which may be used as desired to supplement the summary description provided in the FSAR. The supplementary documentation listed in Enclosure 4 is included in the FSAR package to present complete detail of key topics which are only summarized in the FSAR itself. The FSAR also contains references to topical reports and supporting documents as necessary to provide technical justification and supporting analysis for the FSAR conclusions.

Finally, your letter of August 12, 1975, (reference (2)) requested additional information regarding the evaluations of the consequences of a non-mechanistically postulated failure of in-vessel post-accident heat removal. Specifically, you requested that existing documents and information be combined into a single cohesive document without reference to these previous documents. Your letter also requested that a number of areas be clarified and/or addressed in more detail in this document. Enclosure 5 summarizes the work that is being done to resolve the questions raised in your letter and indicates a schedule for completion.

Very truly yours,

R. L. Ferguson

R. L. Ferguson, Director
FFTF Project Office

FTF:GDB

FFTF SAFETY REVIEW

ERDA Manual Chapter 0540 defines the responsibilities and authorities to assure adequate protection of public health and safety for ERDA-owned reactors, such as FFTF. Listed below are the major organizations which are involved in the review of FFTF with a brief statement of the function(s) performed.

1. Assistant Administrator for Nuclear Energy - Authorization authority for FFTF operation.

2. Division of Reactor Research and Development (RRD) - Overall responsibility for FFTF including assurance that public health and safety is not affected by FFTF operation.

3. Division of Operational Safety (DOS) - Appraises RRD to assure safety responsibilities are being discharged adequately. Also provides information to RRD on certain safety matters.

4. Nuclear Regulatory Commission (NRC) - Provide advice on request from Director, RRD, regarding safety issues and safety of the FFTF design.

5. Director, FFTF Project Office, RRD - Direct responsibility for FFTF safety including all phases of planning, design, construction, QA testing, and operation.

6. Assistant Director, Reactor Safety, RRD - Responsible for conducting independent safety reviews of all phases of FFTF design, construction, testing, and operation.

7. Hanford Engineering and Development Laboratory (HEDL) - Responsible for operation of the FFTF. Provides independent safety review.

ENCLOSURE 2

10 1075

PLANS FOR EARLY RESOLUTION OF MAJOR FFTF SAFETY QUESTIONS

DECAY HEAT REMOVAL BY NATURAL CIRCULATION

Submittal of Final Documentation on Natural Circulation to NRC	February 1976
Meeting with NRC on Natural Circulation	April 1976
NRC Advice and Resolution	May 1976

PIPING INTEGRITY

Technology Meeting on Piping Integrity	November 1975
Submittal of Documentation on FFTF Piping Integrity	December 1975
Meeting with NRC	January 1976
NRC Advice and Resolution	March 1976

LIST OF MAJOR TOPIC HEADINGS IN THE FFTF FSAR

CHAPTER	SECTION	TITLE
1.0		Introduction and General Description of Plant
	1.4	Project Organization and Identification of Principal Contractors, Consultants and Vendors
	1.5	Research and Development
2.0		Site Characteristics
3.0	3.1	Design Criteria Structures Equipment and Systems
	3.2.1	Seismic Classification
	3.2.2	System Quality Group Classification
	3.3	Wind & Tornado Loadings
	3.4	Water Level Design
	3.5	Missile Protection
	3.6	Design Criteria for Sodium Spill Protection
	3.7	Seismic Design
	3.8	Structural and Mechanical Design Criteria
	3.9	Environmental Design of Mechanical & Electrical Equipment
4.0		Reactor
	4.1	System Design Basis
	4.2	Design Description
	4.3	Design Verification
	4.4	Design Safety Evaluation

CHAPTER	SECTION	TITLE
	4.5	Nuclear Evaluation
5.0A		Reactor Vessel
5.0B		Reactor Coolant System
6.0		Containment & Structures
	6.1	Reactor Containment System
	6.2	Structures Other than Containment
	6.3	Head Compartment
7.0		Instrumentation & Control
	7.1	Plant Protection System
	7.2.1	Reactor Plant Control System
	7.2.2	Closed Loop Instrumentation and Control System
	7.3	Instrumentation and Monitoring System
	7.4	Digital Data Handling System
8.0		Electrical System
	8.1	Offsite Power System
	8.2	Onsite Power System
9.0		Auxiliary Systems
	9.1	Maintenance Support System
	9.2	Auxiliary Liquid Metal System
	9.3	Inert Gas Receiving and Processing System
	9.4	Heating and Ventilating System
	9.5	Impurity Monitoring and Analysis System

CHAPTER	SECTION	TITLE
	9.6	Service Piping System
	9.7	Communication System
	9.8	Plant Fire Protection System
10.0		Closed Loop Systems
11.0		Fuel Handling System
	11.1	Reactor Refueling System
	11.2	In-Vessel Handling Machine (IVHM)
	11.3	Closed Loop Ex-Vessel Machine (CLEM) and Transporter
	11.4	Loop Preparation Equipment
	11.5	Interim Decay Storage (IDS)
	11.6	Conditioning Stations
	11.7	Floor Valves
	11.8	Plug Handling Fixtures
	11.9	Core Component Pot
	11.10	Interim Examination and Maintenance Cell Components
	11.11	Bottom Loading Transfer Cask (BLTC) and Dolly
	11.12	Reactor Support Building Handling Equipment
12.0		Radiation Protection
13.0		Conduct of Operations
14.0		Initial Test & Operation
15.0		Safety Analysis

CHAPTER	SECTION	TITLE
	15.1.1	Introduction
	15.1.2	Reactivity Transients
	15.1.3	Loss of Cooling
	15.1.4	Fuel Element Failure Propopagator (FEFP)
	15.1.5	Natural Phenomena
	15.1.6	In-Vessel Fuel Handling
	15.1.7	CLS Safety Analysis
	15.2	Ex-Reactor Safety Analyses
16.0		Quality Assurance
17.0		Technical Specifications (preliminary)
	App. A	Hypothetical Analysis for Test of Containment
	App. B	Piping Integrity Evaluation
	App. C	Fuel Element Failure Propagation (FEFP)
	App. D	Materials Specifications

FSAR PACKAGE DOCUMENTS

Seismic Design Criteria for the FFTF	JABE-WADCO-02
Design Thermal Transients for the FFTF	FFP-664
Revised Load Histogram for the FFTF HTS	FFP-904
Nuclear Systems Materials Handbook	TID-26666
Safety Assurance Summary	HEDL-TME-75-76
Evaluation of the FFTF PPS Response to Loss of Heat Removal Events	FPC-1024
Evaluation of the FFTF During Reactivity Transients	FRT-692
Users Guide for Irradiation of Experiments in the FTR	HEDL-MG-22
CLIRA Safety Analysis Report	HEDL-TC-190
FFTF Startup Test Plan	HEDL-MG-35
Loss-of-Flow (LOF) HCDA Analysis	
Report on the Core Disruptive Phase of an Unprotected Flow Coastdown Accident in FTR	ANL/RAS 74-16
Report on the Analysis of the Initiating Phase of a Loss-of-Flow (without scram) Accident in FTR	ANL/RAS 74-24
An Analysis of the Unprotected Transient-Over-Power (TOP) Accident for the FFTF	HEDL-TME 75-50
HCDA Requirements and Loadings for FFTF Design Evaluation	FS-257
Failure Modes and Effects, Analysis for the FFTF RSS and HTS	WARD-2171-48

PLAN FOR RESPONSE TO REQUEST FROM NRC FOR ADDITIONAL INFORMATION ON FFTF POST-ACCIDENT HEAT REMOVAL

TASK 1 -- Reactor Cavity Liner Evaluation

A liner integrity evaluation report will be issued as a "preliminary" report by November 30, 1975. Liner venting test will be completed and a vent system design finalized by April 30, 1976. The "preliminary" report will be updated to reflect the venting test and venting design details and will be issued as a "final" report by *June 30, 1976.*

TASK 2 -- Consolidation of Existing Information

The two HEDL preliminary reports,

(a) Post-Accident Heat Removal Containment Transients, and

(b) Radiological Evaluation of a Postulated FFTF Core Melt-Through Accident

will be updated to incorporate subsequent analyses which were reflected in later presentations or letters to NRC and will be issued as final reports by June 30, 1976.

TASK 3 -- Debris Formation Following a Postulated Melt-Through

ANL will prepare a letter report by March 1, 1975, on the current best assessment of debris formation and sodium availability following a postulated melt-through covering:

(a) penetration of the reactor vessel into the flat-bottom guard vessel, and

(b) penetration of the guard vessel onto the reactor cavity floor.

TASK 4 -- Evaluation of Containment Alternatives

Existing data will be evaluated and further analyses performed to establish containment operational requirements for postulated melt-through conditions and to assess whether additional design features such as venting, purging or atmosphere cleanup are warranted. A report on this task will be completed by December 1, 1976.

TASK 5 -- Clarification of Prior Information

NRC comments indicated a lack of understanding of some information which has been presented. A letter will be provided clarifying these points by December 15, 1975.

These tasks will address all of the points raised by the NRC letter of August 5, 1975, as shown below:

	TASK	NRC COMMENTS
1.	Reactor Cavity Liner Evaluation	7, 8
2.	Consolidation of Existing Information	Support task 4 and reflects NRC request for a "cohesive" package of documentation
3.	Debris Formation	1
4.	Evaluation of Containment Alternatives	4 (pt.), 5 (pt.), 6, 9
5.	Clarification of Prior Information	2, 3, 4 (pt.), 5 (pt.)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 11 1976

Project No. 448

Mr. Robert L. Ferguson
Director, FFTF Project Office
P. O. Box 550
Richland, Washington 99353

Dear Mr. Ferguson:

On January 20, 1976 copies of Appendix B of the FFTF FSAR, Primary Piping Integrity, were transmitted to NRC for its review in advance of the complete FSAR submittal. A meeting was subsequently held in Bethesda on February 25, 1976 between the NRC Staff and the FFTF Project (HEDL/ERDA) to discuss more fully the project's position and related technical information provided in Appendix B, and a meeting summary dated March 22, 1976 was prepared and transmitted for your information. The NRC Staff has presently completed its preliminary review of Appendix B and other information relating to this subject and its position and guidance is given below.

The NRC Staff has concluded that in their judgement a double-ended instantaneous pipe rupture in the cold leg of the main primary heat transport system has a sufficiently low likelihood of occurrence in the FFTF, and should not be considered an initiating event that could lead to core melt and disruption. The staff's judgement is based on the assumption that the inherently high fracture toughness of Type 304 stainless steel will be retained throughout the service life of the facility and that this will be verified by adequate material test and surveillance programs.

The NRC Staff's recommendations for FFTF are more fully described in the Enclosure to this letter and include a baseline preservice inspection, access for volumetric examination of highly stressed welds, demonstration of adequate sensitivity and reliability of the leak detection system, periodic material surveillance to assure long term piping integrity and establishment of a maximum leak rate for the technical specifications to initiate plant shutdown. The staff however believes that the consequences of small leaks or ruptures should be considered in the design and operation of the facility. The implementation of this will require further interactions with the project.

The staff requests that the FFTF Project (HEDL/ERDA) keep it informed on the disposition and implementation of its recommendation in regards to primary piping integrity, lead detection and inservice inspection for FFTF. It is the staff's opinion that only by assuring that the above provisions are implemented in a meaningful manner, can the instantaneous double-ended primary piping rupture be excluded as an accident initiating event.

In addition to the staff's preliminary evaluation, the Enclosure to this letter includes additional questions on Appendix B which should be answered. Appendix B of the FSAR should then be amended accordingly.

If you have any questions or need further clarification concerning this matter, please notify Mr. H. Holz, the Licensing Project Manager for the FFTF review.

Sincerely,

Original Signed By
Themis P. Speis

Themis P. Speis, Chief
Liquid Metal Fast Breeder
Reactors Branch
Division of Project Management

Enclosure:
As Stated

ENCLOSURE

STAFF EVALUATION REPORT

Report Title: Appendix B, "Primary Piping Integrity Evaluation"
Report Date: January 20, 1976
Originating Organization: ARD, Westinghouse
Reviewed BY: Materials Integrity Section, Materials Engineering Branch

SUMMARY OF APPENDIX B, "PRIMARY PIPING INTEGRITY EVALUATION"

This report presents information and logic to assure that there is no realistic potential for the instantaneous rupture of the primary heat transport system in the FFTF to cause a loss of core cooling capability. Assurance is provided by (a) the overall design of the system, (b) a comprehensive program for materials acceptance, (c) analyses of highly stressed areas and factors affecting the growth of small flaws in the piping to critical size, and (d) in the event of flaw growth, the piping wall would be penetrated locally through the wall causing a sodium leak which would be detected by sensitive, redundant, diverse, and reliable leak detecting systems long before the flaw could reach critical size (i.e., the size causing pipe rupture).

Chapter B2 - Description of Primary Heat Transport System

The primary sodium loops transport heat from the reactor to the intermediate heat exchanger. The sodium in the three primary loops has a common flow path through the reactor. Sodium flows each outlet nozzle through the hot-leg isolation valve to the variable speed centrifugal pumps. The pump discharge is circulated to the shell side of the intermediate heat exchanger where the heat is transferred to the secondary sodium. The primary sodium flows through a check valve, and then to the reactor vessel inlet.

The pipe from the reactor vessel to the pump suction is Type 316 H stainless steel, 28-inch O. D. The remaining pipe is Type 304 H stainless steel, 16-inch O. D. Features incorporated in the design which contribute to the ability of the plant to tolerate a range of sodium leaks with the continued capability of core cooling without damage include: (a) guard vessels for the primary components with guard pipe around vertical pipe, (b) elevated piping, (c) large available sodium inventory, and (d) low shut off pump head at pony motor speed.

Chapter B3 - Acceptance Criteria for HTS Piping

The primary piping system was designed and fabricated to specification HWS 1504, which requires that all forming, heat treating, cleaning, weld procedure qualification, and nondestructive test procedures be approved by ARD. The specifications imposed for the inlet downcomer piping were ASTM A 376-64, Type 304, and RDT Standard M3-3T, "Seamless Austenitic Stainless Steel Pipe (Modified ASTM 376)", February 13, 1969, or equivalent. RDT M3-3T requires that the pipe shall be ultrasonically examined 100% of its length in accordance with RDT F3-8T, "Ultrasonic Examination of Metal Pipe and Tubing for Longitudinal Discontinuities", Level 2. In addition, RDT M3-3T requires that all surfaces shall be liquid penetrant examined in compliance to RDT F3-6T, "Nondestructive Examination", (1969) and paragraph N-627-3 of the ASME Code, 1968 Edition.

The acceptance criteria for liquid penetrant examination for welds were the same as for pipe material. The welding was performed in accordance with the weld qualification procedure, Specification R-6.12.1(c), approved by ARD. The acceptance criteria for radiographic examination of welds is stated in RDT F3-6T. Subassemblies were hydrostatically pressure tested to the requirements of the ASME Code (373 psig), cleaned and shipped to the site.

The procedures for installation of the inlet downcomer were written by CE Erection Department and approved by ARD, HEDL, and RDPO. The installation welds were liquid penetrant and radiographically examined, approved by CE, Bechtel and a Code inspector.

Piping, fittings and components in the HTS outside the inlet downcomer were procured, fabricated and installed to the requirements of RDT F2-2, and to Class 1 requirements of the ASME Code, Section III, 1971 Edition through Summer 1971 Addendum. The manufacturers', fabricators', and installers' programs and procedures were reviewed and approved by the purchasing organizations. The pertinent RDT Standards were (a) M3-3T, "Austenitic Stainless Steel Seamless Pipe - Modified ASME SA-376", F2-4T, "Quality Verification Program Requirements", F3-6T, "Nondestructive Examination", F3-37T, "Special Requirements for Metal Products", F5-1T, "Cleaning and Cleanliness Requirements for Nuclear Reactor Components", and F7-2T, "Preparations for Shipment and Storage".

Welders and welding procedures were qualified in accordance with Section IX of the ASME Code. Audits were performed by HEDL and Bechtel Corporation to assure compliance with procedures, standards and specifications.

The reactor and heat transport systems were constructed from components designed, manufactured, and tested in accordance with Section III of the ASME Code. Prior to initial operation acceptance tests are scheduled to be conducted in the presence of a Code inspector. The requirements of Code Case 1595 will be met as follows: (a) an ambient temperature pneumatic test at 1.20 times the lowest design pressure (18 psig) and (b) an ambient temperature pneumatic test at 1.20 times the design pressure (300 psig) for each of the secondary loops. Upon completion of the pressure tests, the reactor plant will be provided with N-stamp certification.

The acceptance testing is planned to be conducted in three phases: (1) pressure tightness at 10 psig prior to argon purge and sodium fill, and dimensional check of components and piping at ambient temperature to provide a data base for thermal expansion and to verify installation, (2) dimensional check at 400°F, and (3) dimensional check at 400°F after sodium fill, and, if practical, at 800°F; demonstrate tightness of primary isolation valve; operation of HTS pumps to flow conditions at 800°F; cool system to 400°F and visually inspect select welds in highly stressed areas.

Chapter B4 - HEDL's Integrity Analyses of Primary Piping

The inlet downcomer will normally operate in a temperature range between 400° and 800°F and in a pressure range between 0 to 150 psig. Temperature transients may exceed 800°F. The criteria used for the design were for Class 1 components of the ASME Code, 1971 Edition including Addenda through Summer 1972, RDT Standard E 15-2T, and HEDL Seismic Design Criteria. Code Case 1331-5 and RDT Standard F9-1T were applied.

The loading conditions considered were (a) dead weight plus sodium, (b) dead weight less sodium, (c) seismic, (d) thermal expansion from 70° to 792°F, and (e) thermal expansion from 70° to 792°F with blocked-hanger and empty pipe. The stress intensities were calculated according to the rules of NB-3650 of the ASME Code, using Bechtel computer programs, ME632 and ELTEMP.

The results indicate that the inlet downcomers meet the structural requirements imposed by Code Case 1331-5, except the conditions requiring inelastic analyses. Stress analysis was performed by CE and approved by ARD. The report on the final stress analyses has not been issued. The preliminary results indicate that the criteria imposed by the governing documents are satisfied.

A circumferentially-oriented semi-elliptical surface crack located on the inlet downcomer elbow was assumed to represent the greatest potential and flaw growth. The pre-existing crack was assumed to have a length of 1-inch and a depth of 0.082-inch. The ratio of crack depth to wall thickness (a/t = 0.25). The hypothetical crack is the same as used for estimating the fracture behavior of heavy sections of light water reactors. Using an effective stress intensity factor of 8800 psi $\sqrt{in}$ and a conservative crack growth rate for Type 304 stainless steel, a growth of 1.03×10^{-5} inch/cycle was estimated. Assuming 695 reactor shutdowns, a total crack growth of 7.159×10^{-3} inch was estimated, indicating that the crack would not extend through the wall of the pipe.

Chapter 5B - Leak Detection

The fracture toughness of Type 304 H stainless steel was analyzed using the J integral method. At the piping thickness of interest (0.375 inch), a critical J value, Jc, of 4000 inch lb/in^2 was experimentally determined to result in crack initiation at 800°F. The test section of a 12-inch wide panel with a t-inch central crack became plastic before crack growth initiated. The Jc value is approximately equivalent to a K_c fracture toughness value of 385 $/\overline{in}$. The results indicate that the piping and elbow failure mode will be ductile and large scale plasticity would be experienced before fracture.

Eight one-quarter models of the inlet downcomer elbows were tested under combined bending and internal pressure. The models had a through-wall flaw. The mode of fracture was flaw bulging without crack advancement.

Using Paris' straight pipe model and a flow stress of 29,600 psi for estimating the burst strength, a critical flaw size 13-inches long for the design pressure, and 16-inches long for the operating pressure were calculated. The analytical and experimental studies showed that tension, bending and the combined tension-bending stress fields resulted in through-wall cracks prior to attaining the critical crack lengths.

The sodium leak detection methods rely on electrical conductivity, sodium aerosols, cavity and cell sampling, and remote visual examination for indicating leakage from the primary system. In general, transport and detection of sodium leakage are sufficiently understood to assure that if a small leak should occur within the reactor guard vessel, the leak would be detected before significant damage to the integrity of the primary piping would occur.

STAFF'S EVALUATION:

Appendix B presents information and the logic for accepting the concept of "leak before break" in the primary heat transport system of the FFTF. In our judgment, a double-ended instantaneous pipe rupture has a sufficiently low likelihood of occurrence, and should not be considered the initiating event that could lead to core melt and disruption. Our judgment is based on assumption that the inherently high fracture toughness of Type 304 stainless steel will be retained throughout the service life of the facility and that this will be verified by adequate material test and surveillance programs. It is also assumed that the leak detector system will be effective in identifying and locating small flaws before they can experience significant growth or cause deleterious effects to the piping system through accumulation of potentially corrosive materials on the external pipe surface.

Our review of Appendix B shows that the report has not adequately addressed the following areas:

1. Degradation processes that may affect the properties of piping materials,

2. Validation, of the sensitivity, and redundancy of the leak detection systems, and

3. Material surveillance and inservice inspection.

STAFF'S POSITION

We are currently developing Material Engineering Branch (MTEB) positions for inservice inspection and surveillance requirements for liquid metal cooler reactors. Our recommendations for the FFTF are as follows:

1. Perform preservice ultrasonic baseline inspection of the primary heat transport system (where possible, considering the status of construction),

2. Provide access for volumetric ISI of selected highly stressed welds,

3. Model a full size mockup of the reactor and guard vessel to demonstrate the sensitivity and reliability of the sodium leak detection systems in an operating situation and provide method of periodic verification of system operability,

4. Provide periodic systems integrity verification capability by using material surveillance speciments or by removing a highly stressed pipe section or fitting for metallurgical examination, and

5. Establish a maximum acceptable sodium leak rate to initiate reactor shutdown (technical specification).

121.0 MATERIALS ENGINEERING BRANCH (Appendix B, "FFTF Primary Piping Integrity Evaluation"

121.1 The referenced data on thermal aging effects on fatigue crack growth was obtained from relatively short time tests at 1000° or 1200°F. Describe the applicability of this data to the stated 800°F service environment. Discuss the effect of temperature on the distribution, composition and metallographic structure of Type 304, Type 304 H and Type 316 stainless steel, and relate these structural changes to the generation of fatigue cracks and crack extension rates. Discuss uncertainties in extrapolating short time data to predict long term performance.

121.2 The referenced data indicates that there is no difference in the crack propagation between weld material and base material (tests have not been done with aged weld material). Provide data to show that at high strain rates thermally aged weld material does not have an increased crack propagation rate that is larger than anticipated.

Also, discuss the effect of residual welding stresses (which may equal the material yield strength) on crack initiation and propagation.

121.3 A review of the submitted reference material indicates that reliability and maintainability of the sodium leak detection systems has not been adequately demonstrated. The system capability to detect small leaks should be verified under conditions simulating the primary heat transport system in length and configuration of the gas passages, thermal gradients and, for aerosol detectors, the length of the small diameter "sniffer" tubing with its thermal gradients. These tests should be done over the expected range of service environmental conditions. Additionally, verification should be provided to assure that:

(1) The leak detection systems will detect a new leak in the presence of an existing leak.

(2) Small leaks which may become plugged will be detected by two or more detection systems before significant caustic corrosion of pipe or component O.D. can occur (WARD tests have shown as much as 0.060 inch penetration in 500 hours).

(3) Remote testability of the detection systems by means other than electronic circuit checks are possible.

It is also necessary that maximum acceptable leak rate or conditions for reactor shutdown be established and maximum operating time with a known or suspected leak be defined.

121.4 The omission of volumetric examination from the inservice inspection program is not adequately justified. A more complete volumetric examination (preferably ultrasonic) should be conducted during the component fabrication, installation or as a preservice baseline to serve as a minimum substitute for volumetric inservice inspection. It is noted that current ASME Section III fabrication requirements for elevated temperature service include ultrasonic examination of all welds or radiography using two different angles.

121.5 In order to verify the applicability of short time, accelerated test data which is used as the basis for design/reliability calculations, a surveillance program consisting of removable primary heat transport system material samples or suitable test coupons exposed to the service environment should be required.

UNITED STATES
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
DIVISION OF REACTOR DEVELOPMENT AND DEMONSTRATION
FAST FLUX TEST FACILITY PROJECT OFFICE
P. O. BOX 550
RICHLAND, WASHINGTON 99352

AUG 20 1976

Nuclear Regulatory Commission
Washington, D. C. 20555

ATTN: Mr. R. P. Denise
Assistant Director
for Advanced Reactors

Gentlemen:

SCOPE OF NRC REVIEW OF FFTF FSAR

References: (1) Letter, R. S. Boyd to R. L. Ferguson, regarding the NRC review of the FFTF FSAR, dated August 13, 1976.

(2) Letter, R. L. Ferguson to R. P. Denise, regarding the NRC review of FFTF FSAR, dated November 13, 1975.

As a result of our discussion of July 21, 1976, in Bethesda, and the NRC letter of August 13, 1976, [reference (1)], we are providing clarification regarding the scope of the NRC review of FFTF that is being requested. As noted in reference (2), we desire a review of the FFTF design documentation and FSAR safety evaluations so that NRC may provide technical advice regarding the adequacy of the design to ensure safe operation of FFTF. NRC need not review "as built" configurations, reactor operations, or acceptance test results to verify that the plant is constructed and operated in accordance with design criteria and documentation. These latter functions are provided by ERDA in accordance with applicable ERDA Manual Chapters.

Regarding certain specific questions about the scope of the NRC review, the following clarifications are supplied:

1. Site Information (FSAR Chapter 2.0) - As was noted in reference (1), a complete reevaluation of siting considerations and documentation by NRC, in accordance with the NRC Standard review plan, is not necessary at this time because the site suitability was established by NRC and ERDA evaluations during the PSAR review and has been continually reviewed by the FFTF project contractors and ERDA, since the construction authorization.

2. Design Evaluation FSAR (Chapter 3.0 - 12.0 and 15.0) - We desire a thorough technical review of the design documentation and the level 1, 2, and 3 safety evaluations for the reactor and plant systems. Generally, we would like the review of the design to be comparable in technical detail to a licensed plant review. NRC reviews of special topics or issues that have been conducted during or since the construction authorization need not be repeated. Areas that we believe have been generally resolved and require limited NRC reevaluation at this time include HCDA energetics, HCDA design margins, and piping integrity. A few follow-on issues in these areas remain to be resolved, for example, a reassessment of the need for improved containment safety margins is still planned for completion in December 1976. Reviews will, of course, continue to ensure that new experimental information does not significantly affect previous conclusions with regard to adequacy of the FFTF safety margins. Decay heat removal issues are currently under review by the NRC staff, and we hope to resolve these issues in the near future. Specific questions regarding the desired scope or nature of the NRC review in other specific areas within these FSAR chapters can be discussed on case-by-case basis.

3. Operations and Start-Up Testing (FSAR Chapters 13 and 14) - Chapters 13.0 and 14.0 of the FSAR summarize ERDA practices and plans for operations and start-up testing. Although comments and advice regarding our current planning as outlined in these chapters would be welcomed, NRC need not provide a full review in these areas. The organizational structure, planning and control of FFTF operations and start-up testing programs will be implemented by ERDA in accordance with ERDA Manual Chapters in order to ensure safe plant testing and operations.

4. Quality Assurance (Chapter 16.0) - Although we plan to continue to utilize our current quality assurance standards (RDT F2-2, etc.) rather than 10CFR50, Appendix B, ANSI Standards and Regulatory Guides, we do desire a complete review by NRC of the adequacy of our quality assurance program. To facilitate the NRC review. the quality assurance FSAR chapter is being revised to provide certain additional information identified through informal discussions with the NRC staff. Also, we will provide a comparison between the FFTF quality assurance program and the NRC standard review plan requirements. Any areas where there are differences between FFTF controls and accepted NRC criteria will be resolved in the course of the NRC review.

5. Technical Specifications (FSAR Chapter 17) - NRC advice on the FFTF Technical Specifications will be solicited. Beyond the initial review of the Technical Specifications, we anticipate that NRC involvement in final approval and review of future modification would only occur if requested on a case-by-case basis.

Answers to the QO questions, cross references between the FSAR and the NRC Standard Format, and responses to certain other points raised by your August 13, 1976, letter are being prepared. Completion of these tasks is targeted for October 1, 1976. Confirming a telephone conversation with Mr. H. Holz of your staff, we would like to arrange a meeting at your earliest convenience to finalize a schedule for the NRC review. As indicated in earlier correspondence, we wish to complete the NRC review approximately two years from the FSAR submittal date.

If there are any questions regarding this matter, please contact Dr. G. D. Bouchey of my staff.

Very truly yours,

R. L. Ferguson, Director
FFTF Project Office

FTF:GDB:E611

cc: H. Holz, NRC (5)
T. Speis, NRC
H. Gearin, NRC
S. A. Weber, HEDL
D. E. Simpson, HEDL

UNITED STATES
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
DIVISION OF REACTOR DEVELOPMENT AND DEMONSTRATION
FAST FLUX TEST FACILITY PROJECT OFFICE
P. O. BOX 550
RICHLAND, WASHINGTON 99352

JUL 14 1977

Nuclear Regulatory Commission
Washington, D. C. 20555

ATTN: Mr. R. P. Denise
Assistant Director
for Advanced Reactors

Gentlemen:

FFTF SAFEGUARDS AND SECURITY

Reference: Letter, T. P. Speis to R. L. Ferguson, "Transmittal of May 26, 1977 Meeting Summary", dated June 24, 1977.

The reference letter requested that ERDA advise NRC of any plans for requesting NRC staff review of FFTF safeguards and security provisions.

The FFTF safeguards and security reviews and assessments are conducted by ERDA personnel in accordance with ERDA Manual Chapter requirements. Confirming our statements in the May 26, 1977 meeting, it is not necessary for the NRC review of FFTF to include consideration of this area.

The safeguards and security programs and provisions that are being implemented for FFTF meet all pertinent policies, requirements, and procedures defined in ERDA Manual Chapter 2401, 2405, 7401, and 7451. Although the recent (March 1977) 10 CFR 73 changes related to security and other NRC guides and standards in this area are not specifically imposed as requirements for FFTF, ERDA and ERDA Contractors are well aware of the NRC requirements and guides and these have been considered in the formulation of the FFTF security plans. In our opinion, safeguards and security arrangements for FFTF will be at least as effective (though not necessarily identical) as those currently being implemented for licensed reactors.

Very truly yours,

R. L. Ferguson, Director
FFTF Project Office

FTF:GDB:E1164

UNITED STATES
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
DIVISION OF REACTOR DEVELOPMENT AND DEMONSTRATION
FAST FLUX TEST FACILITY PROJECT OFFICE
P. O. BOX 550
RICHLAND, WASHINGTON 99352

AUG 31 1977

Nuclear Regulatory Commission
Washington, D. C. 20555

ATTN: Mr. R. P. Denise, Assistant Director
for Advanced Reactors

Gentlemen:

RESPONSE TO ITEMS IDENTIFIED AT THE JUNE 30, 1977, NRC/ERDA/HEDL MEETING ON HOT CHANNEL FACTORS

Reference: NRC Letter, T. P. Speis to R. L. Ferguson, "Transmittal of Meeting Summary (Fast Flux Test Facility)," dated July 27, 1977.

The referenced letter summarized the requests for additional information made at the June 30, 1977, meeting on hot channel factors. In addition to the requests made at the meeting, the referenced letter requested that the FFTF Project (1) state in writing the Project's commitment that in the event natural convection does not prove to be a reliable and safe way to remove residual heat from the reactor core, engineered safety features will be provided, as necessary, to assure this function and (2) compare data taken during startup with core thermal hydraulic and safety design values and that trip levels be set conservatively until sufficient operating experience is gained to warrant the use of nominal trip values.

In response to the information requests made at the June 30, 1977, meeting, we are enclosing 20 copies of supplemental responses to NRC questions 221.25, 221.54, and 221.65 along with 3 copies of ANL report ANL-ETD-71-03, "Irradiation of Alnico Magnets in EBR-II" which provides the information requested under item 221.6 in the referenced letter. The information requested under item 221.82 is contained in the response to NRC question 221.97 (submitted to NRC as part of Supplement 11 to the FSAR).

The information requested in items 221.2, 221.5, 221.28, and 221.86 will be provided by September 9, 1977, and in item 221.66 by October 28, 1977.

In response to the two additional requests contained in the referenced letter, it is our intent to proceed as follows:

1. Adequate decay heat removal capability will be provided for FFTF. One important part of the FFTF decay heat removal system design is natural convective cooling following total loss of offsite and

onsite electrical power. Key features of this capability will be confirmed via the FFTF Acceptance Test Progran natural convection tests. If necessary as the result of these tests, those plant modifications required to assure this capability will be implemented. Depending on the results from the natural convection tests, these modifications could take the form of either physical plant modifications to enhance natural convection or the addition of features to enhance the reliability of forced circulation cooling.

2. Conservative plant protection system trip settings will be used during the first ascent to full power, while collecting data during the ascent to compare with predictions. The full power trip levels will be set at their nominal values only after operation at full power has confirmed this to be acceptable. Detailed plans concerning the above are currently in preparation.

If you have any further questions, please do not hesitate to contact us.

Very truly yours,

for R. L. Ferguson, Director
FFTF Project Office

FTF:TLK:E1246

Enclosures:
1. Responses to NRC Questions 221.25, 221.54, & 221.65
2. Sodium Orifice Erosion Test
3. Irradiation of Alnico Magnets in EBR-II

cc: w/o encl.
H. Gearin, NRC
H. Holz, NRC
T. Speis, NRC
D. E. Simpson, HEDL
S. A. Weber, HEDL
R. A. Passman, RDD-HQ
H. Hollister, DOES-HQ

JAN 17 1978

Project No. P-448

Mr. R. L. Ferguson, Director
Fast Flux Test Facility
Project Office
Department of Energy
P. O. Box 550
Richland, Washington 99352

Dear Mr. Ferguson:

In a meeting with members of your staff on May 26, 1977, the subject of FFTF safeguards and security was briefly discussed. I indicated that I believed that it would be beneficial to the FFTF Project for the NRC staff to evaluate this area, and that if we did not, this would be an area where NRC staff review of the FFTF would not be fully comparable to that performed for a licensed facility. I requested that the FFTF Project Office inform the NRC staff of their plans in this matter. Your letter of July 14, 1977 indicated that it was not necessary for the NRC review of the FFTF to include consideration of this area, and that, in your opinion, the safeguards and security arrangements for FFTF will be at least as effective (though not necessarily identical) as those currently being implemented for licensed reactors.

In a meeting on September 27, 1977, we further discussed this subject. At that time you indicated that:

(a) The FFTF security and safeguards arrangements are an integral part of the Hanford Reservation security and safeguards arrangements, and that the FFTF arrangements could not be satisfactorily reviewed without also reviewing the Hanford Reservation arrangements;

(b) The FFTF/Hanford Reservation safeguards and security plans had been reviewed by a number of independent DOE organizations and contractors, and you did not anticipate any significant additional benefit from an NRC review; and

(c) You were concerned that an NRC review of the FFTF/Hanford Reservation security and safeguards arrangements could result in an unwarranted exposure of these plans, which would be detrimental to effective security.

I indicated that we would consider this matter further, and provide a response.

Since the NRC staff has not reviewed your plans, we are unable to reach any conclusion regarding their effectiveness compared to those of licensed facilities, nor are we able to reach any conclusion on the completeness or adequacy of the evaluations performed by DOE and its contractors. We do, however, believe that information on the FFTF/Hanford Reservation security and safeguards arrangements could be adequately protected from disclosure if subjected to NRC review.

Since DOE clearly does not wish NRC to review the security and safeguards arrangements for FFTF, we will proceed with our evaluation of the FFTF without including this matter. We will not, therefore, reach any conclusions regarding the adequacy of provisions for safeguards and security for FFTF.

If you have any questions regarding this subject, please contact me.

Sincerely,

Original Signed By
Richard P. Denise

Richard P. Denise, Assistant Director
for Special Projects
Division of Project Management

cc: Mr. L. M. Brenner
US Department of Energy
Safeguards and Security
MS A2-1016
Century 21 Building
Washington, DC 20545

Department of Energy
Fast Flux Test Facility Project Office
P. O. Box 550
Richland, Washington 99352

JUN 19 1978

Nuclear Regulatory Commission
Washington, D. C. 20555

ATTN: Mr. William Gammill, Assistant Director
for Standardization and Advanced Reactors
Division of Project Management

Gentlemen:

NRC REVIEW OF CLOSED LOOP PLANT PROTECTION SYSTEM DESIGN

Recently, we received a verbal request for the design basis analysis for the Closed Loop System Plant Protection System (PPS). The requested information is not complete at this time. The current planning projects the first in-core Closed Loop testing no sooner than irradiation Cycle 7 due to experimental program funding priorities. Consequently, design analysis of the Closed Loop Plant Protection System has not yet been complete due to higher priority work items.

Since Closed Loop experiments will not be included at startup of FFTF, we request that review of the Closed Loop System PPS design be excluded from the scope of NRC review currently in progress.

Appropriate reviews will be performed upon completion of needed analyses prior to installation of Closed Loop experiments in FFTF.

If you have any questions regarding this matter, please feel free to contact Dr. G. D. Bouchey of my staff.

Very truly yours,

A. J. Rizzo, Deputy Director
for Engineering
FFTF Project Office

FTF:GDB:E428

APPENDIX C

Sodium-Concrete Interactions

As discussed in Section 15.3.6 (Containment Response to a Core Meltdown Event) for cases where a self-leveling, sodium-flooded, coolable fuel debris bed is formed, the severity of the containment transient following a postulated core meltdown event is very dependent on the rate and extent of sodium penetration into the reactor cavity concrete floor. This appendix presents the staff's and its consultant's assessment of the state of technology on sodium-concrete interactions in connection with evaluations of FFTF containment margins.

The Effect of Liquid Sodium on Concrete

A number of relatively small-scale tests have been performed by the Handford Engineering Development Laboratory[1] (HEDL) to measure the penetration rate of liquid sodium into magnetite and limestone concretes. A few much larger scale tests of the penetration rate of liquid sodium into limestone concrete have been carried out at Sandia.[2,3] The reported penetration of sodium into concrete by HEDL and Sandia differs significantly, suggesting that there may be a scale factor involved in the penetration mechanism. The HEDL experiments suggest that the reaction of sodium with concrete is self-limiting to a penetration depth which does not exceed 3 in. On the other hand, Sandia's first three tests suggest that penetration proceeds until all the sodium is consumed by the reaction and extends to a depth approximately equal to the initial depth of the sodium pool, 6 in. in the third Sandia test. We understand that later, undocumented, Sandia tests* showed sodium penetrations of less than 6 in.

HEDL Experiments

The twelve HEDL tests reported in Ref. 1 employed 40 to 50 pounds of sodium heated to either 676°C or 871°C while in contact with vertical and horizontal surfaces of magnetite and limestone concrete. In each case, the concrete sample had a volume of 1 cu. ft. and an exposed surface area of 1 sq. ft. The temperatures within the concrete and sodium were monitored by thermocouples and the gas that evolved was analyzed. At the end of the test periods, the excess sodium was decanted, the sample was allowed to cool, and the remaining concrete and reaction products were examined.

*This assessment includes results from only the first three Sandia tests. At least three others have been performed, but the details are not available at the time of this writing.

It was observed that the nonviolent reaction between the concrete and sodium produced a viscous, wide melting point range liquid which had a density between that of sodium and concrete. In the horizontal surface tests, this viscous fluid formed a layer separating the sodium from the concrete. It was felt that this layer protected the concrete from further chemical attack and also served as an inert insulator, thereby limiting the extent of the sodium-concrete penetration.

Equations were fitted to the data from the magnetite and limestone concrete experiments. For magnetite concrete, the equation used is:

$$X = 1.75\ (1-e^{-.2t}) \qquad (1)$$

where X is in inches and t is in hours. The corresponding equation for limestone concrete is:

$$X = 1.64\ (1-e^{-.4t}) \qquad (2)$$

It was assumed in the selection of these equations that the sodium-concrete reaction was zero order with a first order retardation effect.

When the temperature was increased from 676°C to 871°C, the penetration increased approximately 50%. These data, when extrapolated suggest that the ultimate penetration depths never exceeded 3 in. for horizontal specimens. In addition, some cracking and spalling was noted which led HEDL to comment: "some cracking and spalling characteristics of the concrete specimen appear to provide a major influence on the rate and ultimate penetration depth of the sodium-concrete reaction."

In the vertical reaction tests, it was noted that the viscous reaction fluid settled toward the bottom of the reaction vessel giving rise to a higher penetration rate. The observed penetration rate into a vertical surface was determined to be 2.5 times that into a horizontal surface. This is consistent with the idea that the reaction products protect the concrete from the sodium.

Differences Between the Sandia and HEDL Experiments

The Sandia experiments[2,3] differ in several important details from those of HEDL. The sizes of the Sandia tests were considerably larger, with 190 Kg of sodium used in the third test as opposed to 23 Kg in the HEDL tests, and with a limestone concrete crucible with an exposed surface area of 16 ft^2 and a volume of 57 ft^3 as compared to the HEDL test area and volume of 1 ft^2 and 1 ft^3, respectively. The initial sodium temperature was 550°C, and the sodium cooled to approximately 450°C on contact with the cool concrete. An exothermic reaction, occurring 15 min. later, raised the sodium temperature to nearly 800°C and continued until all the sodium was consumed. The reaction was accompanied by loud noises every few

seconds which may have been the result of explosive spallation. The penetration in the horizontal surface progressed well past 3 inches. However, penetration into the vertical surfaces was less than that in the horizontal surface, unlike the HEDL experiments. Also, in the post-test analysis, an apparently spalled, unreacted chunk of concrete, 25 cm in diameter by 7 cm thick, was found in the solidified reaction zone. It is suggested that this chunk spalled at that time when the last of the sodium was being consumed by the reaction. Finally, Sandia, unlike HEDL, reported vigorous stirring and mixing of the concrete reaction products by gases released from the heated concrete. They suggested that gas released near the vertical walls preferentially flowed along these walls thereby protecting them from sodium chemical and thermal effects and thereby reducing penetration.

To explain the above differences, Sandia has proposed a two-phase reaction mechanism. Phase 1 is the initial, quiescent period consisting of the reaction between sodium and water to form NaOH, which is soluble in sodium to about 10 wt% at 500°C. Phase 2 starts when the sodium becomes saturated with NaOH, which precipitates out and sinks to the bottom of the pool because of its higher density. The NaOH layer can directly attack the $CaCO_3$ in the limestone concrete because $CaCO_3$ dissolves readily in hot NaOH. The ratio of sodium to concrete determines whether the interaction remains in Phase 1 or proceeds to Phase 2. If the ratio of sodium to concrete is large, the sodium will not become saturated with NaOH, and the penetration is limited due to the termination of the reaction by reaction product buildup. For small sodium to concrete ratios, the sodium will eventually become saturated with NaOH which leads to Phase 2 and aggressive concrete attack. Sandia proposes that they are observing more extensive attack because of low sodium to concrete ratios which progress the reactions into Phase 2, while the HEDL test results are primarily confined to Phase 1 because of the large sodium to concrete ratios. The mechanism proposed by Sandia could explain the previous HEDL F-1 test,[4] which resulted in extensive concrete damage; however, this test employed basaltic type concrete for which the amount of $CaCO_3$ is much lower (i.e., basaltic concrete is primarily SiO_2). In the F-1 test, extensive concrete spalling also occurred which provides more rapid and greater amounts of water resulting in reaching NaOH saturation sooner thereby getting into Phase 2.

Assessment of HEDL and Sandia Sodium-Concrete Experiments

Clearly, the large-scale tests at Sandia differ significantly from the smaller HEDL tests. Both the Sandia and HEDL reactions appear to start out at approximately the same rates. After some 15 mins., the Sandia reactions accelerated markedly as they entered what they have referred to as a Phase II reaction. Preliminary results with NaOH added to a sodium-concrete test cell at HEDL appear to indicate that, under the conditions employed at HEDL, NaOH has little or no effect in accelerating the reaction rate. Other competing mechanisms, such as spallation, may be important erosive processes in this material. Furthermore, the Phase II mechanism appears to be applicable only to limestone concrete.

In the HEDL tests, the surface was starved for sodium due to the presence of the reaction products. There was an excess of sodium but, because of the quiescent conditions, it couldn't reach a fresh concrete surface to react. Also, gas could more easily diffuse out of a small test particle. In a larger specimen, this gas could cause spallation if the pressure exceeded the concrete tensile strength.

In the Sandia tests, it appears that the vigorous agitation of the material above the concrete may not allow a protective layer of inert reaction products to remain intact. This agitation could bring fresh sodium to the concrete surface until all the sodium is consumed.

Both the HEDL and Sandia tests have produced thermally-induced cracks in the concrete, but these do not appear to contribute significantly to the penetration rate. The sodium does not wet the concrete and does not appear to enter the cracks. In addition, these cracks are most likely low impedance paths for gas being released from the heated concrete. One would expect this gas to expel any sodium which might otherwise tend to enter the cracks. With the much greater depth of sodium in FFTF (16 ft. for the case where all of the reactor vessel inventory spills on the reactor cavity floor), the hydrostatic pressure will be approximately 8 psi, and this may be sufficient to force some sodium into the cracks and enhance erosion.

Some small-scale sodium-concrete reaction tests performed at the Aerospace Corporation,[5] with 2 in. diameter specimens were consistent with the HEDL results for limestone concrete. The extent of penetration was similar, and a self-limiting effect was observed. Comparing the Aerospace, HEDL and Sandia data, it seems that some of the differences may be due to the scale of the experiments.

We believe that spalling may contribute significantly to the greater erosion observed in some of the Sandia experiments. If the spalled chunk of concrete found in the residue of the third Sandia experiment,[3] is at all typical, scale effects may be important in the erosion of concrete by sodium. Since the chunk was 25 cm in diameter, nearly equal to the entire reacting area of the HEDL tests, one would not expect to see spallation on this scale in the HEDL experiments. It is expected that such a chunk will form only in an area in which the internal pressure of the concrete had not been relieved by thermal cracking. If such a chunk were to spall and were raised an inch or two, the reacting surface area locally would be more than tripled (the original surface and the two opposing fractured surfaces, each somewhat larger than the original). The gas flow at the edges of this chunk would be substantially increased, forming an annulus of gas which would disrupt the protective layer of reaction products. The increased transport of sodium to the freshly exposed concrete would liberate heat at a greater rate favoring more spallation locally.

Penetration Rates and Final Penetration Depth

There is good data available for the penetration of sodium into small surface area concrete samples. The HEDL results indicate that at 871°C, sodium should not penetrate beyond 3 in. into a 1 sq. ft. concrete sample. With substantially larger surfaces, Sandia found in their first three tests, that the penetration extended approximately 6 in. after three hours. We tend to favor the Sandia penetration rate findings since the experimental size is closer to the size of the reactor cavity.

HEDL fitted the first three Sandia experiments to a function, Equation (3), based on their small-scale experiments:

$$X = a_1 (1-e^{-a_2 t}) \qquad (3)$$

Values of the coefficients a_1 and a_2 were found to be 12 in. and 0.5 hours. If these assumptions are correct, then the penetration function predicts that sodium penetration will not exceed 12 in.

The constants in Equation (3) above were determined empirically from limited data from experiments with durations as long as 24 hours. It was assumed in the development of these equations that the sodium-concrete reactions were zero order with a first order retardation effect due to reaction product accumulation. However, with the complicated system of reactions known to occur between sodium, NaOH and concrete, the reaction rate is probably more complex than Equation (3) indicates. Each step in the reaction process will have its own rate equation and velocity constant, and the slowest reaction will determine the rate of the overall system.

If the diffusion or transport of sodium through the reaction product layer, rather than chemical kinetics, is the rate-controlling factor, then the reaction rate might approximate a first order function plus a correction for reaction product accumulation. Under the quiescent conditions of the HEDL experiments, where the inert reaction product layer separated and insulated the concrete surface from the excess of sodium, either diffusion or chemical kinetics might be the limiting factor. However, in the more violent conditions of the larger scale Sandia test, the reaction seemed to be limited not by the formation of a protective layer of reaction products, but rather by the exhaustion of the sodium supply.

The results of the three Sandia "large-scale tests" with limestone concrete are as follows:

Test	Duration (min.)	Penetration (in.)
I	22.	3.25
II	40.	3.58
III	180.	6.0

These data, by themselves, do not suggest a saturation effect leveling out at 12 in. penetration. Later Sandia data, the details of which are not yet available, apparently do not show a penetration extending beyond 6 inches. HEDL's proposed "fit" of the three Sandia data points to their function, Equation (3), is unconvincing. Using the same three Sandia test results, a better fit, for example, can be obtained by using a straight line on a log-log plot, such as the function, $X = 4.3t^{(7/2)}$, indicating that penetration continues to increase with time.

We believe that under the experimental conditions of the Sandia tests, the reaction rate will eventually decrease as the result of the lowered concentration of sodium in the debris above the unreacted concrete. The question is how long it will take for this to occur. Given an excess of sodium, and a deep pool, as might be expected in a core meltdown event, the penetration may be greater than that observed by Sandia and may proceed well beyond the 12 in. suggested by HEDL's equation. Larger scale and long-term tests, on the order of 100 hours, are needed to provide data on the total penetration of sodium into basalt type concrete.

A serious difficulty in trying to use the available data occurs because of the small number of experiments. With only one data point for each set of conditions, the data are not statistically meaningful. Additional experiments are needed for each set of conditions.

The data obtained thus far are for experimental conditions vastly different from those existing following a core meltdown event. The trend in the data indicate that rate of penetration increases with temperature, area of exposed surface, and possibly the depth of the sodium pool (the latter due to hydrostatic pressure and convection). The Sandia data were obtained under conditions more closely approximating a core meltdown event. If no new size-related effects become operative under the conditions of a meltdown event, the penetration rate for limestone concrete should be initially that of the Sandia tests and decrease with time as the sodium is consumed and the reaction products build up in some way to protect the remaining unreacted concrete.

Staff Recommendation on Sodium-Concrete Penetration Rate

At present, there are very limited data points for FFTF basalt type concrete, and this data can be interpreted in different ways. In the absence of large-scale, long-term tests clearly showing a self-limiting reaction in the time frame of

interest (~ 100 hours), we must recommend the use of a constant rate of attack for basalt concrete. Furthermore, there is also a lack of long-term data showing a self-limiting reaction for limestone concrete. In view of the differences between the HEDL and Sandia data, reasonably conservative results should be used until there is a satisfactory explanation of the data. Currently, there are conflicting explanations (two-phase reactions versus scale effects) for the difference between the Sandia and HEDL results. Although the available data is insufficient to provide a good basis for a recommendation on the penetration rate of sodium into concrete, a reasonably conservative staff recommendation is provided below until more and better data become available.

For limestone concrete, the rate of attack by sodium, based on the Sandia test, should be on the order of 7 in. per hour for the first half-hour with a subsequent rate of 1 in. per hour thereafter. A comparison of the Sandia and HEDL data for limestone concrete suggests that a scale factor of approximately four may be appropriate. If this is applied to the HEDL magnetite concrete data, the constant sodium penetration rate would be 2 in. per hour and for basalt concrete, the constant sodium penetration rate would be 3 in. per hour. However, the use of such a factor may be arbitrary. Consequently, we consider a penetration rate of 7 in. per hour for the first half-hour followed by a rate of 1 in. per hour thereafter to be appropriate for basalt and magnetite concretes being employed in the FFTF. While a leveling off of the penetration probably occurs due to reaction product buildup, we do not believe it is possible to say at present, when it will occur. Consequently, in the absence of better data, a constant rate should be used.

In the future, experiments are needed with longer exposure to sodium. The experiments should be of the largest size that is practical and should be at least the size of the Sandia experiments. The pool depth should be at least two feet so that a substantial portion of the wall will be exposed to sodium. Based on the Sandia experience, this might enhance spallation. With the greater depth, convection would be more significant. Convection currents are another possible contributor to concrete erosion as they may sweep away reaction products from the center toward the sidewalls, thus exposing fresh concrete to attack by sodium. In addition, the pool and concrete vessel should be presssurized to simulate a greater depth of sodium which might push sodium into any cracks that develop.

The NRC Division of Reactor Safety Research is supporting a confirmatory research program at Sandia Laboratories on sodium-concrete interactions. In order to provide an independent assessment of the HEDL data, we requested researchers at Sandia Laboratories to meet with HEDL personnel to review their sodium-concrete test results. Attachment 1 to this appendix presents Sandia's assessment of HEDL's experiments on sodium-concrete interactions. As indicated in this attachment, Sandia has provided a delineation of shortcomings in regard to HEDL's experimental design and data which makes application to FFTF accident

conditions very questionable. We believe that the Sandia assessment provides further evidence as to the need to take a reasonably conservative position on the sodium-concrete penetration rate until larger-scale, longer-duration data become available.

References

1. "Intermediate Scale Sodium-Concrete Reaction Tests," HEDL-TME-77-99, August 1977.

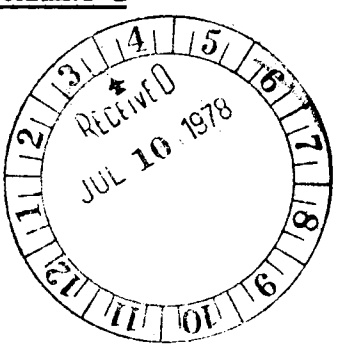
2. Memorandum to T.P. Speis, Chief, LMFBR Branch, NRR, from M. Silberberg, Chief, Experimental Fast Reactor Safety Research Branch, RES, dated August 5, 1977, "Quick-Look Reports on Sandia Preliminary Sodium-Concrete Interaction Tests 1 and 2.

3. Memorandum to T. P. Speis, Chief, LMFBR Branch, NRR, from M. Silberberg, Chief, Experimental Fast Reactor Safety Research Branch, RES, October 31, 1977, "Quick-Look Report on Sandia Preliminary Sodium-Concrete Interaction Test 3.

4. "FFTF Secondary Sodium Fire Protection System Test FI, "HEDL-TME 73-48, April 1973.

5. Swanson, D. G., et al., "Annual Progress Report Evaluation of Materials for CRBRP Core Retention," The Aerospace Corporation, NUREG/CR-0076, May 1978.

ATTACHMENT 1



Sandia Laboratories
Sandia Corporation
Albuquerque, New Mexico 87115

June 30, 1978

M. Silberberg, Chief
Experimental Fast Reactor
Safety Research Branch
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Silberberg:

An information exchange meeting was held on June 14 - 15, 1978, between Sandia and the Hanford Engineering Development Lab on sodium-concrete interactions. This letter summarizes Sandia's assessment of the HEDL experiment.

HEDL reported the results of 13 sodium-concrete interaction experiments - 4 on limestone concrete and 9 on basalt concrete. Sandia feels that, in FFTF considerations, the limestone results are a moot point. Based on the results of their basalt concrete test series, HEDL has concluded that the sodium-concrete reaction is self limiting and that the equation

$$E = 12\left(1 - EXP\left(-\frac{T}{2}\right)\right)$$

where E = erosion in inches
T = time in hours

is a conservative upper limit of erosion - bounding all known experimental data.

Sandia's position can be summarized as follows:

1. Basalt concretes do not seem to offer a basis for rapid, energetic reactions with sodium such as those experienced with limestone concretes. However, no large scale tests have yet been performed on basalt concretes.

2. There is concern that HEDL's fixed sodium pool depth (0.3 meters) has left unanswered the question of pool viscosity. "Additional sodium may produce a more dilute sodium - reaction product mixture and, thus lead to a more sustained reaction."

3. Tests of durations longer than 24 hours should be performed.

4. Additional thermocouples, closer to the reaction face in the concrete, would significantly enhance the heat transfer analysis.

5. More lower temperature information would add to the basic understanding of the chemical reactions and their kinetics.

These points have been discussed with HEDL. It is our understanding that HEDL is fabricating a large scale test facility, with the first test planned for September. They also are planning a 100 hour test with twice the sodium pool depth. Their plans concerning points 4 and 5 are not known.

Devil's Advocate questions that could be asked concerning the FFTF type concretes are:

1. Sodium bulk cooling effect. In an actual reactor situation, the volume of sodium will probably be large enough that the sodium will not be appreciably cooled by contact with the concrete. In reasonably scaled experiments, the sodium is cooled by the concrete and must be brought back to the desired temperature. Might not the reactor situation lead to increased spallation and greater erosion than the experiments?

2. Is high temperature (871°C) sodium the worst case? Are the number of experiments that have been performed at lower temperatures adequate to answer this question?

3. Vertical versus horizontal concrete surfaces penetration by sodium is confused. HEDL results show vertical penetration 2-1/2 times greater than horizontal penetration. HEDL vertical experiments had a sodium-gas interface, whereas their horizontal experiments did not. The same is true of Sandia experiments, however, the Sandia vertical walls were curved surfaces.

4. Only six magnetite concrete experiments have been performed. None of these experiments were large scale or for longer than 24 hours. Further testing of magnetite concrete has not been indicated by HEDL. Magnetite concrete forms the vertical walls in FFTF design. HEDL research indicates maximum erosion occurs on vertical concrete surfaces.

Should you need additional information, please call me on 475-6162.

Sincerely,

Russell Acton
Thermal Test Planning
and Analysis Division 1537

RUA:1537:jr

Distribution:
1537 N. R. Keltner
1537 R. U. Acton
5167 J. E. Smaardyk
5420 J. Walker
5422 R. L. Coats
5831 R. A. Sallach

APPENDIX D

Molten Fuel/Steel-Concrete Interactions

Because of the uncertainties that exist in certain aspects of core meltdown phenomena, two scenarios were evaluated in Section 15.3.6 (Containment Response to a Core Meltdown Event), namely a coolable particulate debris bed and a molten pool. In the coolable debris bed scenario, the core debris fragments, self-levels and forms a sodium-flooded coolable debris bed configuration on the lower cavity floor. The concrete will then be principally attacked by sodium. On the other hand, if the core debris does not fragment or is unevenly distributed, it may remelt due to decay heat and this would expose the concrete to a molten fuel/steel pool. The concrete may also come in contact with a molten pool if the fuel/steel mixture reaches the concrete before the sodium. Based on the limited small-scale experimental information on fragmentation,[1] the staff can not say at this time which of the above two scenarios is much more likely. Accordingly, Section 15.3.6 evaluated the FFTF containment response to those two possible scenarios, and this appendix presents an assessment of the state of technology on molten fuel/steel-concrete interactions.

If a coolable debris bed does not form, experiments at the Aerospace Corporation,[2] showed penetration into concrete by molten UO_2 at the rates of 5.5 in/hr. for basalt concrete, 2.4 in./hr for magnetite concrete and 1.0 in/hr. for limestone concrete. In these experiments, the applied integrated power was 12.5, 13.4 and 6.25 KWHR, respectively. These penetration rates are subject to some qualifications in that the experimenters could not say what fraction of the applied power was coupled into the material. This depends on the coupling between the plasma and material and the conductivity of the liquid. If the coupling is similar for each concrete, since the integrated powers were similar, basalt would seem to be at least resistant to penetration by UO_2. Also, the area heated was only about 1.5 in. in diameter. In order to heat the entire quantity of UO_2, the operator moved the stinger electrode over the entire UO_2 pool. Consequently, the entire pool may not have been molten at all times.

In the Sandia molten core debris experiments,[3] the erosion rate varied from 25-135 cm/hr for basaltic and limestone concretes. Molten steel at 1700°C was used in one set of tests and caused erosion at the low end of the range. A molten steel-mixed oxide mixture (not including UO_2) at 2800°C caused erosion at the high end of the range. In the latter series of experiments, the density of the mixture was much less than for a UO_2 -steel mixture. If the density had been greater, the penetration would also have been greater. Sandia felt that the steel constituent, rather than the oxide, was responsible for most of the

erosion, due to its greater conductivity. Erosion was thought to be caused primarily by melting of the concrete rather than by spalling or chemical attack and seemed to be independent of the type of concrete.

The staff also believes that melting of the concrete constituents will be the dominant cause of penetration. The high density of the core debris means that there can be no protective layer. Fresh concrete is continually exposed to the melt. Spallation will be relatively unimportant because the core debris is at a temperature far above the melting point of concrete. This situation is very different from the attack of sodium on concrete because the sodium temperature is far below the concrete melting temperature. Metallic constituents of the core debris (stainless steel) will probably be able to transfer more energy to the concrete than UO_2, which is a good insulator, at least in its solid state.

If the scenario with molten core debris in contact with concrete is credible, as we believe it is, the penetration rate and extent will be higher than with only sodium in contact with concrete. Sandia's high temperature - high heat flux penetration rate is in excess of 4 ft./hr. However, this is considered to be an upper bound. Taking into account the heat flux from the core debris that would exist for the FFTF, we believe that a reasonable estimate based on the Sandia and Aerospace Corporation data is that the steady state penetration of a molten fuel/steel pool into the reactor cavity concrete will be in the range of 1 to 3 in/hr.

For the last three years, NRC's Division of Reactor Safety Research has been supporting a confirmatory research program at Sandia Laboratories on the materials interactions between molten fuel/steel and concrete. Accordingly, we requested the Sandia researchers to provide us with their current state of understanding of high-temperature melt/concrete interactions. Sandia's response is provided in Attachment 1 to this appendix. Our recommendation as provided above on the penetration rate of a molten fuel/steel pool into the FFTF reactory cavity concrete reflects consideration of the Sandia assessment.

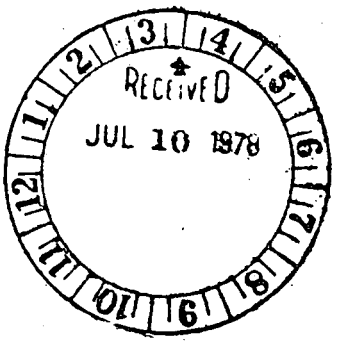
References

1. Baker, L., Jr., et al., "Post-Accident Heat Removal Technology," ANL/RAS 77-2, dated January 1, 1977.

2. Swanson, D. G., et al., "Annual Progress Report Evaluation of Materials for CRBRP Core Retention," The Aerospace Corporation, NUREG/CR-0076, May 1978.

3. Powers, D. A., et. al., "Exploratory Study of Molten Core Material/Concrete Interactions July 1975 - March 1977," Sandia Laboratories, SAND 77-2042, February 1978.

ATTACHMENT 1



Sandia Laboratories
Albuquerque, New Mexico 87115

July 6, 1978

State-of-the-Art
Assessment of Melt-Basalt Concrete Interactions

by

Dana A. Powers

Sandia Laboratories

July 6, 1978

I. MELT PENETRATION

A. Melt erosion of the concrete is due to melting. Spallation of the concrete either by thermally induced stresses or pressurization is an insignificant phenomenon accounting for penetration of no more than 0.5 cm. This is true for a wide variety of concrete including that made with basaltic aggregate.

B. The rate of melting attack is a function of the difference between the solidus temperature of the concrete and the bulk temperature of the melt. Concretes having relatively low solidus temperatures ($\sim 1100^{\circ}C$), such basaltic concrete and limestone concrete made with natural sand, erode much more rapidly than do concretes, such as the CRBR limestone concrete, that contain little silica (< 1 w/o) and have high solidus temperatures ($\sim 1480^{\circ}C$).

The functional dependence of concrete erosion rate on the melting temperature is complicated when the bulk temperature of the melt is less than the liquidus temperature of the concrete. Melting of the concrete is not complete in this case, and residual, solid, refractory material impedes heat transfer from the melt to the concrete. This complexity in the dependence of erosion on temperature is not important for basaltic concrete and silica-rich limestone concrete. The melting temperature ranges of these concretes ($T_{solidus} \approx 1100^{\circ}C$; $T_{liquidus} < 1400^{\circ}C$) are entirely below the solidification temperatures of most phases present in a core melt. The complexity is important for refractory concretes whose melting temperature intervals include the solidification temperatures of the melt phases.

C. The extent of melt penetration is a function of difference between heat generated within the melt and heat losses from the melt. Heat losses from the melt include the enthalpy necessary to raise the temperature of the adjacent concrete. Endothermic reactions of concrete decomposition are major contributors to this enthalpy only in the case of limestone concrete. The enthalpy to raise basaltic concrete to its liquidus temperature is 25 - 50% less than the corresponding value for limestone concrete. Consequently, a melt of given heat content will erode more basaltic than limestone concrete.

Erosion of concrete is also limited by other heat losses from the melt. Extensive gas evoltuion produced in melt attack on limestone concrete contributes to limiting the heat transferred into the concrete to about 25% of the heat available from the melt. In the case of basaltic concrete about 35% of the available heat passes into the concrete.

D. Directional nature of heat flux from the melt to the concrete is influenced by gas generation rates. The ratio of horizontal to vertical heat fluxes is a function of the inverse of the rate of gas generation rate. The ratio in the case of CRBR limestone concrete, which has a high gas generation rate, is no greater than 1 to 7. The ratio for attack on basaltic concrete is about 1 to 3. One consequence of this directional nature of heat flux

is that once a cavern is formed by the melt in the concrete, it tends to grow preferentially downward. The magnitude of this preference, which is dependent on the gas generation rate, influences directly the amount of concrete that must be eroded to produce a given extent of concrete penetration.

E. Metallic melts attack concrete more vigorously than oxidic melts. The ability of a melt to impart heat to concrete - the effective melt thermal conductivity - is the most important melt property for estimating melt attack on concrete. This property completely overshadows the effects of concrete wetting by the melt and the solubility of concrete in the melt. Steel therefore attacks concrete at least 10 times more vigorously than oxidic melts with similar solidification temperatures.

F. The mechanism of heat transfer from the melt to concrete is poorly understood. At least three models have been proposed for this heat transfer:

1) Simple conduction
2) Stagnation flow about isolated gas bubbles.
3) Gas film impedance.

Current speculation suggests that the heat transfer mechanism is temperature dependent. Simple conduction is applicable after solidification, stagnation flow develops at modest temperatures, and gas films arise at very high temperatures. Unfortunately, all of the above models yield predicted heat transfer rates well in excess of those

experimentally observed. The most successful models for predicting melt/concrete interactions rely on empirical heat transfer rates derived from the experiments.

Some of the limitations in the predictive ability of mechanistic heat transfer models may be the result of poor understanding of melt/concrete properties such as surface tension.

II. GAS GENERATION

A. Gases generated during the melt/concrete interaction are the result of thermal decomposition of hydrates and carbonates within the concrete. The dependence of gas generation on concrete erosion arises only insofar as temperature profiles within the concrete are dependent on concrete erosion. Consequently, gases liberated from the concrete are not closely related to the concrete stoichiometry. Water, which is more easily liberated than carbon dioxide, is the predominant gas species even during melt interactions with limestone. Further, gas release from basaltic concrete is quite similar to gas release from limestone concrete even though limestone contains 50% more gas-generating species per unit volume.

The dependence of gas generation on heat transfer within the concrete is well-understood. Models predicting gas generation rates, heat transfer within the concrete, as well as moisture migration within the concrete are in an advanced state of development.

C. Gases liberated from the concrete are chemically reduced to hydrogen and carbon monoxide as they pass through the metallic portions of the melt. Subsequent equilibration of the gases at cooler temperatures yields formation of hydrocarbons. Thermodynamics provides an adequate formalism for understanding the gas and gas-melt equilibria. Data to date show that within the melt the oxygen fugacity of the gas is buffered by the ferrous oxide-iron equilibrium. This buffering and the known elemental composition of the gas allows development of models to predict gas compositions throughout the melt/concrete interaction.

D. Oxidation of metallic portions of a melt is not an unlimited process. In a closed system the oxygen fugacity of the atmosphere above the melt may become sufficiently low that metal oxides are reduced and reincorporated in the metallic melt. The reducibility of the metal oxides may be ordered as follows:

$$NiO > FeO >> Cr_2O_3 > ZrO_2 >> UO_2$$

Consequently, the aggressive metallic portions of the melt may not be completely depleted during the interaction. The extent of concrete penetration may be therefore greater than heretofor expected. The equilibrium metal content of the melt may be greater during melt/basaltic concrete interactions than during melt/limestone concrete interactions.

III. FISSION PRODUCT REDISTRIBUTION

A. Data are being collected on the redistribution of fission product mock materials among the oxidic and metallic portions of the core melt. Preliminary results suggest that the sparging of oxidizing gases through the melt results in depletion of fission products from the metallic phases and concentration of fission products in the oxidic phases. Preliminary data also suggest that non-idealities of the oxide liquid result in depressed vapor pressure for some fission products. Cesium and molybdenum oxides appear to be tightly bound into the oxide melt and not especially volatile even at temperatures near 1600°C.

B. Aerosol generation is extensive and appears to be dependent on both melt temperature and gas generation rates. Aerosol concentrations of up to 130 g/m^3 have been observed for melts at $\sim$ 2800°C. At 1500°C aerosol concentrations amount to about 20-30 g/m^3. Tight coupling of melt temperature and gas generation rates has been an obstacle to quantifying aerosol generation rates and their dependences. Aerosol generation remains at a relatively primitive level of understanding.

C. Aerosols arise from non-fuel sources as well as from fission products and fuel. Oxides of steel and constituents of concrete may constitute as much as 35 w/o of aerosols generated during the melt/concrete interaction. Non-fuel aerosols are especially significant during interacions with basaltic concrete. Basaltic concrete is rich in

alkali metal oxides and silica which may be sparged into the atmosphere above the melt.

Significant quantities of non-fuel aerosols produced during the melt/concrete interaction would be expected to accelerate agglomeration of the particulate matter As a consequence, sedimentation and impaction of the aerosols above the melt would be expected to accelerate.

IV. COMPUTER MODELS

A. INTER - the first preliminary attempt to model melt/concrete interactions. The model is based on the first large-scale melt/concrete tests. It is highly empirical and does not contain all the phenomena discovered in subsequent tests. INTER is not considered a predictive model. It is now used primarily for sensitivity studies.

B. WECHSEL - a continuation of INTER. Contains additional phenomena but not quantitative data from recent tests. Most development to date has been devoted to heat transfer mechanisms.

C. CORCON - An entirely now code which will contain all phenomena and quantitative data observed to date. CORCON will be mechanistic wherever possible. Where mechanisms are not available options will be included to permit use of empirical correlations or proposed but unverified models. CORCON, it is hoped, will be a predictive code.

D. GROWS - UCLA is modifying the GROWS code, which was developed for melt/core catcher interactions, to handle melt/concrete interactions. To date, the code is more primitive than INTER. It is being used to study heat-transfer mechanisms. Limitations of the GROWS formalism suggest that any major development of this code is unlikely.

APPENDIX E

PIPING INTEGRITY

This appendix is part of the regulatory staff safety evaluation of the FFTF. It addresses Appendix B of the FFTF Final Safety Analysis Report. Piping integrity in the FFTF has been a concern because a reactor coolant piping failure could conceivably be a potential accident initiator. Piping integrity is a complex issue because the consequences of a leak vary depending upon size, location, temperature, features provided to mitigate the consequences of the event, detection methods, time of response for protection system action, and most importantly the leak initiating mechanisms and the inherent material and loading characteristics of the reactor coolant system to resist catastrophic failure. On June 11, 1976, by letter to the FFTF Project, we concluded, based upon our review of Appendix B to the Final Safety Analysis Report, that in our judgment a double-ended instantaneous pipe rupture in the cold leg of the main primary heat transport system has a sufficiently low likelihood of occurrence in the FFTF, and that it should not be considered an initiating event that could lead to core melt and disruption. Our judgment is based upon the inherently high fracture toughness of Type 304 stainless steel and the belief that the initial material properties can be retained and assured by adequate material test and surveillance programs.

Our position was conditional and was predicated upon having sensitive, reliable leak detection capability to detect small leaks early and meaningful inservice inspection to detect cracking or material degradation. No position was taken for hot leg piping by the staff, since in the FFTF it appeared that the reactor could withstand hot leg pipe leaks, including double-ended, with protective system action initiated by the first or second level trip signal without leading to core disruption. The FFTF primary heat transport system steel-lined cells and reactor cavity also appeared capable of withstanding sodium sprays and substantial sodium spills (without molten fuel release) without exceeding the capability of failing the structure for one design cycle, using a 5 percent strain criterion for the design of the steel liners.

Neither the staff nor the FFTF Project established a specific design basis sodium leak. Instead parametric studies were performed to indicate the sensitivity of cell pressures from a spectrum of sodium sprays to a pool type spill. In all cases run by the Project and the staff's consultants the cell liners and cavity liner were assumed to function in the sense that they would remain effective in preventing large quantities of sodium from contacting large areas of concrete. Without the cell liners the sodium concrete reaction from a major spill of hot

sodium could have the potential to challenge the containment system. We have concluded that any large leak, which in time from minutes to hours could drain the upper sodium pool from the reactor, could result in unacceptable consequences to containment structures if the cell liners could not be depended upon to limit the hot sodium contact area with the concrete. Without relying upon cell liners to protect concrete surface areas from major sodium spills the result could be a challenge to containment from overpressurization due to the sodium concrete reaction. Secondly, in the event of a substantial pipe leak, one cannot be assured that the reaction product gases and secondary chemical reaction products formed could not make their way back into the reactor coolant system and disrupt the convection heat removal process in adjacent main heat transfer loops to prevent decay heat removal through these loops. For the preceding reasons we have concluded that major cell liners associated with the reactor cavity, pipeways, overflow tank, and intermediate heat exchanger cells perform a very important safety function.

Since the important safety function of the steel liners is to minimize the contact of hot liquid sodium with concrete, we request a verification test which should place emphasis on liner areas subject to potential failure or degradation that could provide sodium access to large concrete surfaces. It is recognized that the reactor cavity will not be accessible following startup. Where accessibility exists in heat transfer system cells dye penetrant or magnetic particle examination may be appropriate. This information can aid in assessing the probable behavior of the reactor cavity liners. The Project is requested to address the problem and initiate an appropriate periodic verification program for cell liners, possibly coordinated with inservice inspection of piping in the main heat transfer system.

With respect to piping integrity in the cold leg downcomer, the only pipes which have the potential for being core disruptive accident initiators with protective system action are the cold leg reactor inlet pipes up to the check valves. The piping integrity argument advocated by the Project is built around a leak before break hypothesis which follows:

(1) Quality assurance prevents defects above a certain size from being built into the piping system;

(2) If a small defect escapes or passes through the quality assurance program, it should not be larger than an acceptable size;

(3) In the existing stress fields of the cold leg piping system, even at points of stress concentration, crack development will be preferentially through the wall rather than along the surface; and

(4) Crack growth rates longitudinally, or circumferentially, will not occur at a high enough rate that critical crack size will be reached (16 inches to

20 inches) for stainless steel piping (3/8-inch wall) for the sizes in the FFTF and that ample time will be available for;

(5) Leak detection systems, of sufficient sensitivity, to respond well in advance to permit an orderly plant shutdown before a self-propagating piping leak develops; therefore

(6) Piping integrity for the cold leg will be maintained sufficiently long that an instantaneous cold leg pipe rupture need not be considered a design basis event leading to core melt; and

(7) Material surveillance and inservice examinations will be initiated to provide further confidence for piping system integrity to assure the continued reliability of the system over the 20-year plant life.

We conditionally agree with the leak before break position proposed by the FFTF Project. The conditions necessary to reach convergence from our point of view are implementation of:

(1) Performance of representative preservice and inservice inspection in considering the status of construction to establish a baseline inspection.

(2) Verification of the installed in plant aerosol leak detection to demonstrate sensitivity and reliablity.

(3) Initiation of an adequate material surveillance program in a prototypic environment to provide advance warning of unanticipated material degradation processes.

(4) That the above mentioned major cell liners be considered as important safety features which require a periodic verification of function (prevention of gross sodium concrete contact) to accommodate an undefined sodium spill up to and including a protected (with scram) double-ended hot leg pipe rupture.

The following discussion is directed to supporting the conclusions arrived at regarding cold leg piping integrity. For this appendix it is sufficient to state that cell liners are installed and can be expected to separate gross amounts of sodium from concrete and are considered as maintaining this function.

There are ~50,000 pounds of hot sodium in the upper reactor pool that are above the elevated piping in the heat transfer system cells which could ultimately drain by gravity out of any but the smallest size piping leak unless the loop isolation valves were closed. If the hot leg leak was hypothesized to be large (double-ended) and happened quickly, this quantity of sodium could be released very rapidly due to pump action and gravity flow or, if in the 28-inch hot leg, by gravity flow only

This quantity of sodium in contact with bare concrete would create a serious safety concern from the gas pressure and hydrogen generated from the sodium-concrete reaction and would in time be a potential challenge to the containment system structures. A larger amount of sodium could also be released into the reactor cavity by a break beyond the hot leg guard pipe as it passes through the reactor cavity. Neither of these situations is considered acceptable.

The consequences of a leak in the cold leg reactor coolant piping beyond the check valve obviously vary depending upon the leak location and size of the leak. The safety concern is the category of leaks and leak locations that can potentially cause boiling in the core through flow starvation to the core by flow passing through the leak even though scram and pump trip have been initiated by the protective system.

An analysis of various sized leaks at various locations , including double-ended pipe rupture, was performed by the Project and by the staff's consultants. These studies included plant protective system action, flow impedance, trip time and pump coastdown time, power generation decay, stored energy, and decay power. Since the criterion for acceptability was not to lose coolable core geometry, it was considered satisfied if boiling did not occur in the hot channel in the core. These studies conclude that the transient temperatures for the hot channel do not reach boiling but that, due to the uncertainties in hot channel factors, the margins against boiling may not be as large as previously thought. Hot leg pipe breaks or breaks in the cold leg between the check valve and the intermediate heat exchanger do not appear to initiate boiling in the core for a protected accident.

Realistically, instantaneous pipe ruptures of large magnitude are considered highly unlikely events and the leak before break rationale is reasonable in our view. Experience with many piping systems has shown that leaks may be expected but experience with high temperature piping systems utilizing new design methods is much more limited. It is still considered necessary that one have adequate leak detection of sufficient sensitivity and meaningful material surveillance to assure continued piping integrity over the life of the plant. We recommend that the Project fully consider the recommendations in Section 19.0 of this report.

APPENDIX F

STAFF ANALYSIS OF FFTF CONTAINMENT
MARGINS EX-VESSEL SCENARIO WITH COOLABLE DEBRIS BED ASSUMPTION

F.1 INTRODUCTION

The NRC Staff performed a series of parametric analyses as part of its technical assessment of the response of the FFTF containment to a postulated core meltdown event. The purpose of the analyses reported in this appendix was to obtain estimates of the time at which the integrity of the containment would be challenged by the formation of either excessive explosive hydrogen concentrations or excessive containment pressures during the transient resulting from a postulated melt-through of the reactor vessel by core debris. The results of these analyses were then used to calculate the radiological release at the site boundary due to the postulated accident.

F.2 ASSUMPTIONS

In these analyses, the occurrence of a core disruptive accident (CDA) was postulated. This CDA either consisted of a 350 MW-sec transient overpower or a 100-110 $/sec loss of flow event. The postulated CDA caused the following releases through the reactor head seals into the atmosphere of the head compartment-reactor containment building:

(1) 1000 pounds of sodium,
(2) 100% of the noble gas fission products from the core,
(3) 25% of the total iodine inventory (75% remained in the reactor vessel),
(4) 1% of the volatile solids, and
(5) 1% of the fuel.

In addition, it was assumed that 133,000 pounds of sodium were ejected into the reactor cavity through the omega seal.

From the reactor containment building (RCB), the above releases could escape to the outside. The RCB leak rate was assumed to be 0.1 vol. % per day at pressures less than or equal to the design pressure of 10 psig. For pressures greater than 10 psig, the RCB leak rate was assumed to be directly proportional to $\sqrt{\Delta P}$, where ΔP is the pressure differential between the RCB atmosphere and the outside environment.

It was postulated that, at three hours after the CDA, the core debris melted through the reactor and guard vessels. All of the core debris material was released into the reactor cavity. Core debris in particulate form distributed itself over the steel liner protecting the floor of the reactor cavity. Sodium from the reactor coolant

system (including the remaining inventory from the reactor vessel and drainable piping) was released into the cavity at 1142°F assuming that: (1) no heat had been removed by the heat transport system, and (2) the fission product decay power remaining in the vessel after the postulated CDA went into raising the temperature of the sodium until vessel penetration occurred. Except for the initial releases noted above, the distribution of fission product decay power was treated in the same manner as in the FFTF Containment Margins Report.[1] A self-leveling, coolable debris bed was assumed to form on the cavity floor liner covered by 619,000 pounds of sodium (the primary system inventory less the initial amount released to the RCB). The circular portion of the cavity floor liner directly under the guard vessel (346 ft^2) was assumed to fail upon contact with sodium, leading to a sodium-concrete reaction. This sodium-concrete reaction continued until it completely penetrated the 3-foot thick floor of the reactor cavity, spilling the sodium into the unlined subcavity below. The subsequent sodium-concrete reaction in the subcavity was assumed to last until the sodium had boiled off completely. The sodium-concrete heat of reaction and data for the water release from heating concrete were taken from the Containment Margins Report.[1] It was assumed that the cavity leaked continuously into the RCB through a one-foot diameter hole (hole size arbitrarily chosen to ensure that the atmospheres of these two cells would have approximately the same pressure). The recombination reaction between hydrogen and oxygen was also considered. The steel liners protecting the portion of the cavity walls above the sodium pool were assumed to be vented to the heating and ventilation cooler room.

F.3 ANALYTICAL MODEL

These analyses were performed using the CACECO computer code.[2] The version of CACECO used was the version of the code transmitted to the Argonne Code Center (and described in Ref. 2) with several modifications which the staff obtained from the author of the CACECO code, R. D. Peak.[3] These modifications consisted of:

(1) consideration of the activity of the isotope Na-24,
(2) elimination of the hydrogen-oxygen recombination reaction, and
(3) restriction of nozzle flow to one direction only.

The CACECO model of the FFTF used for this analysis was essentially the same as that used for the ex-vessel cases described in the Containment Margins Report.[1]

F.4 DISCUSSION OF ANALYSES

The analyses described in this appendix were performed specifically to evaluate the effect of various sodium-concrete reaction rates on the predicted transient response of the FFTF containment to a postulated core melt-through event. In addition, the effect of directing into the sodium pool the water that was released from behind the intact liners protecting the submerged outer floor and walls of the reactor cavity was assessed. This assumption arose from a staff concern regarding possible blockage of the vent path for water released from behind these intact liners.

An ex-vessel scenario (same terminology used in Ref. 1) which used the assumptions identified in Section F.2 was chosen as the base case for these analyses (hereinafter referred to as Case 1). Case 1 also assumed that the sodium-concrete reaction rate was 3 inches/hr and that the water released from behind the submerged liners in the cavity was directed into the sodium pool. In Case 1, no venting or purging of the hydrogen accumulating in the RCB was allowed. Case 2 was identical to Case 1, except that venting and/purging of the RCB were performed when the hydrogen concentration in the RCB reached 4%. In Case 3, hydrogen recombination was considered along with the assumptions of Case 2. In Cases 4 through 8, the sodium-concrete reaction rate was varied from 2 inches/hour to 0.5 inch/hour in 0.5 inch/hour increments. Otherwise these cases were identical to Case 2. Cases 9 through 13 were identical to Cases 2 and 4 through 8, except that the water released from behind the submerged intact liners in the reactor cavity was vented to the heating and ventilation cooler room. Case 14 was identical to Case 2 except that it used a variable sodium-concrete reaction rate proposed by the Project. This rate was given by the expression:

$$X = 12(1 - e^{-0.5t})$$

where X = penetration in inches

t = time in hours

This reaction was assumed to terminate after 12 hours.

Case 15 was identical to Case 9, except that the proposed variable sodium-concrete reaction rate was used. Case 16 duplicated Case 716[4] as run by R. D. Peak at HEDL, except that the water released from behind the submerged cavity liners was directed into the sodium pool in the cavity. Case 17 was identical to HEDL's Case 716. Table F.1 summarizes the CACECO runs performed in the subject analyses.

F.5 RESULTS

Figures F-1 through F-3 show the CACECO predictions of RCB pressure, RCB hydrogen concentration, and mass of the sodium pool in the reactor cavity for the various sodium-concrete reaction rates for those cases where the water released from behind the submerged intact cavity liners was directed into the sodium pool. From Figure F-1, it can be seen that, from the time when the core debris melted through the vessels until the time that the sodium-concrete reaction completely penetrated the cavity floor, the RCB pressurization rate was rather slow. However, when the cavity floor failed and the sodium spilled onto the bare concrete subcavity walls and floor, there was a marked increase in the pressurization rate. This was due to the increased surface area of bare concrete exposed to attack by sodium, which resulted in a more severe transient. For the cases where the cavity floor was completely penetrated by the sodium-concrete reaction, the time that the RCB pressure reached its design value of 10 psig ranged from 16 hours for the 3.0 inch/hr case to about 76 hours for the 0.5 inch/hour reaction rate case. In each of these cases, the RCB pressure reached the design value within an hour after the time of cavity floor failure. It should also be

TABLE F.1

SUMMARY OF CACECO FFTF RUNS

CASE NO.	DEFINITION
1	EX-VESSEL SCENARIO USING ASSUMPTIONS SPECIFIED IN STAFF'S 2/13/78 LETTER. WATER FROM SUBMERGED CAVITY WALL DIRECTED INTO CAVITY POOL.
2	SAME AS CASE 1 EXCEPT NO RCB VENTING AND H_2 PURGING ALLOWED.
3	SAME AS CASE 2, EXCEPT H_2 RECOMBINATION CONSIDERED.
4	SAME AS CASE 2, EXCEPT THAT THE CAVITY FLOOR LINER REMAINS INTACT.
5	SAME AS CASE 2, EXCEPT THAT THE Na^- CONCRETE REACTION RATE = 2.0 INCHES/HR.
6	SAME AS CASE 2, EXCEPT THAT THE Na^- CONCRETE REACTION RATE = 1.5 INCHES/HR.
7	SAME AS CASE 2, EXCEPT THAT THE Na^- CONCRETE REACTION RATE = 1.0 INCHES/HR.
8	SAME AS CASE 2, EXCEPT THAT THE Na^- CONCRETE REACTION RATE = 0.5 INCHES/HR.
9	SAME AS CASE 2, EXCEPT THAT WATER FROM CAVITY SUBMERGED WALL VENTED TO H & V COOLER ROOM.
10	SAME AS CASE 9, EXCEPT THAT Na^- CONCRETE REACTION RATE = 2.0 INCHES/HR.
11	SAME AS CASE 9, EXCEPT THAT Na^- CONCRETE REACTION RATE = 1.5 INCHES/HR.
12	SAME AS CASE 9, EXCEPT THAT Na^- CONCRETE REACTION RATE = 1.0 INCHES/HR.
13	SAME AS CASE 9, EXCEPT THAT Na^- CONCRETE REACTION RATE = 0.5 INCHES/HR.
14	SAME AS CASE 2, EXCEPT THAT A VARIABLE Na^- CONCRETE REACTION RATE WAS USED.
15	SAME AS CASE 9, EXCEPT THAT A VARIABLE Na^- CONCRETE REACTION RATE WAS USED.
16	HEDL CASE 716 AS RUN BY R. D. PEAK.
17	SAME AS CASE 16, EXCEPT THAT WATER FROM SUBMERGED CAVITY WALL DIRECTED INTO SODIUM POOL IN CAVITY.

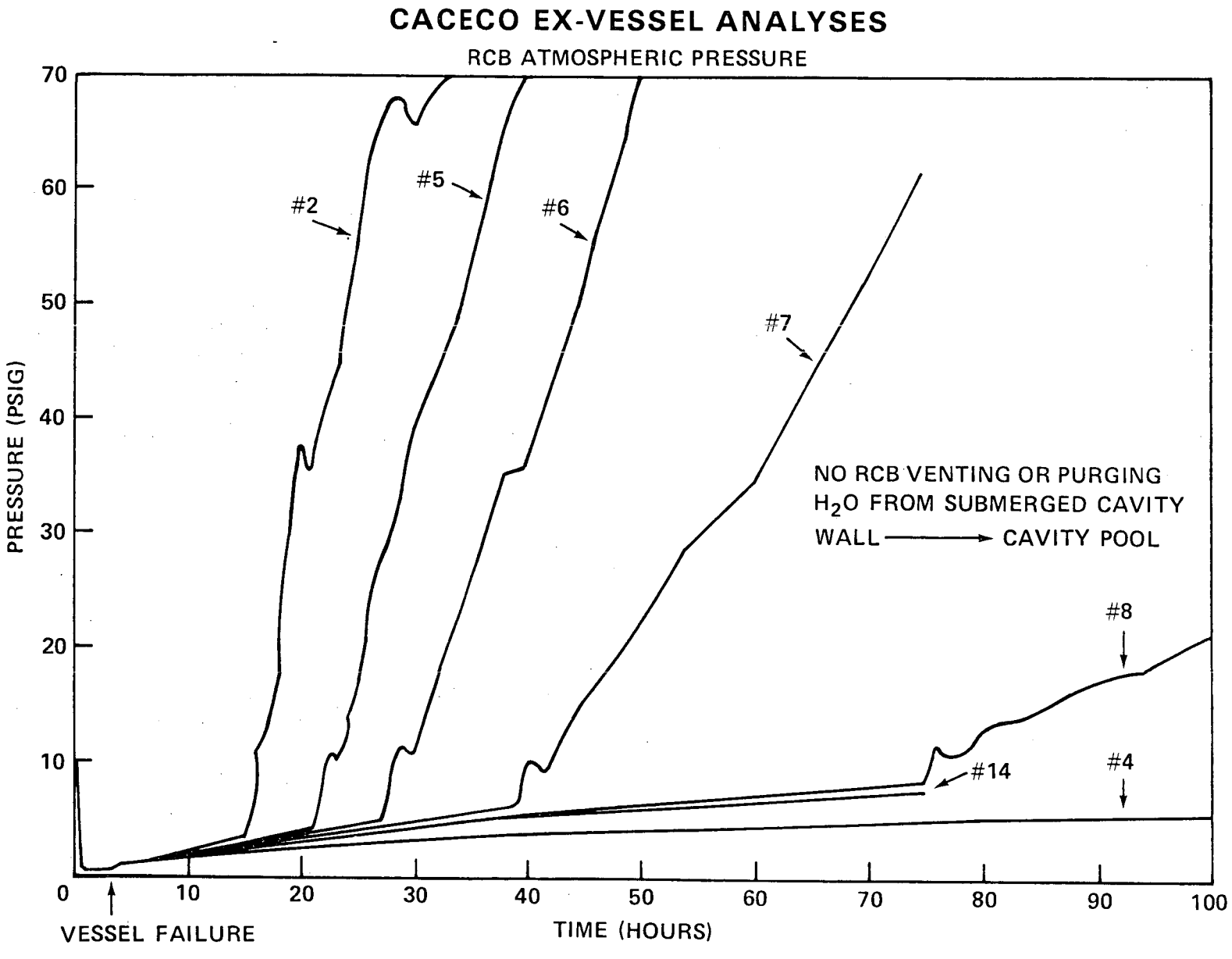

CACECO EX-VESSEL ANALYSES
RCB ATMOSPHERIC PRESSURE
PRESSURE (PSIG)
70
60
50
40
30
20
10
0
#2
#5
#6
#7
#8
#14
#4
NO RCB VENTING OR PURGING
H_2O FROM SUBMERGED CAVITY
WALL ⟶ CAVITY POOL
VESSEL FAILURE
10
20
30
40
50
60
70
80
90
100
TIME (HOURS)

Figure F-1

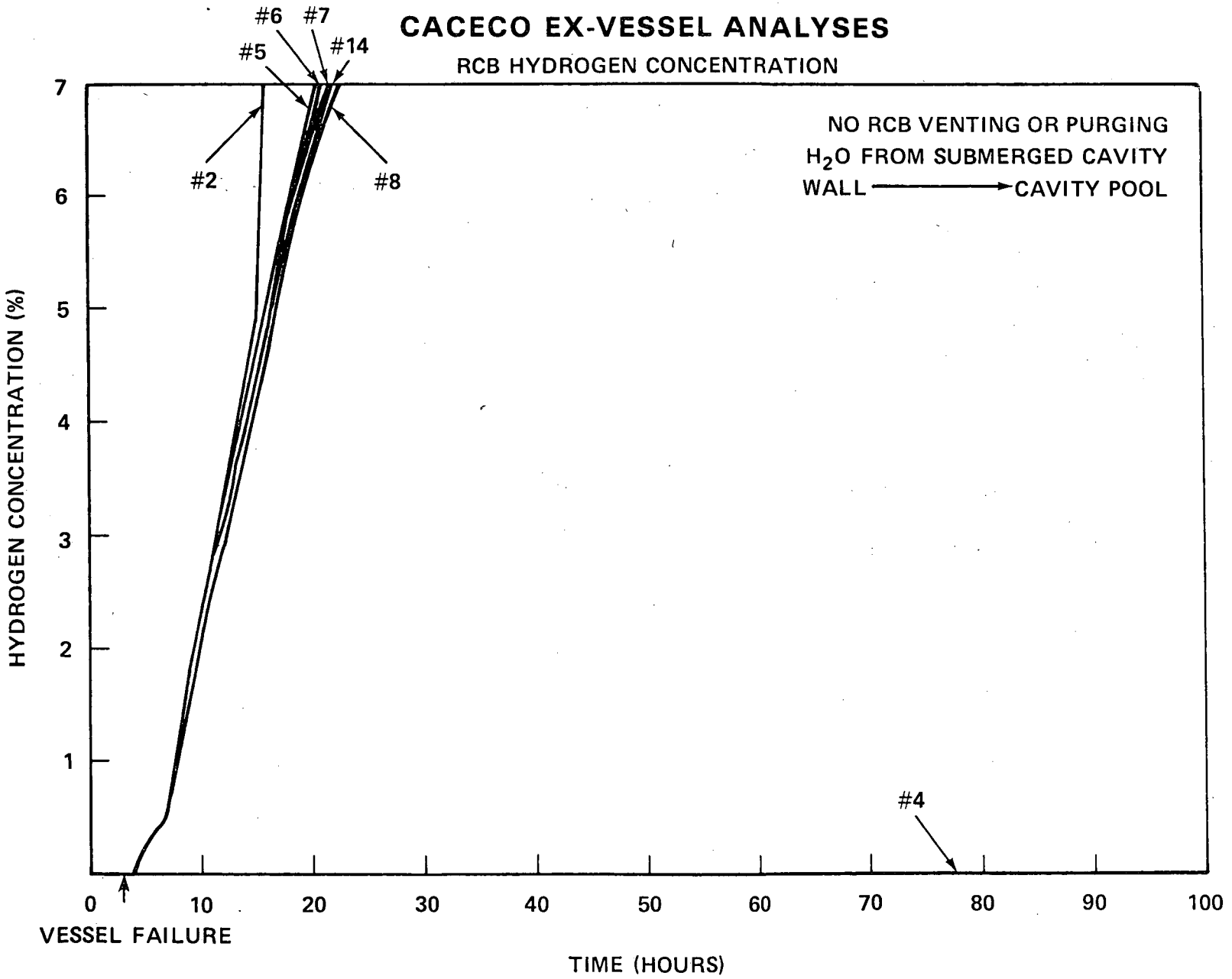

CACECO EX-VESSEL ANALYSES
RCB HYDROGEN CONCENTRATION
NO RCB VENTING OR PURGING
H_2O FROM SUBMERGED CAVITY
WALL → CAVITY POOL
#2
#4
#5
#6
#7
#8
#14
HYDROGEN CONCENTRATION (%)
TIME (HOURS)
VESSEL FAILURE
0
10
20
30
40
50
60
70
80
90
100
1
2
3
4
5
6
7

Figure F-2

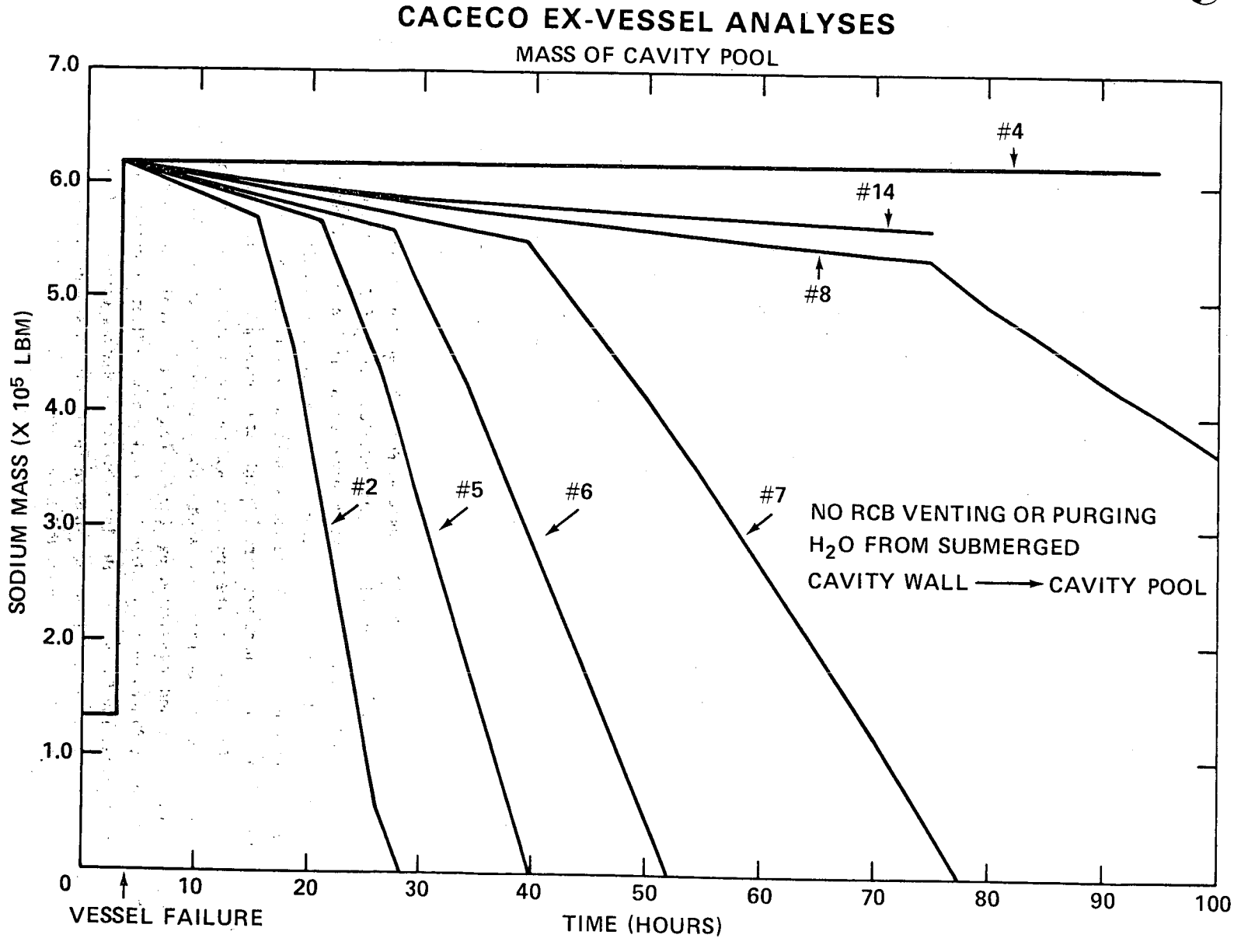

CACECO EX-VESSEL ANALYSES
MASS OF CAVITY POOL
SODIUM MASS (X 10⁵ LBM)
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0
#4
#14
#8
#2
#5
#6
#7
NO RCB VENTING OR PURGING
H2O FROM SUBMERGED
CAVITY WALL → CAVITY POOL
VESSEL FAILURE
10
20
30
40
50
60
70
80
90
100
TIME (HOURS)

Figure F-3

noted that the predicted pressures for Cases 8 and 14 were almost identical. This indicated that, for the first 75 hours of the analysis, the transient behavior for the scenario with the proposed HEDL variable sodium-concrete reaction rate was very similar to the case with a constant 0.5 inch/hour reaction rate. Of course, since the proposed HEDL reaction rate model specified that the reaction would be terminated after 12 inches of penetration, the cavity floor did not fail in Case 14. Therefore, one would not expect the results for Cases 8 and 14 to be similar after 75 hours. Since venting and purging of the RCB were not allowed for all these cases (except for Case 1), the predicted RCB pressures increased indefinitely. If venting of the RCB had been assumed, the pressure transient would have been terminated at the time venting occurred.

Figure F-2 shows that, in all cases where the submerged cavity liner vent system was assumed to fail, the hydrogen concentration in the RCB reached 4 vol. % at times between 13 and 16 hours, regardless of the sodium-concrete reaction rate. At 12 hours (corresponding to the cavity floor penetration time for Case 2), the hydrogen generation rate increased significantly because of the increased sodium-water reaction due to the large amounts of water released from the subcavity concrete walls and floor into the sodium pool. Although the cavity floor penetration times for the other cases have not been shown on this figure, these cases would also exhibit similar increases in hydrogen generation beginning at cavity floor failure time.

Figure F-3 shows the mass of the cavity sodium pool as a function of time. The knee observed in the curves for Cases 2, 5, 6, 7, and 8 corresponds to the time that initiation of rapid generation of sodium vapor was predicted by CACECO. This rapid boiloff began shortly after the sodium spilled into the subcavity. From this figure, it can be seen that the times when the sodium pool was predicted to boil dry ranged from 28 hours for the 3.0 inches/hour case to more than 100 hours for the constant 0.5 inch/hr case. Since the cavity floor was not penetrated for Case 14, no rapid boiloff was predicted for this case.

Figures F-4 through F-6 show the RCB pressure, RCB hydrogen concentration, and cavity pool mass, respectively, as functions of time predicted by CACECO for those cases where the submerged intact cavity liners were vented to the heating and ventilation cooler room. Examination of these figures revealed a number of significant differences from the corresponding figures for the cases where the water released from behind the submerged intact cavity liners was directed into the sodium pool. Comparison of Figures F-1 and F-4 shows that the predicted RCB pressures had the same general behavior in that the pressurization rates increased significantly after cavity floor penetration. However, this increased pressurization rate was not quite as rapid for the cases where the intact cavity liners were vented to the heating and ventilation cooler room. The pressure histories for Cases 9 and 10 also exhibited a maximum occurring 15-20 hours after cavity floor failure. This corresponded to the time that the sodium pool boiled dry. It can be seen that the predicted RCB pressures reached the 10 psig design value later for the cases where the intact cavity liners were vented to the heating and ventilation cooler room.

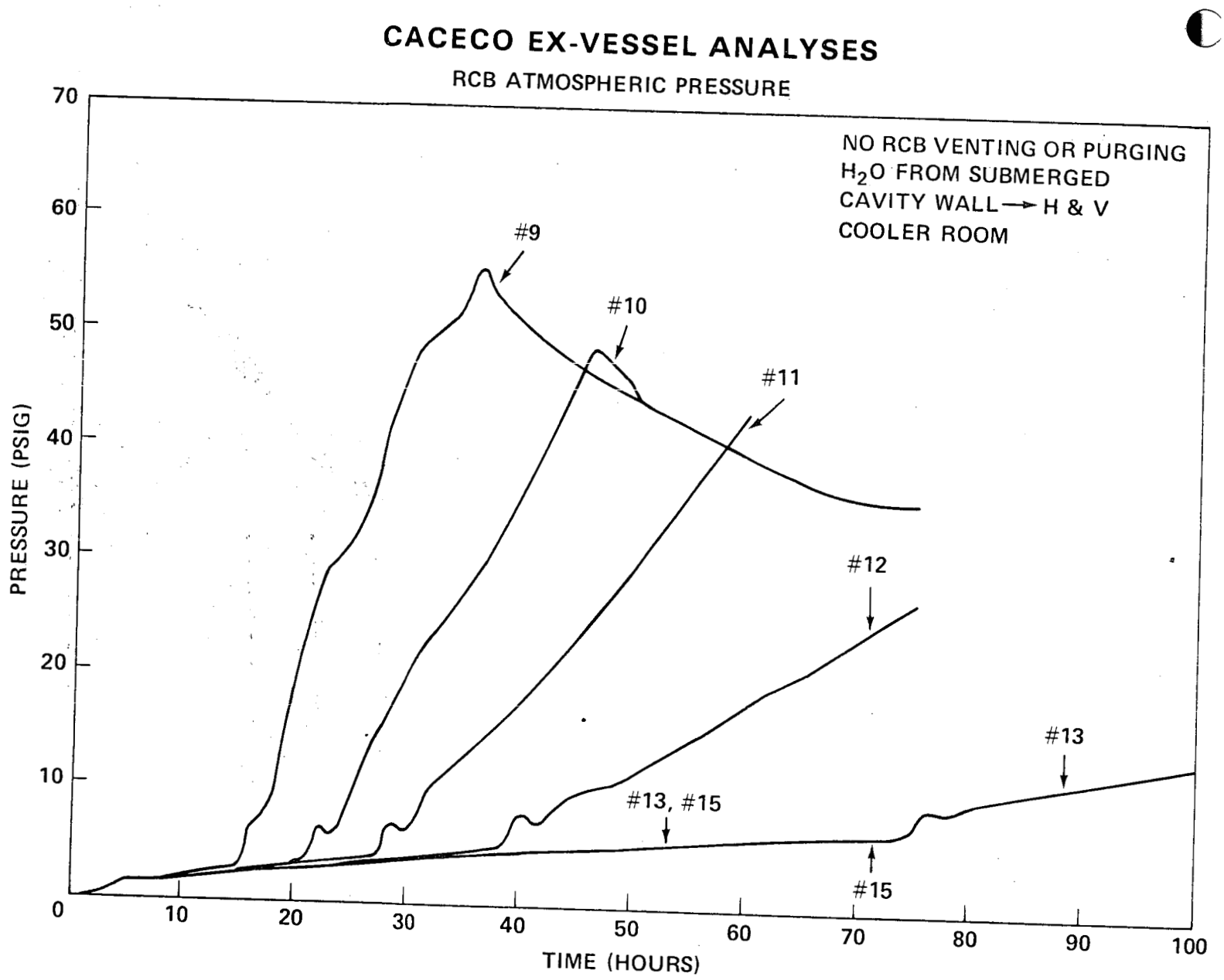

CACECO EX-VESSEL ANALYSES
RCB ATMOSPHERIC PRESSURE
NO RCB VENTING OR PURGING
H_2O FROM SUBMERGED
CAVITY WALL → H & V
COOLER ROOM
#9
#10
#11
#12
#13
#13, #15
#15
PRESSURE (PSIG)
TIME (HOURS)
0
10
20
30
40
50
60
70
80
90
100

Figure F-4

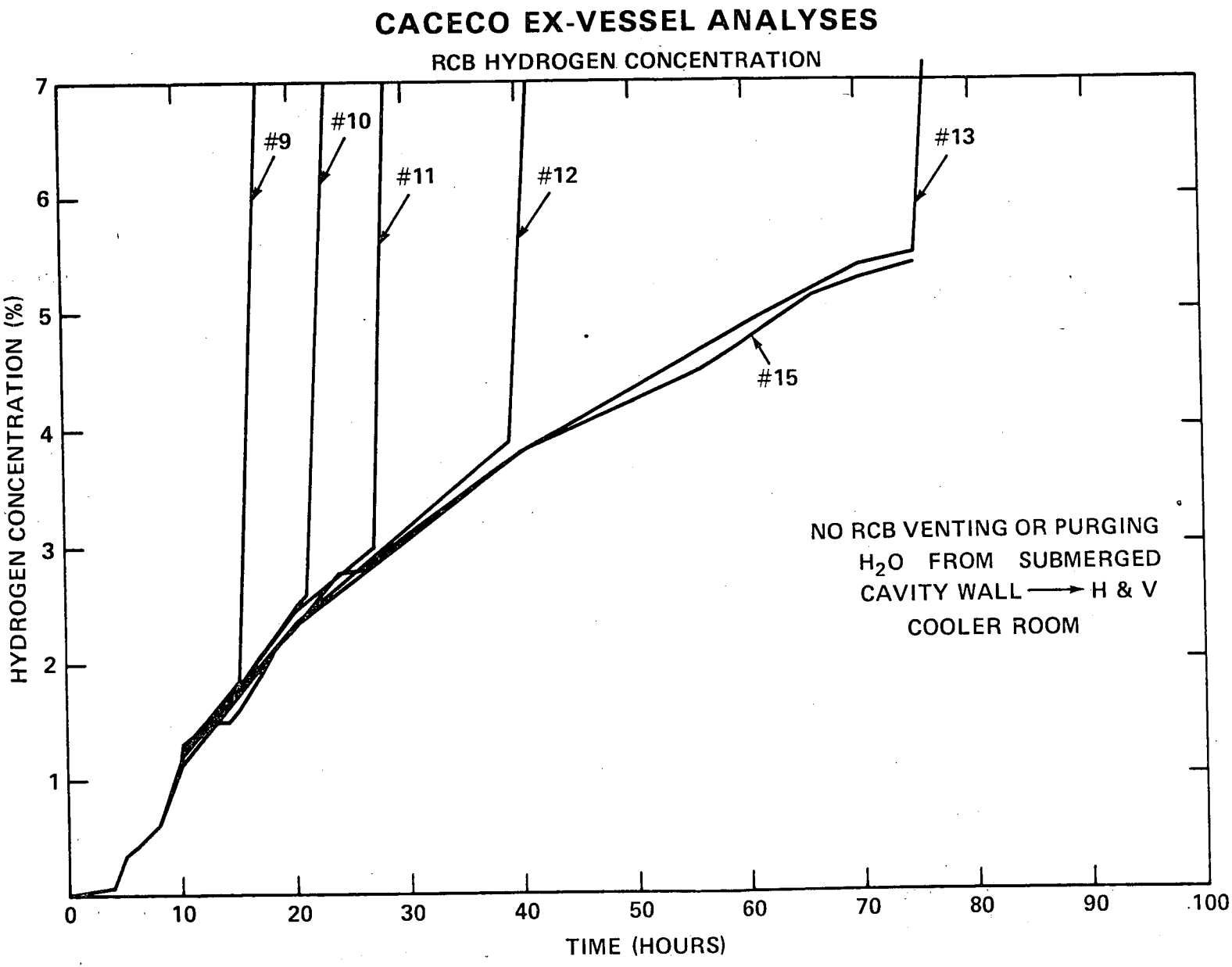

CACECO EX-VESSEL ANALYSES
RCB HYDROGEN CONCENTRATION
HYDROGEN CONCENTRATION (%)
TIME (HOURS)
#9
#10
#11
#12
#13
#15
NO RCB VENTING OR PURGING
H_2O FROM SUBMERGED
CAVITY WALL → H & V
COOLER ROOM
0
10
20
30
40
50
60
70
80
90
100
1
2
3
4
5
6
7

Figure F-5

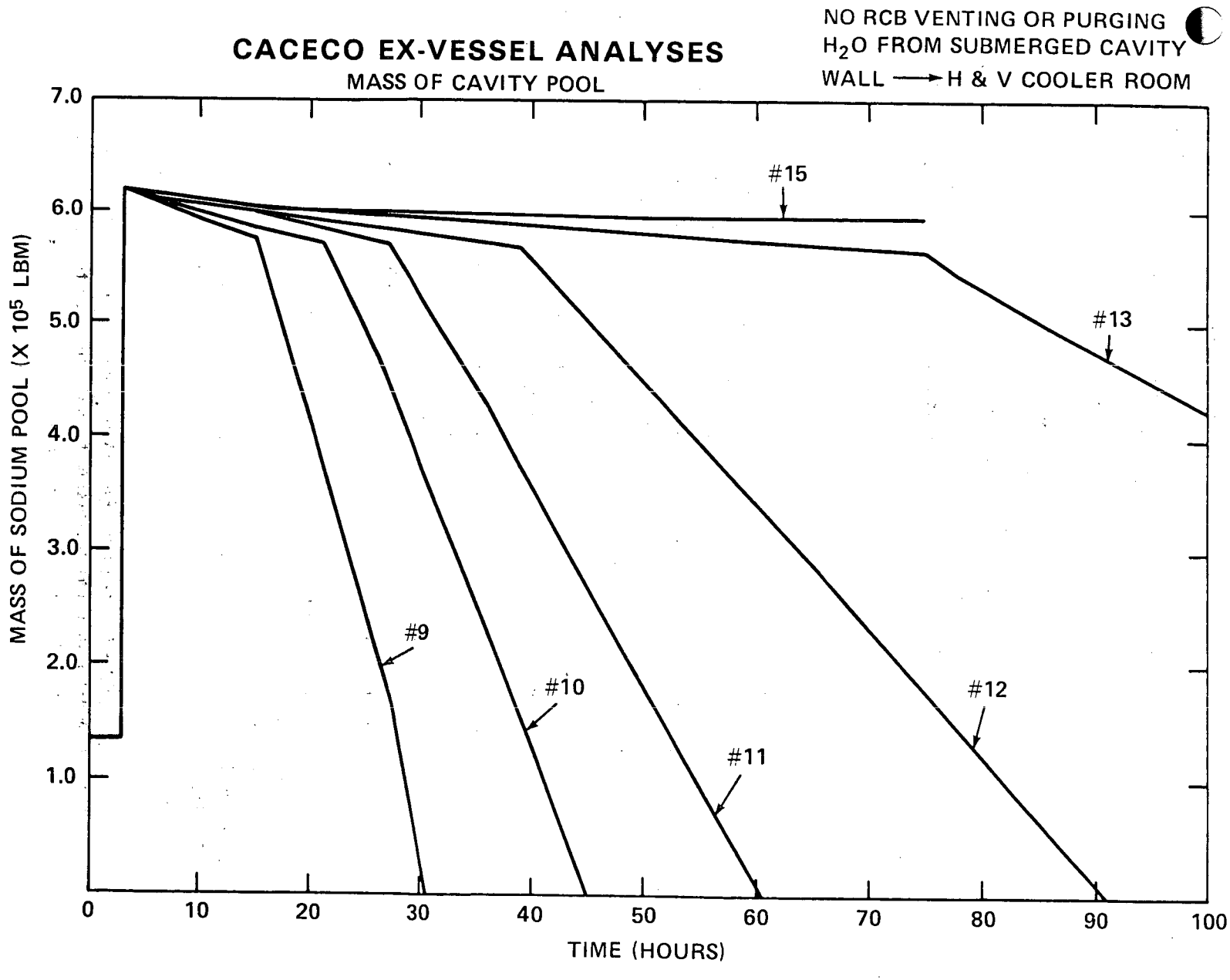

CACECO EX-VESSEL ANALYSES
MASS OF CAVITY POOL
NO RCB VENTING OR PURGING
H_2O FROM SUBMERGED CAVITY
WALL → H & V COOLER ROOM
MASS OF SODIUM POOL (X 10^5 LBM)
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0
10
20
30
40
50
60
70
80
90
100
TIME (HOURS)
#15
#13
#12
#11
#10
#9

Figure F-6

Figure F-5 showed a marked contrast with F-2 regarding hydrogen generation. For the cases in which the intact cavity liners were vented to the heating and ventilation cooler room, the RCB hydrogen concentration did not reach 4 vol. % until after the sodium spilled into the subcavity. This difference was accounted for by the larger amounts of water released into the sodium pool in the cases depicted in Figure F-2. This meant that a sodium-water reaction of greater magnitude took place, which produced more hydrogen. The total area of concrete releasing water into the pool in these cases ($\sim$ 1400 ft^2) was much greater than for the cases where only that portion of the liner which was assumed to fail ($\sim$350 ft^2) released water into the sodium pool.

Figures F-7 through F-9 show a comparison of the CACECO predictions of RCB pressure, RCB hydrogen concentration, and the mass of the sodium pool in the reactor cavity, respectively, for the cases with an assumed sodium-concrete reaction rate of 3.0 inches/hour.

According to Figure F-7, at any given time, the RCB pressure for the case where the water released from behind the submerged intact cavity liners was vented to the heating and ventilation cooler room was lower than the RCB pressure for the cases where it was directed into the sodium pool. This was caused by the higher rate of sodium vapor generation in Cases 2 and 3. Examination of Figure F-9 bears this out. It should be noted that, from Figure F-8, the assumption that, under certain conditions, the hydrogen will recombine with the oxygen in the RCB extends the time that the RCB hydrogen concentration reaches 4% from about 12 hours in Case 2 (or $\sim$15 hours for Case 9), to approximately 18 hours for Case 3 (assuming recombination occurred). This means that the recombination assumption only delays the time at which action must be initiated to keep the hydrogen concentration below 4 vol. % by about 6 hours.

Figure F-10 shows a comparison of the time when the predicted RCB pressure reached the 10 psig design value as a function of sodium-concrete reaction rate for the two sets of water release path assumptions. From this comparison, it can be seen that the time at which the predicted RCB pressure reached 10 psig ranged from approximately 15 hours for Case 2 to 81 hours for Case 13 in those cases where the cavity floor was completely penetrated. This time is relatively independent of the water release path assumption. Figure F-11 shows the time at which the predicted RCB hydrogen concentration reached the 4 vol. % level for the cases where the water released from behind the intact submerged cavity liners was directed into the sodium pool as a function of sodium concrete reaction rate. It also compares these cases with the cases where this water was vented into the heating and ventilation cooler room. From this figure, it can be seen that this time is relatively insensitive to the sodium-concrete reaction rate for the cases where the released water went into the sodium pool. For the cases where this water was vented into the heating and ventilation cooler room, the time required to reach the 4 vol. % value varied from approximately 15 hours for the 3.0 inches/hour reaction rate case to about 45 hours for the proposed HEDL variable reaction rate case.

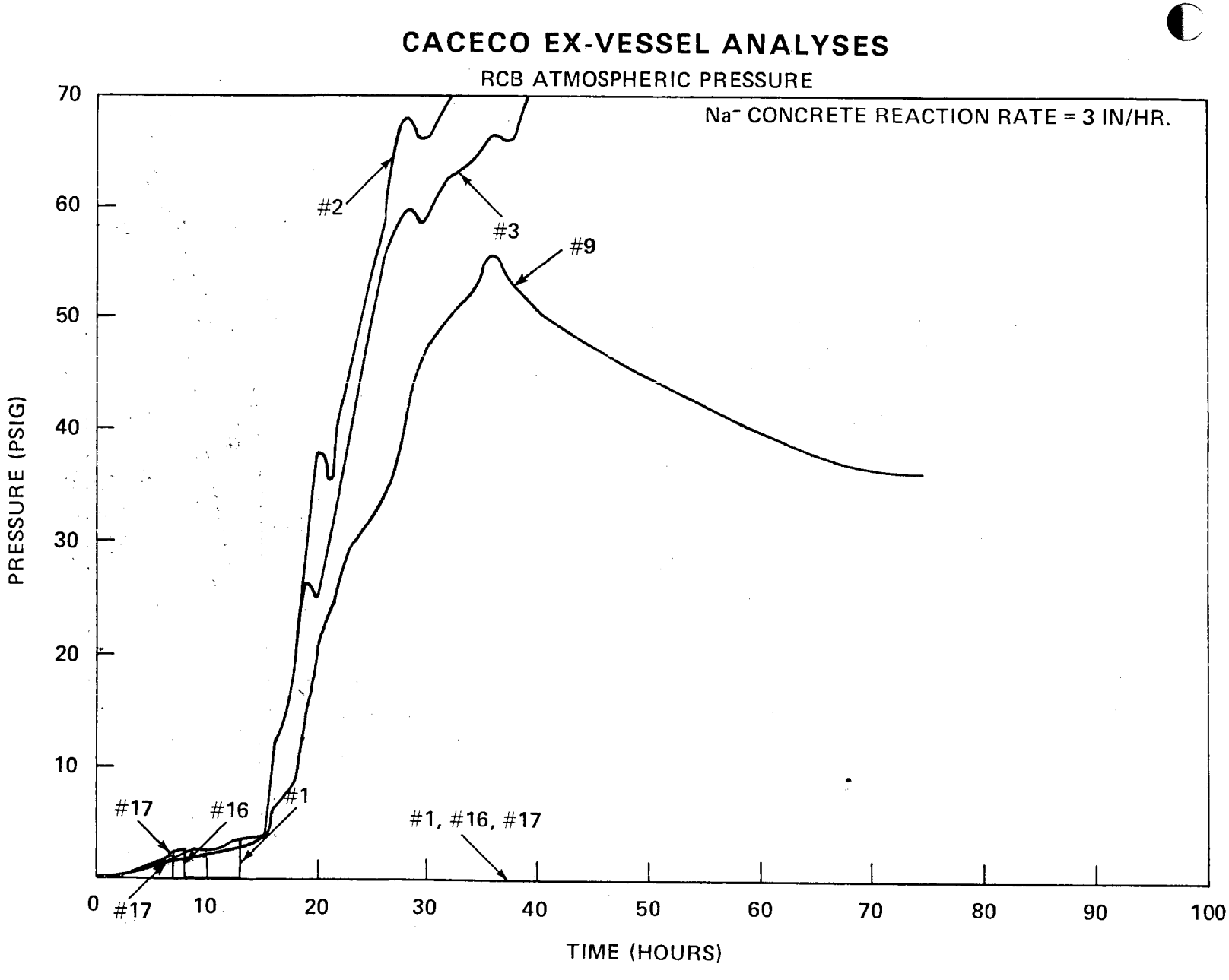

CACECO EX-VESSEL ANALYSES
RCB ATMOSPHERIC PRESSURE
Na⁻ CONCRETE REACTION RATE = 3 IN/HR.
PRESSURE (PSIG)
TIME (HOURS)
70
60
50
40
30
20
10
0
10
20
30
40
50
60
70
80
90
100
#1
#2
#3
#9
#16
#17
#1, #16, #17

Figure F-7

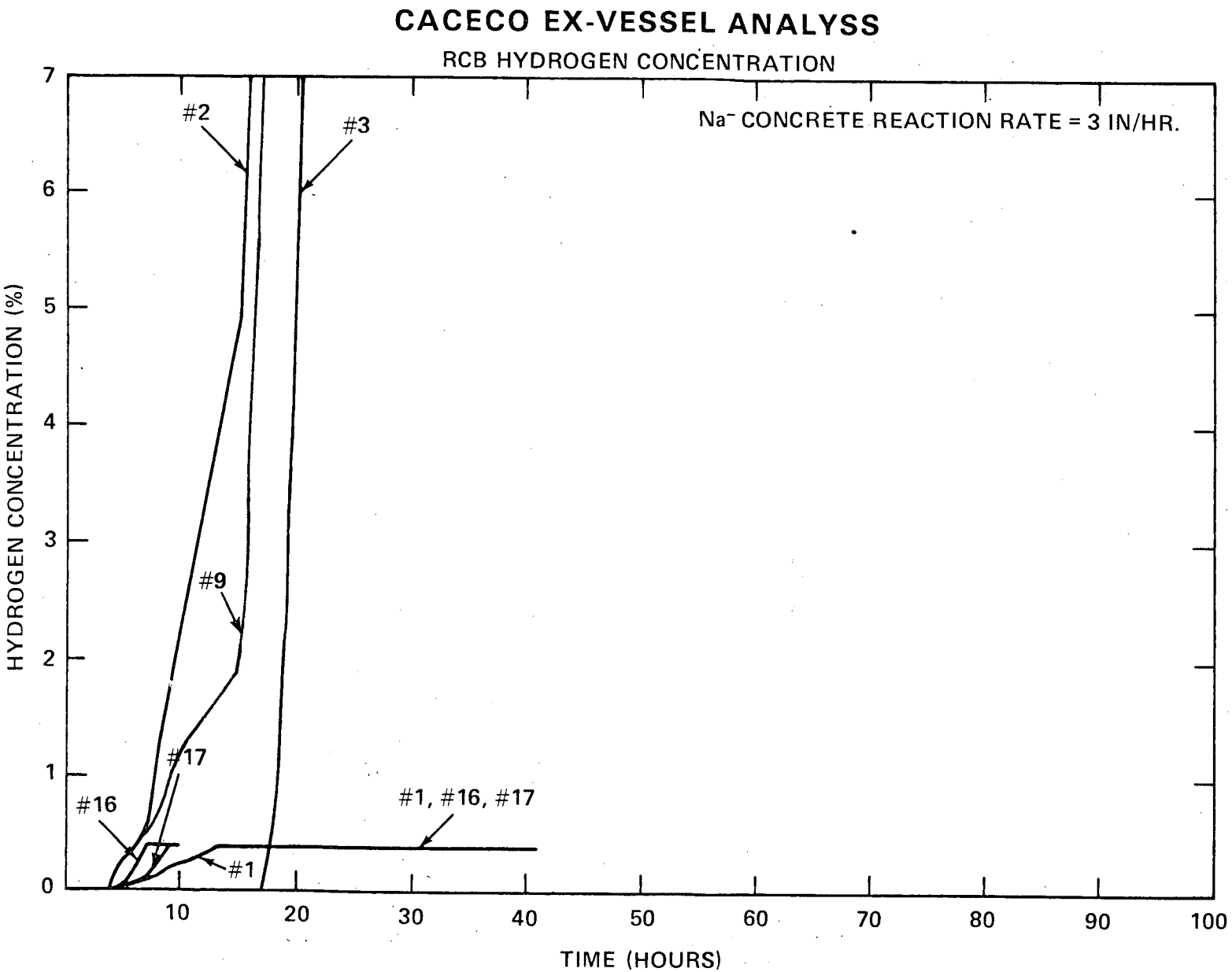

CACECO EX-VESSEL ANALYSS
RCB HYDROGEN CONCENTRATION
Na⁻ CONCRETE REACTION RATE = 3 IN/HR.
HYDROGEN CONCENTRATION (%)
TIME (HOURS)
#2
#3
#9
#17
#16
#1
#1, #16, #17
0
1
2
3
4
5
6
7
10
20
30
40
50
60
70
80
90
100

Figure F-8

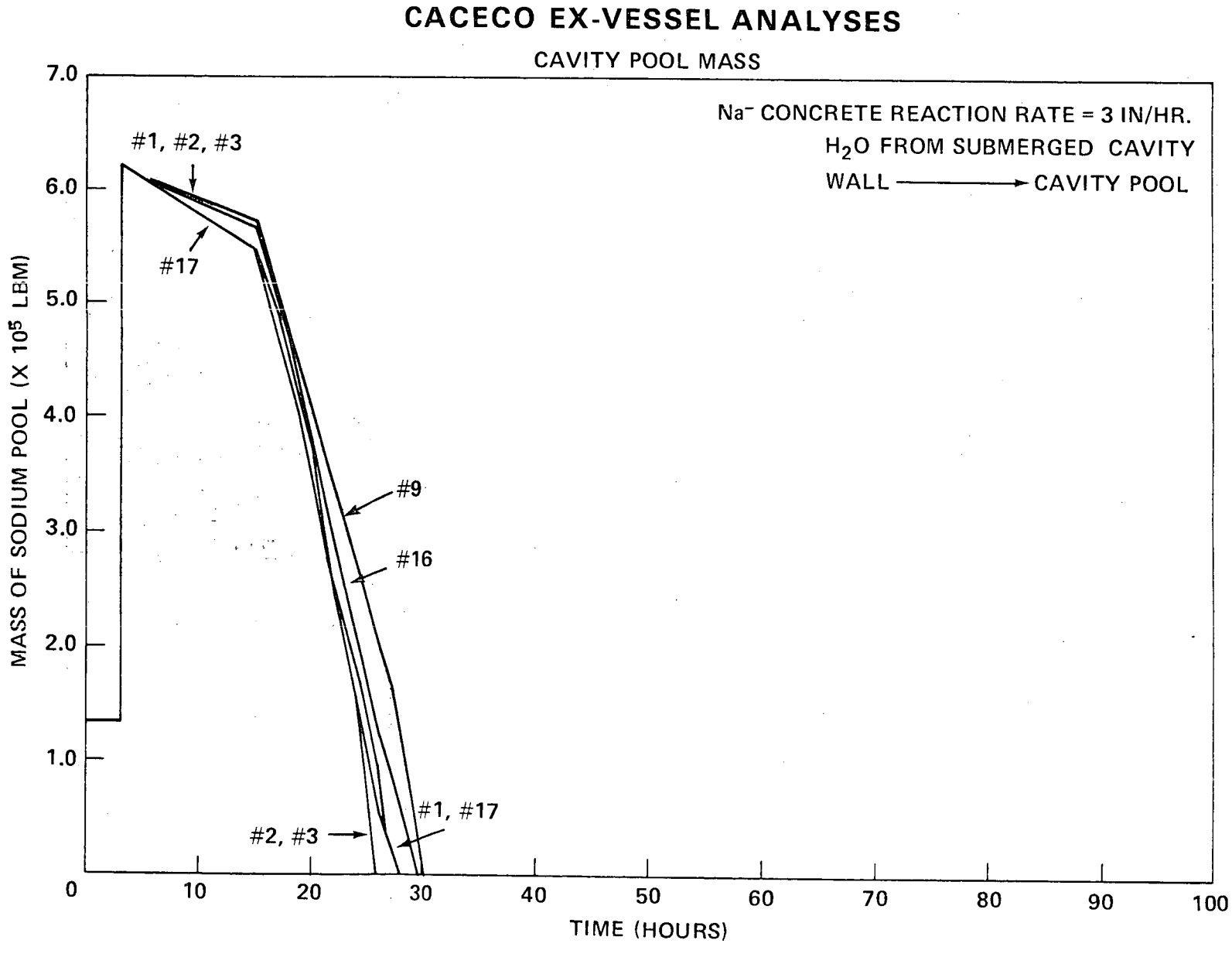

CACECO EX-VESSEL ANALYSES
CAVITY POOL MASS
Na⁻ CONCRETE REACTION RATE = 3 IN/HR.
H_2O FROM SUBMERGED CAVITY WALL → CAVITY POOL
#1, #2, #3
#17
#9
#16
#1, #17
#2, #3
MASS OF SODIUM POOL (X 10^5 LBM)
TIME (HOURS)
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0
10
20
30
40
50
60
70
80
90
100

Figure F-9

CACECO EX-VESSEL ANALYSES

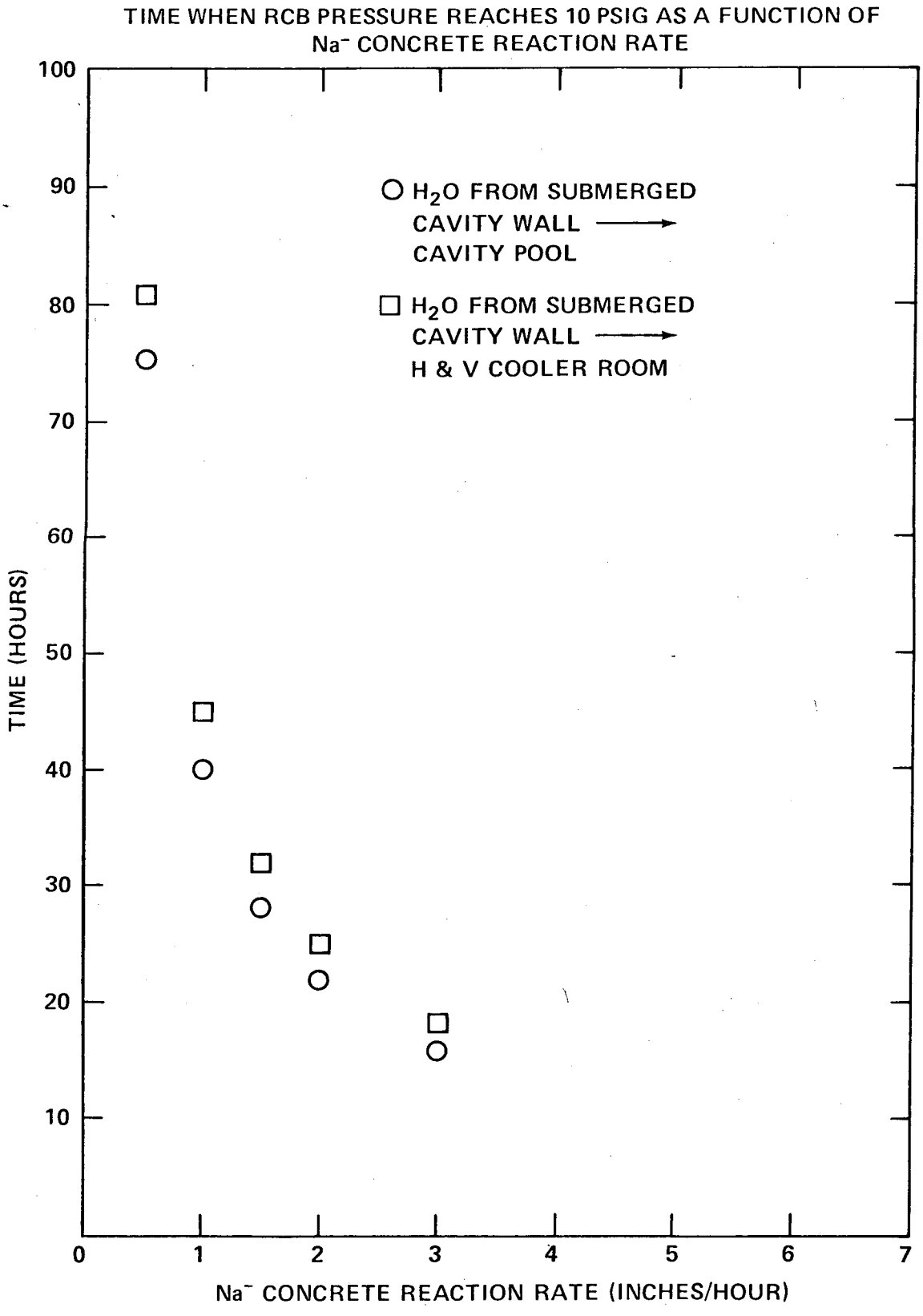


Figure F-10

CACECO EX-VESSEL ANALYSES

TIME WHEN RCB HYDROGEN CONCENTRATION REACHES 4% AS A FUNCTION OF
Na^- CONCRETE REACTION RATE

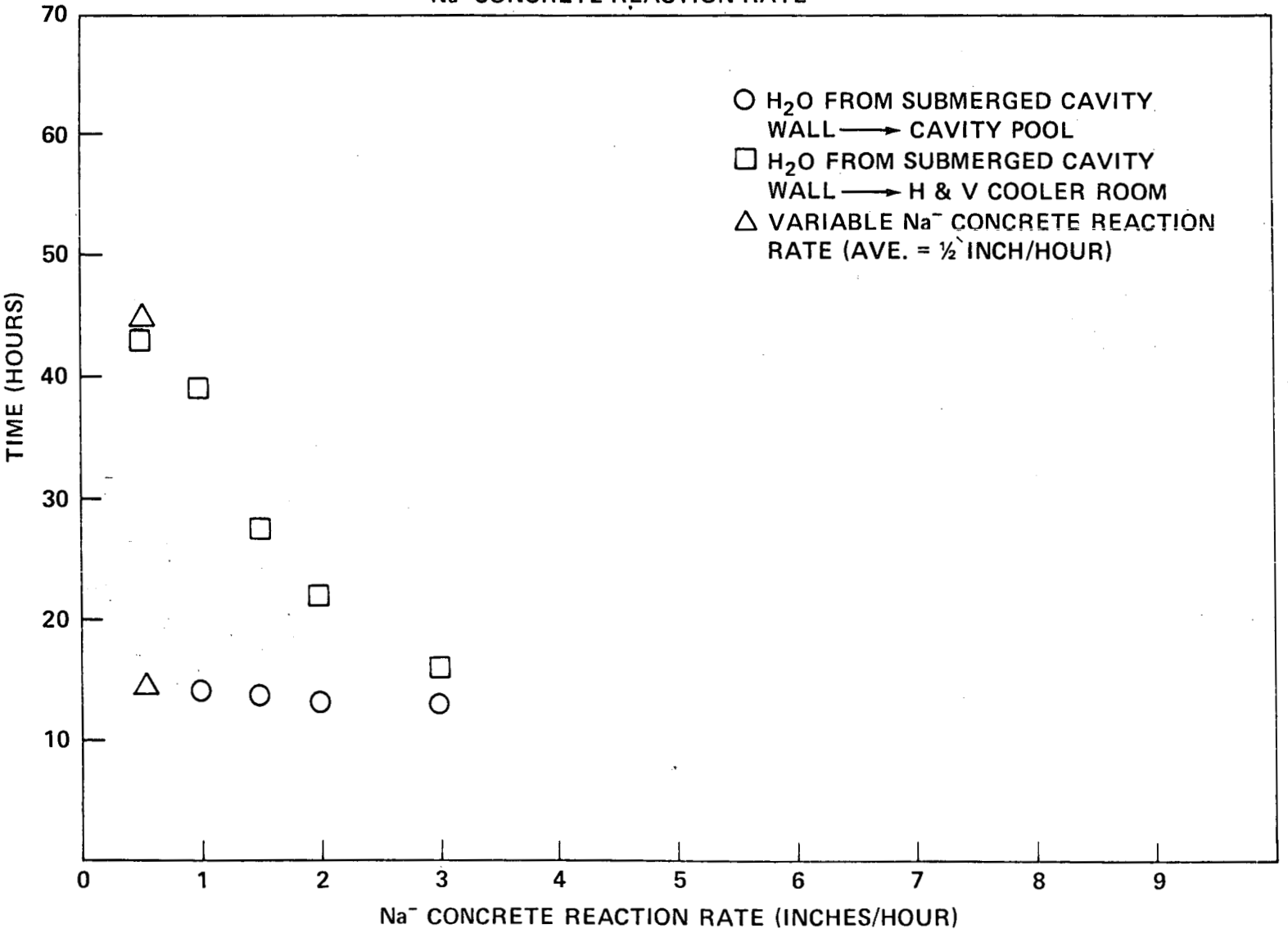


Figure F-11

Table F.2 contains a summary of the CACECO results for all of the cases run in this study. It should be noted that, with the exception of the case where the liner remained intact after the postulated melt-through, the only cases where the RCB shell temperature was predicted to be less than 500°F were those that assumed a reaction rate of 0.5 inch/hour. (The proposed HEDL reaction rate is, on the average over a 24-hour period, 0.5 inch/hour.) Complete time histories of the predicted RCB shell temperature were not available for all of the cases. Therefore, these results indicated that further analysis of the RCB shell temperature behavior during the course of the transient was required and this analysis is reported in the next section.

F.6 FURTHER ANALYSES

On May 22-23, 1978, the staff and its consultants met with the FFTF Project to discuss the issue of sodium-concrete reaction rates. During this meeting, the staff stated the position that a sodium-concrete reaction rate of 7 inches/hour for the first 0.5 hour and 1.0 inches/hour thereafter with no self-termination was considered an appropriate choice for realistic analyses of core meltthrough events, based on the available data regarding materials interactions. Subsequent to this meeting, the staff performed additional analyses considering this position.

In particular, the staff used the CACECO code[2] to examine the effect of the transient following a core melt-through event on the RCB shell temperature. These analyses were performed to determine whether the integrity of the containment penetration seals would be challenged by high temperatures (greater than 250°F) prior to the time at which RCB venting was required before purging to reduce the hydrogen concentration.

In this study, a constant sodium-concrete reaction rate of 1.0 inch/hour was used. Since the analyses discussed in the previous sections had indicated that there were essentially no differences between the results for the case using HEDL's proposed variable sodium-concrete reaction rate and a constant rate of 0.5 inch/hour while the reaction was in progress, it was felt that the use of a constant rate of 1.0 inch/hour was justified.

Two CACECO runs were made using Case 12 described in the previous sections of this Appendix as a base case. In one case, it was assumed that the RCB was vented to atmospheric pressure when the RCB hydrogen concentration approached 4 vol. %. An air purge was then initiated to maintain the hydrogen concentration at or below 4 vol. %. In the second case, the hydrogen concentration limit was assumed to be 6 vol. %. This latter case was run to determine the effect of the higher concentration limit on the vent/purge rates and RCB shell temperature. Figure F-12 compares the predicted RCB shell temperature for these two cases and for Case 12. The predicted RCB shell temperatures for the two vent/purge cases were as much as 300°F-400°F higher than the Case 12 predictions. This was due to the introduction of air into the RCB to dilute the hydrogen concentration. The oxygen in the incoming air reacted with the sodium

TABLE F.2

SUMMARY OF CACECO RESULTS

CASE NO.	TIME P_{RCB} = 10 PSIG (HOURS)	TIME RCB H_2 CONC. = 4% (HOURS)	MAX. RCB ATM. TEMP. (°F)	TIME T_{MAX} OCCURRED (HOURS)	REPRESENTATIVE RCB SHELL TEMP. (°F)	TIME T_{SHELL} OCCURRED (HOURS)
1	P_{MAX} = 9.3 PSIG AT 16 HRS.	13.0	2531.4	26.0	1287.0	41.0
2	15.7	13.0	1474.2	26.0	1000.0	41.0
3	15.7	19.8	1443.6	26.0	988.0	41.0
4	P_{RCB} = 5.8 PSIG AT 96 HRS.	N/A	148.5	96.1	141.0	96.0
5	22.3	14.0	1395.4	38.0	994.0	50.0
6	28.0	14.0	1312.9	50.0	1000.0	50.0
7	40.0	15.0	1187.4	75.0	926.0	75.0
8	75.5	15.0	448.0	100.0	346.0	100.0
9	18.1	15.9	1302.2	30.0	801.0	50.0
10	25.3	21.5	1197.5	44.0	712.0	69.0
11	43.5	27.0	1082.0	60.0	—	—
12	45.0	39.5	733.5	75.0	576.0	75.0
13	80.0	42.0	252.2	100.0	217.0	100.0
14	>100.0	14.0	170.0	75.0	149.0	75.0
15	>100.0	44.5	150.0	75.0	136.0	75.0
16	P_{MAX} = 2.7 PSIG AT 7.7 HRS.	9.0	1479.0	29.0	1094.0	29.5
17	P_{MAX} = 2.5 PSIG AT 7.0 HRS.	7.0	1460.0	28.0	1096.0	28.0

CONTAINMENT BUILDING SHELL TEMPERATURE
EX-VESSEL COOLABLE DEBRIS BED CASES

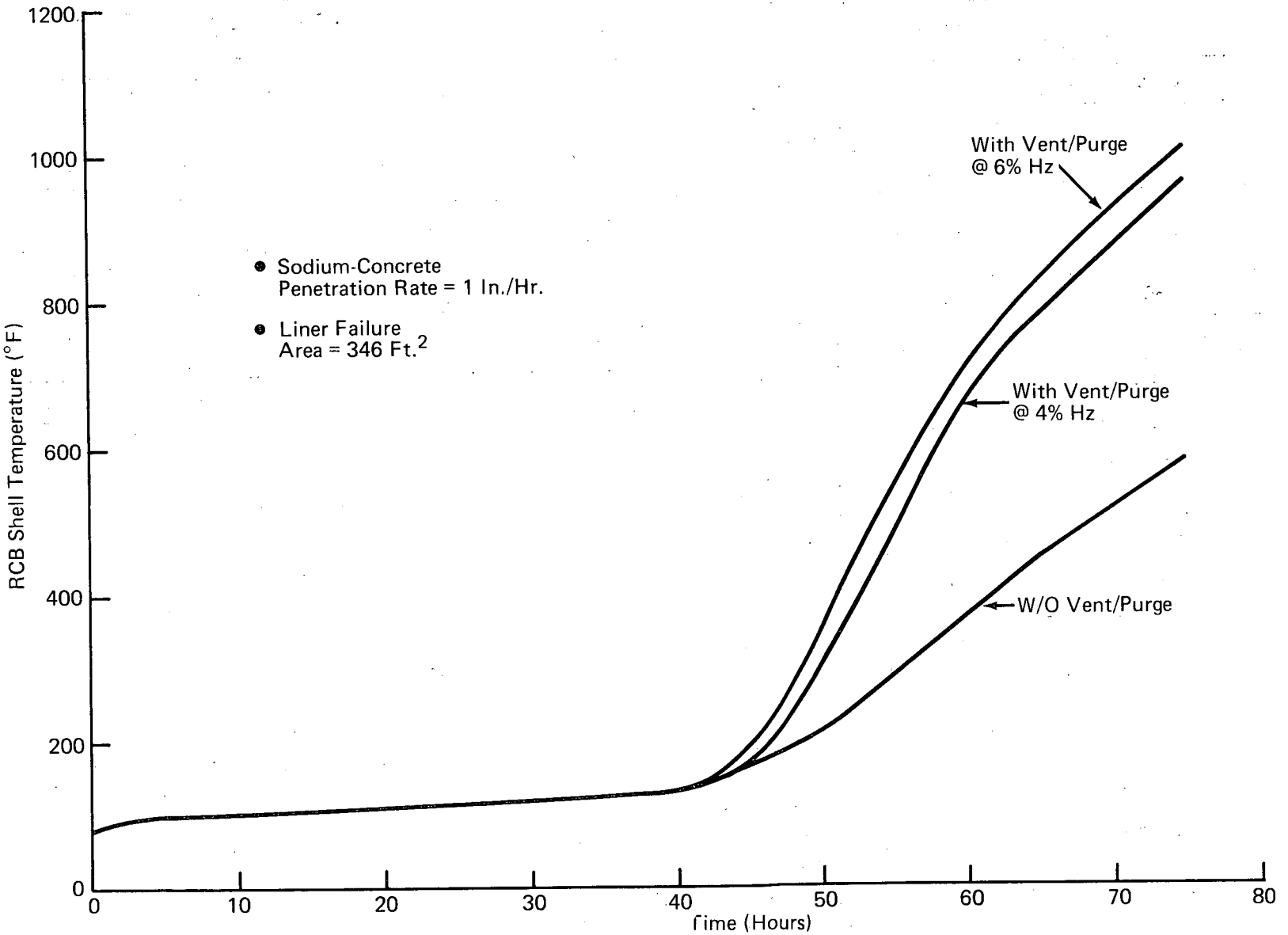


Figure F-12

vapor leaking into the RCB from the reactor cavity. This exothermic reaction (5700 BTU/lbm of Na) generated a significant amount of heat, thereby raising the temperature of the RCB atmosphere and causing a subsequent increase in the temperature of the RCB shell. However, in both cases, the RCB shell temperature did not exceed 200°F until after 45 hours. Since RCB venting occurred at about 40 hours, this meant that the integrity of the RCB penetration seals would not be challenged before the RCB was vented.

F.7 CONCLUSIONS

From the results of these analyses, the staff reached the following conclusions regarding the ex-vessel scenario:

(1) If the vent path for water released from behind the submerged intact cavity liners remains unobstructed, the time required for the RCB hydrogen concentration to reach a value of 4 vol. % (an indication that positive action is required to avoid challenging the containment due to the formation of explosive hydrogen concentration) was dependent on the sodium-concrete reaction rate.

(2) The time required for the RCB pressure to reach the design value of 10 psig was dependent on the sodium-concrete reaction rate. This dependency was observed whether or not the above mentioned vent path functioned in the desired manner.

(3) The time at which the containment could be challenged due to overpressure or formation of explosive concentrations of hydrogen ranged from as soon as 7 hours to as late as 100+ hours, depending upon the assumed scenario and conditions.

(4) The proposed HEDL variable sodium-concrete reaction rate resulted in a predicted containment transient which differed only slightly from that for a constant 0.5 inch/hr reaction rate for the period when both reactions were in progress.

(5) For the one inch per hour sodium-concrete reaction rate, the predicted containment shell temperatures were within acceptable limits as far as the integrity of the containment penetration seals (F250°F) prior to RCB venting was concerned.

REFERENCES

1. Containment Margins in FFTF for Postulated Failure of In-Vessel Post-Accident Heat Removal, HEDL-TME-77-18, April 1977.

2. Peak, R. D., "Users Guide to CACECO Containment Analysis Code," HEDL-TC-859, June 1977.

3. Peak, R. D., Private Communication

4. Peak, R. D., Private Communication

APPENDIX G

BIBLIOGRAPHY

NOTE: Documents referenced in or used to prepare this Safety Evaluation Report, excluding those listed in the Final Safety Analysis Report, may be obtained at the source stated in the Bibliography or, where no specific source is given, at most major public libraries. Correspondence between the Commission and the Department of Energy (FFTF Project Office) and Commission Rules and Regulations Guides may be inspected at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. Specific documents relied upon by the Commission's staff and referenced in this Safety Evaluation Report are listed as follows:

Section 1.0

1. Comparison of Foreign and U.S. LMFBR Plant Parameters, USAEC, WASH-1354, November 1974.

2. Liquid Metal Fast Breeder Reactor Program, Proposed Final Environmental Statement, USAEC WASH-1535, December 1973.

Section 2.0

3. Baldwin, J. L, Climates of the United States, U.S. Department of Commerce, Environmental Data Service, Washington, D.C., 1973.

4. Orgill, M. M., and G. A. Sehmel, "Frequency and Diurnal Variation of Dust Storms in Continguous U.S.A. Atmospheric Environment," Vol. 10, pp. 813-825, 1976.

5. Sagendorf, J. F., and J. T. Goll, NUREG-0324, XOQDOQ, "Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations," (DRAFT). U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washingont, D.C., 1976.

6. Stone, W. A., D. E. Jenne, and J. M. Thorp, "Climatography of the Hanford Area," BNWL-1605, Battelle Pacific Northwest Laboratories, Richland, Washington, 1972.

7. Thom, H.C.S., "New Distributions of Extreme Winds in the United States," Journal of the Structural Division, Proceedings of the American Society of Civil Engineers - July 1968, pp. 1787-1801, 1968.

8. U.S. Department of Commerce, Environmental Data Service, Local Climatological Data, Annual Summary with Comparative Data - Yakima, Washington. National Climatic Center, Asheville, N.C., 1976.

9. U.S. Department of Commerce, National Oceanic and Atmospheric Administration, "Climate of the States," Waster Information Center, Inc., Port Washington, N.Y., 1974.

10. Washington Public Power Supply System, Preliminary Safety Evaluation Report - Construction Stage, Nuclear Power Project Number 1, NRC Docket Numbers 50-460 and 50-513, July 1974.

11. Yanksey, G. R., E. H. Markee, and A. P. Richter, "Climatography of the National Reactor Testing Station," IDO-12048, Air Resources Field Research Office, Idaho Falls, Idaho, 1966.

Section 3.0

12. American Concrete Institute, "Building Code Requirements for Reinforced Concrete," ACI 318-1971, Detroit, Michigan.

13. American Institute of Steel Construction, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," Seventh Edition, New York, New York, 1970.

14. American Society of Civil Engineers, "Wind Forces on Structures," Final Report of the Task Committee on Wind Forces of the Committee on Load and Stresses of the Structural Division, Transactions of ASCE, Paper No. 3269, Vol. 126, Part II, New York, New York, 1961.

15. American Society of Mechanical Engineers, "ASME Boiler and Pressure Code," 1971 and 1972 Editions, Section III and VIII, New York, New York.

16. Bechtel Power Corporation, "Designs of Structures for Missile Impact," BC-TOP-9A, Revision 2, September 1974.

17. Hanford Engineering Development Laboratory, "Seismic Design Criteria for the Fast Flux Test Facility, Richland, Washington," Report No. JABE-WADCO-02, July 1975.

18. Hanford Engineering Development Laboratory, "Seismic Analysis Report for Building Structures," Report No. HEDL-BR-5853-C-2, Revision 2, Vol. 1, May 1977; Vol. 2, September 1977; Vol. 3, August 1977; and Vol. 4, September 1977.

19. Hanford Engineering Development Laboratory, "Cell Liner Integrity Evaluation for the Fast Flux Test Facility," Report No. HEDL-TC-823, Rev. 1, October 1977.

20. Hanford Engineering Development Laboratory, "Analysis of FFTF Reactor Cavity Liners under Ex-vessel PAHR Conditions," Report No. TC-572, February 1976.

21. Hanford Engineering Development Laboratory, "Containment Margins in FFTF For Postulated Failure of IN-vessel Post-accident Heat Removal," Report No. HEDL-TME 77-18, April 1977.

Section 4.0

22. Fast Flux Test Facility System Design Description No. 31, Rev. 48, Hanford Engineering Development Laboratory, December 31, 1975.

23. Sim R. and A. Veca, "FFTF Fuel Pin Final Design Support Document" (Transmitted by FCF-214) December 10, 1971, and updated (Sec. 3.3, 3.4, and 4.0) by letter FCF-272, ARD- FCF-213, February 28, 1972.

24. Eberlein W. C., R. E. Bears, J. D. Watrous, and R. J. Jackson, "Revised FFTF Cladding Temperature Integrity Limit and Update Evaluation of the FFTF Fuel Pin Transient Design Procedure," HELD-TME 76-32, May 1976.

25. Leggett R. D., et al, "Linear Heat Rating for Incipent Fuel Melting in UO_2 - PuO_2 Fuel," HEDL-SA 464, Trans. American Nuclear Soc. 15, p. 752, October 1972.

26. Leggett R. D., et al, "Influence of Burnup on Heat Rating-to-Melting for UO - PuO Fuel," HEDL-SA-737, Trans. American Nuclear Soc., 19, p. 136, October 1974.

27. Dienst V. and A. Zimmerman, "UO_2 - PuO_2 Fuel Pin Capsule Irradiations of the Mol 8B and C Test Groups: Evaluation of Measuring and Post-Irradiation Examination Results," EURFNR-3187, September 1976.

28. Mueller I., "Studies on the Mechanical Interaction Between Oxide Fuel and Cladding in the Case of Power Changes in Fast Breeder Fuel Rods," EURFNR-1409, November 1976.

29. Ross D. F. (NRC), Memorandum to R. P. Denise, "CRBR SER Input," September 19, 1977.

30. Baars R. E., "Evaluation of FFTF Fuel Pin Transient Design Procedures," HEDL-TME-75-40, September 1975.

31. Jackson R. J., "Evaluation of FFTF Fuel Pin Design Procedure Vis-a-vis Steady State Irradiation Performance in EBR-II," HEDL-TME-75-48 Addendum, November 1975.

32. Travis M. L, "Exposure Dependent Cladding Deformations," WARD-D-0146, July 1976.

33. Nuclear Systems Materials Handbook, TID 26666, Hanford Engineering Development Laboratory, Richland, WA.

34. Bowen P. H., "Adhesive Wear (Galling) of Materials in Liquid Sodium," Report 70-1BS-HAWAR-R1, Westinghouse Research Laboratories, May 1970.

35. Kleefeldt K., "Investigation of Wear and Friction Between Fuel Rod Bundle and Spacer Grids in Sodium," Report KNK-1290, Karlsruhe Nuclear Center, October 1970 (EURFNR-854).

36. Liquid Metal Engineering Center, "Annual Technical Progress Report, January-December 1972," LMEC-73-1, February 28, 1973, pp. 1-135 to 1-144.

37. Liquid Metal Breeder Reactor Program, Thirteenth Quarterly Report, August-October 1975, GEAP-13923-13, November 1975.

38. Wald J. W. and I. S. Levy, "LMFBR Fuel Pin Cladding Transient Performance Capabilities: An Analysis of FCTT Data," BNWL-2041, March 1976.

39. Liquid Metal Breeder Reactor Program, Sixteenth Quarterly Report, May-July 1976, GEAP-13923-16, August 1976.

40. "HEDL Transient Irradiation Testing Program," HEDL TI-76035-3, October 13 and 14, 1976.

Section 6.0

41. Shire P. R., J. P. Hale, and G. R. Franz "Sodum Spill and Fire Evaluation Report for the Fast Flux Test Facility" HEDL-TME-75-122, April 1976.

42. Shire P. R., "SPRAY-2 Code Description," Argonne Code Center, June 1976.

43. Beiriger P., et al., "SOFIRE-II User Report," AI-AEC-13055, Atomics International, 1973.

44. Tsai, S. S., R. D. Gasser, and W. T. Pratt, "Sodium Fire Evaluation for the Fast Flux Test Facility," BNL-NUREG 24635, August 1978.

Section 15.0

45. FFTF Fuel Pin Final Design Support Document, Issued December 10, 1971; revised February 28, 1972, R. Sim and A. Veca.

46. HEDL-TC-556, "Simulation of the Overall FFTF Plant Performance (IANUS-Westinghouse Proprietary Code), SL Addition, T. B. McCall and C. F. Wolfe, December 1975.

47. Wilburn, N. P. et al, "An Updated Assessment of the Unprotected Transient Overpower Accident in the FTR," HEDL-TI 7585, October 1977.

48. Fast Flux Test Facility, Preliminary Safety Analysis Report, WADCO Corporation, September 1970.

49. Culley, G. E. et al, "Fast Reactor Safety Implications of Recent Assessments of Fuel Pin Transient Behavior, "International Conference on the Engineering of Fast Reactors for Safe and Reliable Operation, Kernforschungszentrum Karlsruhe, October 9-13, 1972, (pp. 626-641).

50. "Safety Evaluation of the Fast Flux Test Facility," USAEC, Directorate of Licensing, Washington D. C. October 31, 1972.

51. Supplement No. 1 to the "Safety Evaluation of the Fast Flux Test Facility," USAEC, Directorate of Licensing, Washington D. C. December 13, 1974.

52. Meyer, J. F. et al, "An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Accident Energetics," NUREG-0122, NRC, Washington, D. C. March 1977.

53. Waltar, A. E. et al, "Melt-III, A Neutronics, Thermal Hydraulics Computer Program for Fast Reactor Safety Analysis," HELD-TME 74-47, December 1974.

54. Dunn, F. E. et al, "The SAS3A LMFBR Accident Analysis Computer Code," ANL/RAS 75-17, April 1975.

55. Rumble, E. T., "A Hypothetical Overpower Excursion Model for LMFBR's," Ph. D. Dissertation in Engineering, UCLA, 1974.

56. Page, R. J. et al, "Preliminary Data Report: TREAT H 6," ANL/RAS 77-55, December 1977.

57. Koopman, R. N. et al, "TREAT Transient Overpower Experiment R12," Transactions ANS, Vol. 28, June 1978 (pp. 482).

58. Spencer, B. W. et al, "Camel TOP/Fuel Sweepout Single-Pin Test C2," ANL/RAS 77-22, July 1977.

59. Bohl, W. et al, "A Preliminary Study of the FFTF Hypothetical Flow Coast Down Accident," ANL/RAS 71-39, Revision 1, April 1971.

60. Dunn, F. E. et al, "The SAS2A LMFBR Accident Analysis Code," Proc. Conf. on New Developments in Reactor Mathematics and Applications, CONF-710302, March 1971.

61. Stevenson, M. G. et al, "Report on the Analysis of the Initiating Phase of a Loss-of-Flow (Without Scram) Accident in the FTR," ANL/RAS 74-24, Argonne National Laboratory, September 1974.

62. Jackson, J. F. et al., "Report on the Core Disruption Phase of an Unprotected Flow-Coast-down in the FTR," ANL/RAS 74-16 Argonne National Laboratory, August 1974.

63. Fauske, H. K. and Ferguson, D. R., "An Updated Assessment of the Unprotected Loss-of-Flow Accident in the FTR," ANL/RAS 77-42, October 1977.

64. Miao, C. and T. G. Theofanous, "Transient Two-Dimensional Sodium Boiling in LMFBR Subassemblies: The HEV-2D Code," PNE 78-130, Purdue University (1978).

65. Kikuchi, Y. and K. Hoga, "Transient Boiling of Sodium in a 19-Pin Bundle Under Loss-of-Flow Conditions," JAPFNR-262, June 1977.

66. Theofanous, T. G. et al, "Boiling Incoherency Effects in Unprotected Loss-of-Flow Accidents," ANS Trans. 27, 537 (1977).

67. Bohl, W. R. and T. J. Heames, "CLAZAS: The SAS3A Clad Motion Model," ANL/RAS 74-15, Argonne National Laboratory, August 1974.

68. Theofanous, T. G. et al, "Incoherency Effects in Clad Relocation Dynamics for LMFBR CDA Analyses," Nuclear Eng. Design, Vol. 36, No. 1, January 1976.

69. Henry, R. E. et al, "28-Pin Cladding Relocation Experiments," ANS Trans. 27, 498 (1977).

70. Fauske, H. K, "Some Comments on Cladding and Early Fuel Motion," (Interim Report), ANL/RAS 76-11, Argonne National Laboratory, March 1976.

71. Palm, R. G. et al, "F1 Phenomenological Test on Fuel Motion," (Interim Report), ANL/RAS 76-11, Argonne National Laboratory, March 1976.

72. Deitrich, W. L. and R. W. Ostensen, "An Assessment of Fission-Gas-Driven Fuel Disruption and Dispersal in a Hypothetical LMFBR Loss-of-Flow Accident," ANL/RAS 77-4, February 1977.

73. Cantley, D. A. et al, "Axial Expansion at Mixed Oxide Fuel Measured In-Pile," GEAP-10403 (1971).

74. Theofanous, T. G., "Multiphase Transients with Coolant and Core Materials in LMFBR Core Disruptive Accident Energetics Evaluation," NUREG/CR-0224, USNRC, July 1978.

75. Ostensen, R. W. and R. J. Henninger, "Recriticality in the Transition Phase in CMFBR Accidents," ANS Trans. 24 346 (1975).

76. Boudreau, J. E. and J. F. Jackson, "Recriticality Considerations in LMFBR Accidents," CONF-740401, 1265 (1974).

77. Jackson, J. F. and R. B. Nicholson, "VENUS-II, An LMFBR Disassembly Program," ANL-7951, September 1972.

78. Padilla, A., "Transient Analysis of Fuel-Sodium Interactions," Trans. Am. Nucl. Soc. 13 375 (1970).

79. Florence, A. L. and G. R. Abrahamson, "Simulation of a Hypothetical Core Disruptive Accident in a Fast Flux Test Facility," HEDL-SR-1, May 1973.

80. Stepnewski, D. D. and C. R. Adkins, "Final Report of Task Force for Assessment of Additional FFTF Design Margin," WARD-317, August 8, 1974.

81. Cho, D. H. and M. Epstein, "Work Potential Resulting from a Mechanical Disassembly of the Voided FFTF Core," ANL/RAS 74-17, August 1974.

82. Change, Y. W. and J. Guildys, "REXCO-HEP" A Two-Dimensional Computer Code for Calculating the Primary System Response in Fast Reactors," ANL-75-19, June 1975.

83. Graham, J., "HCDA Requirements and Loadings for FFTF Design Evaluation," FS-257, Revision 1, July 1976.

84. "Containment Margins in FFTF for Postulated Failure of In-Vessel Post Accident Heat Removal," HEDL-TME 77-18, April 1977.

85. "FFTF Secondary Sodium Fire Protection System Test FI," HEDL-TME 73-48, April 1973.

86. Memorandum to T. P. Speis, LMFBR Branch, NRR, from M. Silberberg, Chief, Experimental Fast Reactor Safety Research Branch, RES, dated August 5, 1977, "Quick-Look Report on Sandia Preliminary Sodium-Concrete Interaction Tests 1 and 2."

87. Memorandum to T. P. Speis, Chief, LMFBR Branch, NRR, from M. Silberberg, Chief, Experimental Fast Reactor Safety Research Branch, RES, October 31, 1977, "Quick Look Report on Sandia Preliminary Sodium-Concrete Interaction Test 3."

88. Peak, R. D. and D. D. Stepnewski, "Computational Containment Analysis Code," ANS Trans., Vol. 21, p. 274, June 1975.

89. Baker, L. Jr. et al, "Post-Accident Heat Removal Technology," ANL/RAS 77-2, dated January 1, 1977.

90. Gasser, R. D. and W. T. Pratt, "Containment Response to Postulated Core Meltdown Accidents in the Fast Flux Test Facility," BNL-NUREG 24141-R, Revised August 1978, Published March 1978.

91. "Post-Accident Heat Removal Assessment for the FFTF," HEDL-TME 75-60, November 1974.

92. Murfin, W. A., "A Preliminary Model for Core/Concrete Interaction," Sandia Laboratories, SAND 77-0370, August 1977.

93. "Experimental Study of Hydrogen Formation and Recombination Under Postulated LMFBR Accident Conditions," HEDL-TC-730, December 1976.

94. "Intermediate Scale Sodium-Concrete Reaction Tests," HEDL-TME-77-99, August 1977.

95. For a description of the GROWS Code see Chapter V of: R. Kumas, L. Baker, Jr., and M. G. Chasanov, "Ex-Vessel Considerations in Post-Accident Heat Removal," ANL/RAS 74-29, October 1974.

96. Powers, D. A. et al, "Exploratory Study of Molten Core Material/Concrete Interactions July 1975 - March 1977," Sandia Laboratories, SAND 77-2042, February 1978.

97. Stepnewski, D. and R. Peak, CACECO Code Modifications, in HEDL Quarterly Technical Report, October-December 1974, HEDL-TME 74-4, Vol. 3 (1974). See also R. D. Peak and D. Stepnewski, Trans. Am. Nucl. Soc. 21 p. 274 (1975).

98. Gieseke, J. A., K. W. Lee, and L. D. Reed, HAARM 3 Users Manual, BMI-NUREG-1991 January 5, 1978.

99. USNRC In-House Calculations of Radiological Effects of Accidents Using TACT-4, Unpublished.

100. Bell, M. J., ORIGEN, the ORNL Isotope Generation and Depletion Code, ORNL-4628, (1973).

101. Memorandum to L. G. Hulman, Chief, Hydrology-Meteorology Branch, DSE, from R. Jubach, "Trip Report to INEL for Sodium Combustion Tests," June 28, 1978.

102. FFTF Secondary Sodium Fire Protection System Test F-1, HELD-TME 73-48, April 1973.

103. Cybulskis, P., "Effect of Engineered Safety Features on the Risk of Hypothetical LMFBR Accidents," Nuclear Safety 19, 2, p. 190, March-April 1978.

104. Jordan, S. and Y. Ozawa, "Fuel Particle and Fission Product Release from LMFBR Core Catcher," CONF-761001, October 1976.

105. Armstrong, D. R., F. J. Testa and D. Raridon, "Molten UO_2-Sodium Dropping Experiment," Trans. Am. Nucl. Soc. 13, p. 660, November 1970.