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MASTER

CORE DESIGN DEVELOPMENT NEEDS IN RELATION TO FUEL FAILURE PROPAGATION, SODIUM BOILING AND CLAD/FUEL-SODIUM THERMAL INTERACTION

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CONTRACT AT(04-3)-189 PROJECT AGREEMENT 10

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TABLE OF CONTENTS VOLUME I

<i>Section</i>		<i>Page</i>
	ABSTRACT	1-1
1.0	INTRODUCTION	1-2
	1.1 Scope of Work	1-2
	1.2 Priority Conclusions and Recommendations	1-2
	1.3 Priority Areas for Safety Research	1-2
	1.4 Priority Area for Design Development	1-5
	1.5 Recommendations for Further Definition Studies	1-7
	1.6 Study Methods and Approach	1-7
2.0	FAULT TREE FORMAT FOR ASSESSING DESIGN NEEDS	2-1
	2.1 Selection of Fault Trees	2-1
	2.2 Fault Tree Final States	2-3
	2.3 Fault Tree Initiating Events	2-4
	2.4 Relationship to LMFBR Design Philosophy	2-6
	2.5 Nomenclature and Usage for Fault Trees	2-8
	2.6 Fault Tree Examples	2-9
	2.7 Assignment of Quantitative Probabilities	2-16
3.0	CONCLUSIONS AND RECOMMENDATIONS	3-1
	3.1 Introduction	3-1
	3.2 Fault Tree Results	3-1
	3.3 Design Development Needs	3-13
	3.4 Safety Research Needs	3-25
	APPENDIX I REVIEW OF LMFBR DESIGN AND SAFETY PHILOSOPHY	I-I
	APPENDIX II OVERALL FAULT TREES	II-I
	APPENDIX III MECHANISTIC FAULT TREES AND STATE OF THE ART REVIEW	III-I
	APPENDIX IV FAULT TREE DOMINANT CUT-SETS	IV-I
	NOMENCLATURE AND DEFINITIONS	IV-XV

TABLE OF CONTENTS VOLUME II

<i>Section</i>		<i>Page</i>
	ABSTRACT	4-1
	INTRODUCTION	4-2
4.0	OPERATING SAFETY LIMITS FOR FUEL PINS	4-4
	4.1 Cladding Overheating	4-4
	4.2 Overpower and Fuel Melting Perturbations	4-30
	4.3 Mechanical Perturbations In Excess of Design Loads	4-41

TABLE OF CONTENTS **VOLUME II (continued)**

<i>Section</i>	<i>Page</i>
5.0 OPERATING SAFETY LIMITS FOR FUEL CHANNELS	5-1
5.1 Uniform Overheating of the Channel	5-1
5.2 Response to Dynamic Loading	5-5
6.0 LONG TERM FUEL FAILURE PROPAGATION EFFECTS	6-1
6.1 Fuel-Sodium Reaction Product Swelling	6-1
6.2 Fuel-Clad Debris Blockage	6-22
6.3 Degradation of Fuel Failures Within A Single Pin	6-48
7.0 RAPID FUEL FAILURE PROPAGATION EFFECTS	7-1
7.1 General Characteristics of Sodium Boiling	7-1
7.2 Sodium Hammer and Vapor Bubble Collapse	7-17
7.3 Molten Fuel/Clad – Sodium Thermal Interactions	7-44
APPENDIX A DERIVATIONS FOR ANALYSIS	A-1
APPENDIX B POTENTIAL MECHANISMS FOR FUEL EXPULSION	B-1
APPENDIX C DERIVATION OF THE "WAVE" EQUATION	C-1

LIST OF ILLUSTRATIONS

VOLUME I

<i>Figure</i>	<i>Title</i>	<i>Page</i>
1-1	Dominant Accident Chain	1-3
2-1	Overall Fault Tree Example	2-10
2-2	Example of a Mechanistic Fault Tree	2-11
2-3	Example of a Dominant Cut-Set	2-13
2-4	Example of a Mechanistic Dominant Cut-Set	2-14
2-5	Fault Tree Logic	2-15
3-1	Sensitivity of Mean Annual Cost to Changes in Initiating Event Frequency	3-7
II-1/17	Overall Fault Trees	A/P
III-1	Q101 Crack Failure Progresses to Large Area Failure in the Same Pin (Also applies to Q115 and 610)	III-II
III-2	Q102 Crack Failure Progresses to Burst Failure in the Same Pin (Also applies to Q117, 600)	III-V
III-3	Q103 (104, 105) Initial Crack Failure Induces Burst (Large Area, Crack) Failure in an Adjacent Pin (Also applies to Q112, 113, 114)	III-VI
III-4	Q201 Large Area Failure Progresses to a Burst Failure in the Same Pin (Also applies to Q215, 620)	III-VII
III-5	Q202 (203, 204) Initial Large Area Failure Induces a Burst (Large Area, Crack) Failure in an Adjacent Pin (Also applies to Q212, 213, 214)	III-VIII
III-6	Q301 (302, 303) Initial Burst Failure Induces Burst (Large Area, Crack) Failure in Adjacent Pin (Also applies to Q311, 312, 313)	III-IX
III-7	Q416 (417, 418) Partial Blockage with Reduced Flow Causes Burst (Large Area, Crack) Failures in Many Pins	III-X
III-8	Q34 (42, 54) Many Pins with Large Area Failures Cause Rupture of the Channel Walls of One (Few, Many) Bundle	III-XI
III-9	Q36 (43, 55) Many Pins with Crack Failures Cause Rupture of the Channel Wall of One (Few, Many) Bundle	III-XII
III-10	Q28 (30, 32), Q39 (40, 41) and Q51 (52, 53) One (Few, Many) Burst Fuel Pin Causes Rupture of the Channel Wall in One (Few, Many) Bundle	III-XIII
III-11	Q650-667 One (Few, Many) Burst Failure in One Bundle Causes Many Burst (Large Area, Crack) Failures in Few (Many) Sound Bundles	III-XIV
III-12	Q406 (407, 408) Partial Blockage with Coolant Boiling Causes Burst (Large Area, Crack) Failures in Many Pins	III-XV
III-13	Q701 (702, 703), 811 (812, 813) Channel Voiding Causes Many Crack (Large Area, Burst) Failures in a Few (Many) Sound Bundles	III-XVII
III-14	Q37 (44, 56) Channel Voiding Causes Immediate Failure of the Channel Walls of One (Few, Many) Bundles	III-XVIII
IV-1	Events Which Result In Reactor Contamination By Fission Gas From A Bundle With One Failed Fuel Pin	IV-I
IV-2	Mechanisms Through Which Crack Failure In One Pin Progresses To Large Area Failure In The Same Pin (Q-115, Q-101, Q-610)	IV-II
IV-3	Events Which Result In Reactor Contamination By Fission Gas From A Sound Bundle With A Few Failed Pins	IV-III
IV-4	Mechanisms Through Which A Large Area Failure In One Pin Induces Crack Failures In One Or More Adjacent Pins (Q-204)	IV-IV
IV-5	Events Which Result In Reactor Contamination By Fission Gas From A Bundle With Many Failed Pins	IV-IV

LIST OF ILLUSTRATIONS **VOLUME I (continued)**

<i>Figure</i>	<i>Title</i>	<i>Page</i>
IV-6	Mechanisms Through Which Large Area Failure In One Pin Induces Burst Failure In An Adjacent Pin (Q-202)	IV-V
IV-7	Mechanisms Through Which Burst Failures Induce Crack Failures In Adjacent Pins (Q-313)	IV-V
IV-8	Events Which Result In Reactor Contamination By Fuel And Fission Product Release From A Bundle With One Failed Pin	IV-VI
IV-9	Events Which Result In Reactor Contamination By Fuel And Fission Product Release From A Bundle With A Few Failed Pins.	IV-VI
IV-10	Events Which Result In Reactor Contamination By Fission Product And Fuel Release From A Bundle With Many Failed Fuel Pins	IV-VII
IV-11	Events Which Result In Reactor Shutdown To Remove One Fuel Bundle With Undamaged Wall	IV-VIII
IV-12	Mechanisms Through Which A Burst Failure Induces A Burst Failure In One Or More Adjacent Pins (Q-311)	IV-VIII
IV-13	Events Which Result In Reactor Shutdown To Remove A Few Bundles With Undamaged Walls	IV-IX
IV-14	Mechanisms Through Which Fuel Pin Burst Failures In One Bundle Cause Fuel Pin Failures In Other Bundles, Without Damaging The Channel Walls (Q-662)	IV-X
IV-15	Events Which Result In Reactor Shutdown To Remove Many Bundles With Undamaged Walls	IV-XI
IV-16	Events Which Result In Reactor Shutdown To Remove One Bundle With Damaged Wall	IV-XII
IV-17	Mechanisms Through Which Many Burst Failures Cause Channel Wall Failure (Q-32, -41, -53)	IV-XII
IV-18	Events Which Result In Reactor Shutdown To Remove A Few Bundles With Damaged Walls	IV-XIII
IV-19	Events Which Result In Reactor Shutdown To Remove Many Bundles With Damaged Walls	IV-XIV

LIST OF ILLUSTRATIONS **VOLUME II**

<i>Figure</i>	<i>Title</i>	<i>Page</i>
4-1	Rupture Stress of Irradiated Type 304 and 316 Stainless Steel	4-11
4-2	Stress for Rupture as a Function of Time-To-Rupture Type 304 & 316 Stainless Steel	4-12
4-3	Corrosion Rate vs Temperature for Stainless Steel Type 304 & 316	4-14
4-4	Free Energy of Formation for Oxides of Stainless Steel Constituents and Mixed-Oxide Fuel	4-15
4-5	Fuel-Clad Interaction	4-16
4-6	Assumed Cladding Attack Rate	4-18
4-7	Stress Histories for Sealed Pins	4-19
4-8	Stress Histories of Sealed Pins	4-20
4-9	Stress Histories for Vented Pins	4-21
4-10	Stress Histories of Vented Pins	4-22

LIST OF ILLUSTRATIONS **VOLUME II (continued)**

<i>Figure</i>	<i>Title</i>	<i>Page</i>
4-11	Time-To-Failure vs Temperature	4-23
4-12	Time-To-Failure vs Temperature	4-24
4-13	Operating Life for Case S3B	4-25
4-14	Operating Life for Case V2B	4-26
4-15	Sealed-Vented Comparison (15 Mil Clad)	4-27
4-16	Comparison of Optimistic, Conservative and Damage Accumulation Failure Criteria for Case S3B	4-28
4-17	Burst Pressure vs Clad Temperature	4-29
4-18	Fuel Melt Fraction Required to Fail Clad	4-33
4-19	Fuel Pin Interior Pressure History	4-34
4-20	Failure (Breach of Primary Containment) Correlation	4-36
4-21	Fuel Pin Interior Pressure History	4-37
5-1(a)	Recommended Tensile Properties Above Total Fluences of 10^{22} nvt As A Function of Test Temperature — Type 304 and 316 Stainless Steel	5-2
5-1(b)	Recommended Tensile Properties as a Function of Total Fluence At 600° and 700°C — Type 304 and 316 Stainless Steel	5-2
5-2	Channel Internal Pressure Which Causes Yielding as a Function of Channel Wall Temperature	5-3
5-3	Variation of Channel Response to Internal Loads as a Function of Channel Temperature	5-4
6-1	Schematic Diagram of the Structural Problem	6-4
6-2	Spiral Wire Spacer Contact Points	6-5
6-3	Grid Spacer Contact Points	6-6
6-4	Effect on Fuel Pin Deflection of the Axial Extent of the Thermal Gradient, (2ℓ), Spiral Wire Spacers	6-8
6-5	Effect on Fuel Pin Deflection of the Spiral Wire Lead Distance	6-8
6-6	Effect on Fuel Pin Deflection of the Axial Extent of the Thermal Gradient, (2ℓ), Grid Spacers	6-9
6-7	Effect on Fuel Pin Deflection of the Distance Between Grid Spacers	6-9
6-8	Effect on Fuel Pin Bending Stress of the Axial Extent (2ℓ) of the Thermal Gradient, Spiral Wire Spacers	6-10
6-9	Effect On Fuel Pin Bending Stress Of The Spiral Wire Lead Distance	6-11
6-10	Effect on Fuel Pin Bending Stress of the Axial Extent (2ℓ) of the Thermal Gradient, Grid Spacers	6-12
6-11	Effect on Fuel Pin Bending Stress of the Distance Between Grid Spacers	6-13
6-12	Effect on Adjacent Pin ΔT of Initial Pin Swelling ($\Delta L = 3''$, $F = 0$)	6-14
6-13	Effect on Adjacent Pin ΔT of the Interchannel Flow Sweeping Factor ($\Delta L = 3''$)	6-15
6-14	Effect on Adjacent Pin ΔT of the Axial Extent of Initial Pin Swelling (Flow Sweeping = 1/3)	6-16
6-15	Effect on Adjacent Pin ΔT of the Axial Extent of Initial Pin Swelling (Flow Sweeping = 2/3)	6-17
6-16	Size Distribution	6-24
6-17	Spiral Wire Spacer	6-27
6-18	Minimum Particle Size Which Will Be Trapped In A Wire Wrap Fuel Assembly	6-28

LIST OF ILLUSTRATIONS **VOLUME II (continued)**

<i>Figure</i>	<i>Title</i>	<i>Page</i>
6-19	Minimum Particle Size Which May Be Trapped Between Fuel Pins and the Channel Wall	6-29
6-20	Minimum Particle Size Which May Be Trapped in the Gap Between Adjacent Fuel Pins	6-30
6-21	Grid Spacer	6-31
6-22	Minimum Particle Sizes Which May Be Trapped in a Grid Spacer Fuel Assembly.	6-33
6-23	Minimum Particle Size Which May Be Trapped Between Fuel Pins and the Channel Wall	6-34
6-24	Size of Fuel Particles at Which Coolant Drag Forces Balance Gravitational Forces	6-35
6-25	Computer Model	6-37
6-26	Local Clad Temperature Increase Due to Trapped Fuel Debris	6-39
6-27	Thermal Gradients in Fuel Pin Cladding Due to Fuel Debris	6-40
6-28	Thermal Gradients in Fuel Pin Cladding Due To Fuel Debris	6-41
6-29	Local Channel Wall Temperature Increase Due To Trapped Fuel Debris	6-42
6-30	Predicted Local Temperature Variation Along Channel Wall Caused By Fuel Debris	6-43
6-31	Local Temperature Variation Along Channel Wall Caused By Fuel Debris	6-45
6-32	Thermal Response of Channel Wall and Fuel Cladding to Contact with Molten Fuel	6-46
6-33	Models	6-52
6-34	Hoop Stresses for Hot Pin (18 kw/ft)	6-56
6-35	Hoop Stresses for Cool Pin (11 kw/ft)	6-57
6-36	Pin Temperature Changes After Bonding	6-59
6-37A	Thermal Stress In Fuel	6-60
6-37B	Thermal Stress In Fuel	6-61
6-38	Internal Pressure of Failed Fuel	6-62
6-39	Arch-Restraint Model	6-63
6-40	Out-Of-Pile Burst Test	6-64
7-1	Significance of Sodium Superheat in Loss-of-Flow Accidents	7-3
7-2	Two Phase Flow Patterns for Sodium at 10 PSIA, Thru 14% Quality	7-7
7-3	Two Phase Flow Patterns for Sodium at 10 PSIA, Thru 90% Quality	7-8
7-4	Two Phase Flow Patterns for Sodium at 50 PSIA, Thru 15% Quality	7-9
7-5	Two Phase Flow Patterns for Sodium at 50 PSIA, Thru 90% Quality	7-10
7-6	Sodium Two-Phase Friction Pressure Drop Multiplier vs. Quality At Five Pressures.	7-12
7-7	Void Fraction vs. Quality for Sodium at 10 PSIA	7-13
7-8	Steady State Bundle Hydrodynamics	7-14
7-9	Bubble Collapse Force, Cavitation Water at 1 atm	7-20
7-10	316 SS Cavitation Damage	7-22
7-11	Vapor Bubble Potential Energy	7-24
7-12	Sodium Hammer Pressures	7-32
7-13	Pulse Propagation Velocity and Duration	7-33
7-14	Channel Deflection and Strain	7-34
7-15	Comparison of Sodium Column Kinetic Energy and Channel Strain Energy.	7-36
7-16	Radial Pressure Gradient Within Shock Pulse	7-40
7-17	Test Layout for Water Hammer Effect on Core	7-42
7-18	Thermal Interaction Energy Conversion Efficiency Probability Distribution	7-49
7-19	Sketch of Rolling Diaphragm Piston Seal System	7-59

LIST OF ILLUSTRATIONS VOLUME II (continued)

<i>Figure</i>	<i>Title</i>	<i>Page</i>
7-20	Sketch of Temperature, Pressure, Dilation, and Reactor Power Histories for a 5 ms Transient with 30 gm/cm ² Piston Loading and Atmospheric Initial Pressure	7-59
7-21	Sketch of Temperature, Pressure, Dilation and Reactor Power Histories for a 3 ms Transient with 36 gram/cm ² Piston Loading and 120 PSIA Initial Pressure	7-60
7-22	Pressure — Time Histories for Fixed Volume Runs with Reactor Periods and Maximum Temperatures as Indicated	7-62
7-23	Initial Peak Pressures as a Function of Piston Loading for Runs With 5ms Reactor Period and 350°C Maximum Temperature	7-62
7-24	Pressure History Sketches for Electrically Heated Fixed Volume Experiments with 2% Air Bubble Above and Capsule Full of Water Below	7-64
7-25	Drawing of the Shock Tube Experiment	7-65
7-26	Temperature Dependence of the Peak Pressure from Water Impact upon Aluminum or Silver. Arrows Indicate Shots Where Pressure Rise Was Terminated by Crucible Failure.	7-66
7-27	Pressure History from Water Impact Upon Molten Aluminum At 950°C.	7-68
7-28	Particle Size Distribution of Type 304 Stainless Steel Samples At ~ 2200 To 2300°C Dropped Into Sodium ²²	7-71
7-29	Particle Size Distribution of Type 301 Stainless Steel Samples Dropped Into 250°C Sodium	7-72
7-30	Plot of Weber Number vs Number of Fragments for Various Molten Metals Dropped into Room Temperature Water	7-74
7-31	Heat Flux From Swinging-Arm Experiment With ¼-in. dia Silver Sphere Moving Through Water at 13.9 ft/sec	7-76
7-32	Heat-Transfer Rate for ½-in. dia Tantalum Sphere in Sodium (Sphere speed corrected to 10 ft/sec)	7-77
7-33	Apparatus for Treat Facility Will Evaluate Consequences of Water Column Impacting on Molten UO ₂	7-81
7-34	Piston Autoclave Apparatus	7-82
7-35	Top View of 9-Pin Fuel Cluster For High Energy Meltdown Experiments In Treat In A Sodium Environment	7-83
7-36	Treat S—3 Results	7-84
7-37	Treat S—4 Results	7-85
7-38	Hicks-Menzies Thermodynamic Equilibrium Model	7-90
7-39	Temperature Profile at Time $t > 0$	7-92
7-40	Pressure and Temperature Profiles	7-95
7-41	Thermodynamic Processes of the Model	7-96
7-42	Work Potential vs. Fuel Particle Size	7-106
7-43	Work Potential vs Initial Fuel Temperature	7-107
7-44	Work Potential vs Initial Coolant Temperature	7-107
7-45	Work Potential of Clad Sodium Interaction vs Particle Size	7-108
7-46	Effect of Superheat on Transferred Energy	7-109
7-47	Unrealistic Calculation of Effect of Superheat on Work Energy as Predicted by Original Spherical Geometry Thermodynamic Model	7-110
7-48	"Constant Mass" Thermodynamic Model	7-111
7-49	Work Potential vs Particle Size (Constant Mass Model)	7-112
7-50	Work vs. Coolant/Fuel Mass Ratio	7-113
7-51	Work Potential vs. Initial Fuel Temperature (Constant Mass Model)	7-114

LIST OF ILLUSTRATIONS **VOLUME II (continued)**

<i>Figure</i>	<i>Title</i>	<i>Page</i>
7-52	Work Potential vs. Initial Coolant Temperature (Constant Mass Model)	7-115
7-53	Effect of Superheat on Work Potential (Constant Mass Model)	7-116
7-54	Effect of Superheat At Variable Particle Sizes	7-117
7-55	Work Potential of Clad Sodium Interaction vs. Particle Size (Constant Mass Model)	7-118
7-56	Pressure Wave Shape	7-121
7-57	ANL Bulk Heating Model	7-123
7-58	Pressure Generated by Addition of Heat to Sodium under Spherical Acoustic Constraint	7-126
7-59	Pressure Generated by Addition of Heat to Sodium under Spherical Acoustic Constraint	7-127
7-60	Pressure Generated by Addition of Heat to Sodium under Spherical Acoustic Constraint	7-128
7-61	Pressure Generated by Addition of Heat to Sodium under Spherical Acoustic Constraint	7-129
7-62	Coolant Mass Model	7-130
7-63	Relationship of Heat Generation in Coolant and Fuel Particle Size	7-131
7-64	Acoustic Periods for Water and Sodium	7-132
A-1	Channel Response to Internal Load – Varying Thickness	A-2
A-2	Channel Response to Internal Load – Increased Yield Strength	A-3
A-3	Channel Response to Internal Loads – Varying Yield Strength and Wall Thickness	A-4
A-4	Geometry, Loading and Moment Diagram of Elastic Regime of Channel Response to Internal Loads	A-6
A-5	Loading and Moment Diagrams for the Plastic Regime of Channel Response to Internal Loads	A-8
A-6	Effect Of Corner Radius On Assumed Flat Length	A-13
A-7	Membrane Resistance of Hex Flat	A-16
A-8	Deflection vs Energy	A-20
A-9	Strain vs Energy	A-21
A-10	Pressure vs Energy	A-22
A-11	Deflection vs Energy	A-23
A-12	Strain vs Energy	A-24
A-13	Pressure vs Energy	A-25
A-14	Deflection versus Energy	A-26
A-15	Strain versus Energy	A-27
A-16	Pressure versus Energy	A-28
A-17	Loading and Geometry for External Loads	A-30
A-18	Channel Response to External Loads – Varying Thickness	A-31
A-19	Channel Response to External Loads – Increased Yield Strength	A-32
A-20	Geometry for Rotation at Hinge	A-39
A-21	Deflection vs Energy	A-45
A-22	Strain vs Energy	A-46
A-23	Pressure vs Energy	A-47
A-24	Deflection vs Energy	A-48
A-25	Strain vs Energy	A-49
A-26	Pressure vs Energy	A-50
A-27	Deflection vs Energy	A-51
A-28	Strain vs Energy	A-52
A-29	Pressure vs Energy	A-53
A-30	Continuous Fuel Rod Tube	A-55
A-31	Continuous Fuel Tube	A-56

LIST OF ILLUSTRATIONS **VOLUME II (continued)**

<i>Figure</i>	<i>Title</i>	<i>Page</i>
A-32	Fuel Tube Loading Diagram	A-57
A-33	Fuel Tube Stress Distribution	A-60
A-34	Fuel Tube Loading Diagram	A-61
A-35	Load Diagram for Three Hinge Regime	A-63
A-36	Plastic Deformation Regime – After Developing Third Hinge	A-65
A-37	Fuel Tube Response – Deflection vs Energy – SY = 13,000	A-68
A-38	Fuel Tube Response – Strain vs Energy	A-69
A-39	Fuel Tube Response – Pressure vs Energy	A-70
A-40	Fuel Tube Response – Deflection vs Energy	A-71
A-41	Fuel Tube Response – Strain vs Energy	A-72
A-42	Fuel Tube Response – Pressure vs Energy	A-73
A-43	Fuel Tube Response – Deflection vs Energy	A-74
A-44	Fuel Tube Response – Strain vs Energy	A-75
A-45	Fuel Tube Response – Pressure vs Energy	A-76
B-1	The Measured Gas Pressure Versus Time of Operation. Element LFT (from Reference 1)	B I-5
B-2	Pin B3B Pressure And Temperature History	B I-6
B-3	Radial Power Density Profiles	B I-7
B-4	Temperature Profiles	B I-8
B-5	Sodium Vapor-Vapor Pressure (ATMS) Versus Temperature	B II-2
B-6	Sodium Logged Pin Pressure During Startup	B II-5
B-7	Sodium Vapor Flow From a Sodium Logged Pin During Startup.	B II-6
B-8	Defected Fuel Arch Stress	B III-4
B-9	Cladding Model For Stress Calculations	B III-5
B-10	(None)	B III-6
B-11	Stress in Failed Cladding	B III-7
B-12	Deflection of Failed Clad	B III-8
B-13	Force on Clad Due Internal Pressure	B III-9
B-14	Strength and Deflection Limits of Failed Fuel	B III-10
B-15	Test Pin Configuration	B IV-2
C-1	The WAVE Fuel/Sodium Interaction Model	C-2

LIST OF TABLES VOLUME I

<i>Table</i>	<i>Title</i>	<i>Page</i>
2-1	Response to Signals of Abnormal Core Conditions Based on the Reference LMFBR	
	Design and Operating Philosophy	2-7
2-2	Quantitative Fault Tree Inputs	2-17
2-3	Initiating Event Frequencies, Events Per Year	2-19
3-1	Recommendations Summary	3-2
3-2	Calculated Frequency and Mean-Annual-Cost of Fuel Failure Effects and Abnormal	
	Conditions in a Reactor Design Similar to the GE 1000 MWe LMFBR (GEAP-5678)	3-6
3-3	Reference Case Contribution from Each Initiating Event (\$1000 per year)	3-8
3-4	Influence of LMFBR Fuel Assembly Design Parameters on the Mean-Annual-Cost of	
	Fuel Failure Effect	3-11
3-5	Fault Tree Sensitivity Study	3-12
3-6	Fault Tree Sensitivity Study	3-27
I-1	Appendix I (Discussion of the Tabulated Results)	I-II
I-2	Areas of Safety Investigation: LMFBR Follow-On Studies	I-III
III-1	Qualitative Fault Tree Input for Sodium Boiling	III-XVI
III-2	Qualitative Fault Tree Input for Violent Boiling	III-XVII

LIST OF TABLES VOLUME II

<i>Table</i>	<i>Title</i>	<i>Page</i>
4-1	Case Identification	4-6
4-2	Conservative Failure Times	4-7
4-3	Failure Times Assuming Damage Accumulation	4-8
4-4	Optimistic Failure Times	4-9
4-5	Partial Summary of Task C Experimental Results Indicating Functional Relationships	
	Involved in a Fuel Volume Melt-Clad Breach Failure Criterion	4-35
6-1	Case Characteristics	6-4
6-2	Pin to Spacer Contact Force in the Direction of the Deflection (pounds), $\Delta T = 100^\circ F$	6-18
6-3	Safety Limits Based on Mechanical Loading	6-19
7-1	Radial Velocity Due to Sodium Hammer	7-38
7-2	Pressure Drop in Cross Flow	7-39
7-3	Summary of Work-Energy and Thermal-Mechanical Conversion Estimates	7-47
7-4	Summary of Damage Thresholds and Their Probabilities of Occurrence	7-48
7-5	Pressure and Void Data for Runs at Various Initial Pressures and Ambient Initial	
	Temperature	7-61
7-6	124 Gram Loading 5 ms Period	7-69
7-7	Results of Dropping Materials Into $250^\circ C$ Sodium	7-73
7-8	Results on Dropping Various Materials Into Water	7-73
7-9	Fragmentation of Liquid Gallium Dropped Into $30^\circ C$ Water	7-74
7-10	Summary of Results of Piston Autoclave Experiments	7-84
7-11	Comparison of Thermodynamic Equilibrium Calculations	7-89
7-12	Wave Energy and Total Energy Versus Time	7-122

**LIST OF TABLES
VOLUME II (continued)**

<i>Table</i>	<i>Title</i>	<i>Page</i>
A-1	Parameter Studies (Hexagonal/Fuel Pin)	A-1
B-1	(None)	B I-6
B-2	Sodium Vapor Flow From Fuel Pin	B II-4
C-1	Solutions of the Fuel/Sodium Interaction Problem	C-12

ABSTRACT

*A comprehensive assessment of safety-related/core design information needs was conducted. * A basic hypothesis was that core performance under off-design and accident conditions will affect the choice of core design parameters. The economic consequences of fuel failure effects and random accidents were compared with fault tree results based on a "mean-annual-cost" concept.*

It is concluded that fuel failure effects dominate the expected cost penalties. Principal development recommendations are for integral tests of: 1) fuel failure propagation and 2) loss-of-cooling consequences.

* Excluding DBA accommodation and other licensing considerations.

1.0 INTRODUCTION

1.1 SCOPE OF WORK

This report covers the contract year 1970 (October 1, 1969--September 30, 1970) effort on PA-10 Tank G-6.3.4. The objective of this work is to define core design information needs and related priorities for development work in the areas of fuel element failure, sodium boiling and safety limits. The study is focused on the implications of fuel failure and failure propagation events with regard to core safety and economic performance, excluding DBA accommodation and other licensing considerations. Required development activities identified in this study are described, and related to on-going programs where such exist.

The study was carried out by first conducting a broad survey of LMFBR hazards and analysis, safety development and core design efforts. Priority problem areas were determined using a fault tree type of analysis. When data or analyses were lacking in an area, the area was analyzed to determine design parameter sensitivities. The understanding gained of core performance in off-design conditions was then used to outline the further information needed for core design. Recommendations for further specific development were then made where appropriate.

This report consists of two volumes. Volume I contains recommendations and fault tree analyses. Volume II is an in-depth study of specific phenomena identified by the fault trees as having a major impact on reactor safety and core design.

1.2 PRIORITY CONCLUSIONS AND RECOMMENDATIONS

The principal result of this study is a list of high priority research and development needs. First, it is necessary to establish the credibility of hypothesized mechanisms of accident progression. The most important mechanisms needing further safety research, in order of priority, are:

- clad/fuel sodium thermal interaction;
- fuel debris flow blockage;
- fuel-sodium reaction product swelling; and
- local sodium boiling stability.

Second, it is necessary to establish the impact of core design changes on potential thresholds. The highest priority areas needing design development are:

- clad failure from spacer contact forces;
- fission product cladding attack; and
- channel containment capability.

The priority items are the major uncertainties of the dominant fault tree accident chain. The first group of items relates to the uncertainty of the occurrence and/or significance of potentially damaging phenomena. The second group of items relates to the ability of the core design to accommodate the effects of these phenomena. The dominant fault tree accident chain involves the propagation of fuel failures leading to local sodium boiling, clad and fuel meltdown, and bundle-to-bundle propagation as shown in Figure 1-1. It must be emphasized that the accident chain does not progress upward if any propagation mechanism (or both of two parallel propagation mechanisms) does not occur under reactor conditions. This is, in fact, the probable case. Fuel swelling and debris fallout have not been observed to a serious level in operating reactors. The experience is not sufficient, however, to permit closing the books on these possibilities. It is considered prudent to research these areas to attain acceptable confidence in the safety and operability of the LMFBR.

The following highlights the priority information needs and development recommendations as viewed at this time. The conclusions are obviously subject to revision and could be placed on a more firm basis through additional investigation. Many significant areas should be the subject of analytical study in a continuing effort which takes an integrated view of fuel failure propagation and sodium boiling. Other recommendations of lower priority, are found in sections 3 through 7 in conjunction with discussion of specific topics.

1.3 PRIORITY AREAS FOR SAFETY RESEARCH

Safety research is needed to establish the level of credibility of all potentially damaging mechanisms. The following discussions are ordered to show the qualitative priority ranking. This priority ranking was established by considering the following:

- The fault tree results showing the potential economic penalties if certain damaging phenomena are found to occur.

Figure 1-1 Dominant Accident Chain

- The relative speed of propagation mechanisms with respect to possible safety system response.
- The influence on licensing efforts to assure the safety of the LMFBR in concept and practice.

1.3.1 Clad/Fuel Sodium Thermal Interaction

The general phenomenon of clad/fuel-sodium thermal interaction (violent boiling) is potentially capable of generating damaging pressures, particularly if the clad and fuel are molten. Three accident categories are considered:

- local clad and fuel meltdown due to local flow blockage;
- whole bundle meltdown due to loss of flow, bundle blockage or progressive local blockage; and
- power transients.

The first two are on the dominant accident chain described above. It must be determined if the pressures generated are sufficient to cause structural damage to:

- adjacent fuel pins;
- channel walls of the affected and adjacent bundles;
- control rod guide tubes;
- reactor vessel components; and
- system, building and containment structures.

Current and planned modeling efforts appear to be adequate for scoping studies but not for prediction of accident consequences.

The recommendation is for prototypic in-pile loss-of-flow tests to measure the pressures and observe the resulting structural damage effects directly.

The PA-10, Task C-2 tests in GETR are directly applicable to this highest priority safety research recommendation. The schedule of these tests seems too long compared with FFTF and Demonstration Plant needs, particularly the multi-pin tests permitting better simulation of core structures. Proposals for such tests in TREAT (by ANL) raise serious questions about the pre-loss-of-flow conditions as well as a limited test duration. Tests in reactors such as GETR or those proposed for SEFOR eliminate these problems. It is recommended that the program for in-pile loss-of-flow tests be reviewed with the aim of expediting and expanding the current Task C-2 program to reflect its highest priority.

1.3.2 Fuel Debris Flow Blockage

Fuel debris can potentially cause rapid propagation of failures across a fuel bundle. When a pin fails by bursting, fuel debris may be ejected and the particles may be transported to adjacent flow channels where they may be caught on spacers. In this situation, the adjacent fuel pins could fail due to the restricted cooling. This could lead in turn to more pin failures and so on. This condition could progress at almost the thermal time constant of the cladding temperature limit (~ 1 second) from row to row. A serious local blockage (causing sodium boiling) could then result in a time period short in comparison to the capabilities of detection instrumentation. It is this prospect which warrants classification of fuel debris flow blockage as a high-priority research area.

Certain key phenomenological questions arise when considering fuel debris flow blockage.

- Do fuel pins fail in the "burst" mode such that fuel debris is released to the coolant streams?
- Are fuel particles transported to adjacent flow channels (others beyond those surrounding the failed fuel pin)?
- Are fuel particles trapped next to fuel pins in sufficient mass to cause damaging undercooling?

Satisfactory (negative) answers to these questions could not be found during this study, forcing a conservative assumption that these phenomena may indeed occur.

Several design parameters have a potential bearing on the question of fuel debris flow blockage:

- fission gas accommodation as it affects the possible driving pressure causing fuel expulsion on failure;
- fuel form – annular pellets may prevent central void pressures higher than plenum or vent pressure;
- fuel form – powdered fuel may release different quantities and sizes of fuel particles than pellet fuel; and
- pin spacing (pitch) and spacer design will affect the way fuel particles are moved laterally to adjacent pins and the available trapping sites.

Selection of these design parameters for LMFBF's may be affected if fuel debris flow blockage is a real problem.

An analysis of the thermal effects of local debris blockage is recommended to accurately determine the size of blockage capable of causing serious undercooling.

Out-of-pile burst test simulations are recommended to scope the questions of debris ejection, transport, and trapping in a fuel bundle. A transparent fuel assembly with one real fuel pin subjected to a burst failure might permit visualization of the fuel debris flow blockage. This perhaps could be done in an ANL transparent test section previously used for gas release studies.

The planned program at ORNL to experimentally measure the thermal perturbations of local blockages seems to satisfy the need for empirical confirmation of the above recommended analysis.

An in-pile test program is considered vital to this subject. Already planned as part of the PA-10, Task B, Series 10 tests is a multi-pin test of failure propagation. This study would perhaps suggest a burst test at

power of an edge pin (7-rod array) with fission gas pressure monitors (or other techniques) on all other pins to indicate their failure times (if they fail). An early test of this type would provide invaluable guidance for defining a continuing program to understand the propagation potential of fuel debris flow blockage.

1.3.3 Fuel-Sodium Reaction Product Swelling*

The observation has been made of severe fuel-sodium reaction product swelling in PA-10, Task B tests of defected fuel pins. In contrast, no mention is made of this problem by operators of existing oxide-fueled sodium-cooled fast reactors (BR-5, DFR, Rapsodie). This anomaly must be resolved. Assuming the worst, that swelling occurs in the reactor, this swelling is a dominant mechanism leading to fuel failure propagation and subsequent power cost penalties. Unlike propagation due to debris blockage, propagation due to swelling is expected to be relatively slow but capable nevertheless of leading to the more serious problems of local sodium boiling.

The key research requirements are to determine if swelling will occur in a reactor environment and, if so, how it is affected by fuel O/M ratio, which is considered the dominant design variable affecting swelling.

The recommended research is already planned under PA-10, Task B, in-pile flowing sodium tests of failed fuel. The key to resolving this question is the success of capsule designs capable of controlling the sodium-oxygen content as it would be in-reactor.

Uncertain (probably high) oxygen content in capsules B9A and E1J-B2 may have been solely responsible for the observed fuel swelling. Resolution of this uncertainty is required before proceeding further with research into fuel-sodium reaction product swelling.

1.3.4 Local Sodium Boiling Stability

A local flow blockage which degrades thermal conditions to the point of local boiling becomes a more serious and potentially more expensive incident if local sodium boiling is unstable. Thermal-hydraulic instability leads to the entire fuel bundle boiling, which causes fuel bundle meltdown at full power. Conversely, if local boiling is stable, sufficient time may be gained to detect the situation and shut down. The boiling noise may itself provide an additional signal for detection of the situation.

During this study, no record of discussions of local boiling stability were found, forcing a conservative assumption of instability. The recommended test program appears satisfied at ORNL under the fuel failure propagation program. At the end of measurements of the thermal perturbations due to local blockages, the flow is to be decreased to initiate local boiling at the simulated partial blockage. Power input will be continued to the end of heater lifetime. This type of test can potentially answer the question of local boiling stability. Much is dependent upon heater lifetime, which is very uncertain in respect to an unknown time constant of boiling instability, and upon the importance of testing at prototypic pressures. It seems prudent to plan for expansion of the boiling portion of the ORNL test program based on the priorities of this study.

1.4 PRIORITY AREAS FOR DESIGN DEVELOPMENT

Design development is needed to establish the impact of core design changes on potential failure thresholds. The following discussions are ordered to show the qualitative priority ranking.

*Analysis (P.A. 53) completed since this work was done indicates that fuel-sodium reaction product swelling is limited by oxygen availability. A small clad defect apparently will admit sufficient sodium to form the reaction product and cause swelling. As soon as a significant size crack is formed (~ 15 mils), however, oxygen diffusion from the fuel to the sodium is sufficient to deplete the oxygen supply and to stop the reaction. Fuel swelling is thus limited and does not appear to be a serious problem.

1.4.1 Clad Failure From Spacer Contact Forces

The potential impact of spacer contact forces on clad survival is noted twice on the dominant accident chain. Fuel failure propagation within a fuel bundle may possibly be a strong function of the spacer design as it affects the clad-spacer contact forces. The principal mechanism causing excessive clad-spacer contact forces is the thermal gradient due to local blockages or fuel swelling (both discussed above).

The second area noted is fuel failure in adjacent bundles due to excessive deformation of the channel wall and resultant force on the spacers and fuel pins.

The clad failure threshold due to a spacer contact force could not be established from the state-of-the-art.

A general transient mechanical clad failure model is needed, including the effects of spacer contact forces. Empirical confirmation of the failure threshold also is warranted.

The nature of the failure can be as important as the failure itself, requiring the failure models to portray the details of the post-failure condition. In particular, if failures occur such that fuel is ejected, the result is far more serious than if the clad only cracks. Several design features may interact to make each design behave differently.

The combination of a sealed pin under pressure and excess spacer contact force may fail in the "burst" mode, for example, while a vented pin probably would not. Sufficient out-of-pile simulations of clad failures are needed to permit correlation of these effects.

1.4.2 Fission Product Cladding Attack

Significant corrosion on the inside of the cladding has been observed. This attack is attributed to a grain boundary corrosion by fission products. The corrosion rate is believed to be very temperature-sensitive above normal operating temperatures. Based on extrapolation of the limited data to high temperatures, this fission product cladding attack was found to significantly shorten the lifetime of an overheated fuel pin. Fuel failure propagation caused by local undercooling due to fuel debris or fuel swelling is more likely if this corrosion mechanism occurs. Particularly affected is the reference vented fuel design because of its otherwise relative insensitivity to overtemperature transients (lack of a primary stress component, fission gas pressure).

There is a need to correlate the rate of fission product cladding attack at high temperatures (to 1800°F). If confirmed to be significant and if resistance to overtemperature conditions is considered necessary, development aimed at inhibiting this cladding attack would be justified. With the current conservative view of failure frequencies and propagation phenomenon, it appears advisable to include consideration of overtemperature resistance in fuel design criteria.

It is recommended that in-pile tests be conducted of short-time (100 to 1000 hours) fission product cladding attack in the range of 1300 - 1800°F. The principal variable is expected to be the burnup at which the over-temperature condition is imposed. These tests could be the extension of P.A. 10 Tasks B and E tests to higher temperatures.

Thermodynamic studies aimed at isolating the corrosion agent and specifying an inhibitor should be pursued.

1.4.3 Channel Containment Capability

Analyses conducted during this study emphasized that a hexagonal channel has limited ability to contain pressures significantly above design conditions. The most likely source of damaging pressures is violent boiling after clad/fuel meltdown, but the magnitude remains very uncertain. Also, the probability of experiencing a meltdown is considered very low. Primarily for these reasons, it is not known if the hexagonal

channel of an LMFBR will be expected to contain any significant pressures (steady state or transient) above normal operating conditions.

There is a need to study and resolve the question of the channel containment criteria.

It is recommended that a parallel effort be aimed at correlating the containment capabilities of hexagonal channels. In a core mockup, a series of tests could be run simulating the possible channel pressurization accidents:

- pneumatic tests to simulate vapor expansion and fission gas release;
- hydraulic tests to simulate sodium hammer; and
- explosive tests to simulate violent boiling.

Because the measure of containment will be the effect on adjacent control rods and fuel pins in adjacent channels, these should be simulated. The deformations and strains of control rods and fuel pins should be measured as well as the effect on the pressurized and adjacent channels.

1.5 RECOMMENDATIONS FOR FURTHER DEFINITION STUDIES

It is believed that the fault tree concept used in organizing this study is a useful and necessary element to gain a sufficient overview of the interaction of core design and safety needs. It is recommended that studies of this type continue and be expanded. In particular, expansion into the following areas is recommended:

- The quantitative fault tree analysis should be extended to include explicit mechanisms of accident progression.
- An extensive assessment of initiating event frequency should be conducted.
- The speed and timing of fuel failure propagation events should be evaluated with relation to safety instrumentation response and operating philosophy.
- A cost benefit study of research and development programs should be conducted based on the mean-annual-cost results from the fault tree studies.

The following identifies the key uncertainties which appear amenable to further analysis. Scoping and sensitivity studies of some of these areas would serve to reduce the present uncertainties concerning their probability of occurrence and significance to the fault tree.

- Clad Failure Thresholds
 - due to spacer contact
 - due to vapor bubble collapse
- Local Boiling Stability
- Failure Propagation Potential Due to Local Fuel/Clad Sodium Thermal Interaction
- Limitation of Fuel/Clad-Sodium Thermal Interaction Work Potential Due to Presence Of Gas Or Vapor
- Structural Restraint of Fuel/Clad-Sodium Thermal Interaction
- Modeling Of Energy Deposition On Fuel Pins And Channels By Pressure Pulses
- Bound The Degree Of Fuel Debris Blockage Considered Serious.

1.6 STUDY METHODS AND APPROACH

The starting point for this study was a review of the recently completed LMFBR follow-on studies (References 17-21), in particular, the design and safety philosophy of each major contractor. This review is summarized in Appendix I. The purpose of the review was to determine how these philosophies affect the contractors' judgment of the importance of various sodium boiling phenomena and fuel failure effects. The results of the review emphasized the current consensus on LMFBR problem areas. In general, there is concern for every aspect of core behavior under off-design conditions. No specific areas are noted as being understood sufficiently well to not require further research. A general shift in emphasis is noted to the more likely but less consequential hazards such as fuel failure propagation and local flow blockage (compared to the earlier [mid-1960's] emphasis on power transient DBA).

There is a general understanding of the design implications of accidents which initially affect the whole core and which may lead to core disassembly. These accidents (reactivity insertions and whole-core flow reductions) have been the target of numerous analytical studies and basic research in the 1960's.

Therefore, the major emphasis of this study was placed on the design implications of fuel failure effects and local flow blockage. The development of understanding of core behavior under local fuel failure and blockage conditions is relatively in its infancy, and a study of this subject, with the object of developing information needs, is thus most appropriate. This study is not directed specifically toward the DBA and its possible licensing implication. Therefore, recommendations for further studies and testing in this area have not been made.

The 1000 MWe design studies by General Electric Company^{1,7} will form the basis for evaluating core design information needs. These studies anticipate an LMFBR design philosophy in the 1980's which permits minimization of design features specifically incorporated for the blast effects of core disassembly incidents. This philosophy is based on development of reliable detection instruments and shutdown systems which will reduce the probability of a disassembly incident to a sufficiently low value that blast effects need not be specially provided for. It is also dependent on the development of quantitative methods to show that the product of the probability and consequences of any accident is acceptable. This typifies the probabilistic approach to safety. This philosophy tends to direct the major attention of safety research toward the more probable but individually less consequential accidents.

An original hypothesis underlying this study was that the behavior of the core under fuel failure and/or accident conditions might influence the choice of core design parameters. It is not obvious, even after these studies, to what degree the hypothesis is true. What has been developed is a better understanding of the possibilities for design influence from safety related research and development. It is clear from this study that, based on conservative predictions of off-design core performance, it is prudent and necessary to gain a fundamental knowledge of off-design core performance to allow the reactor designer to optimize the core design and operation of the LMFBR for minimum cost power. The safety implications of such research will probably affect the containment and protective system design to a greater extent than the core design. The direct economic implications are far more numerous and in some cases more uncertain than the safety implications. For this reason, the economics of fuel failures and/or accident conditions have been used as a basis for evaluating information requirements and development priorities. This economic measure of the consequences of reactor faults permits a relative evaluation of all faults, not just those posing a public hazard. As noted by Walke,³ the major implication of fuel failures in a BWR is in costs—rather than in public safety. The same effect is expected in a commercial LMFBR.

A fault tree type of analysis has been used to organize this study. A fault tree requires the inclusion of all relevant factors and forces the quantifying of the present state of knowledge. A fault tree (or a similar organizational tool) appears not only useful but necessary to carefully delineate the current state of knowledge and to logically assign priorities for future work. The results of the fault tree analysis are the probable frequencies of certain undesirable end states. By estimating the cost associated with each end state, the results are also expressed as the "mean annual cost."

The absolute level of the calculated mean-annual costs in this report is quite high because of the use of conservative transition probabilities. A 90% confidence level was used, implying that the real probability of a mechanism occurring under the given situation has a 90% chance to be less than the assigned probability. With essentially unknown phenomena, this requires assigning a unity probability to their occurrence, even though intuition suggests the mechanism might not occur at all. The use of probabilities to describe the occurrence of transition mechanisms is thus seen to be a quantification of engineering judgment and should not be confused with probabilities as used in statistics. The absolute cost predictions are thus a function of the state-of-the-art and will decrease on a learning curve determined by safety research results which improve our understanding of basic mechanisms.

The fault tree analysis has pointed to the accident chains which appear to contribute most substantially to the safety-related mean-annual cost. The interpretation of these results can be approached in two basic ways. In the first instance, the sensitivity of the results to the probabilities of certain mechanisms occurring will demonstrate the need to increase the basic understanding of these mechanisms. Safety research is then needed to increase our knowledge of these areas. In the second case, after a basic understanding of the phenomenon is gained, the sensitivity of the results to design parameters will suggest confirming development tests of alternate designs.

2.0 FAULT TREE FORMAT FOR ASSESSING DESIGN NEEDS

Given that a fuel pin failure or any other abnormal condition occurs somewhere in the reactor, three questions are relevant:

1. Does the failure induced by the initial event extend to the adjacent pins, to the adjacent fuel assemblies, or to still more remote assemblies?
2. Do the released fission products and fuel from both the initial failure and the induced failures add significantly to the radioactive contamination of the reactor systems?
3. Must the reactor be shut down to remove the failed fuel?

If the answer to all three of these questions is negative then the occurrence of the fuel pin failure or other abnormality is insignificant.

However, the answers to these questions are interdependent. If the failure propagates to adjacent pins, there is probably a point beyond which the added radioactive contamination is significant. Even if the contamination is insignificant, there is probably a point beyond which the assembly is structurally unsound and the difficulty of removing it is significantly increased. If the reactor is shutdown immediately to remove the failed fuel, then the probabilities of failure propagation and of excessive contamination are reduced.

It is apparent, then, that trade-offs exist in the fuel design features, the plant design features, and the plant operating philosophy to minimize the net effect of failures and abnormal conditions. For instance, an operating philosophy which requires the immediate detection, location, and removal of each failed fuel pin will probably require complex instrumentation and will probably incur operating cost penalties due to frequent reactor shutdowns and to reduced average fuel burnup. However, this operating philosophy will allow cost savings in the provisions for handling highly contaminated cover gas, contaminated primary sodium, and contaminated equipment.

In addition, the design of the fuel assembly will affect both the frequency of initial pin failures and the probabilities of failure propagation. However, the incentive to make compromises on the fuel assembly design will depend upon the cost savings and safety benefits to be realized in the remainder of the reactor system.

An orderly framework is required in which to logically compare the many effects involved (fuel design and performance characteristics, system design and performance, and reactor operating philosophy). One such framework is a fault tree.

The material which follows illustrates the development and use of these trees. In particular, fault trees are developed for various hypothetical initiating events which lead to other possible end states. Probabilities for the paths between the initiating events and the end states are estimated. These probabilities, together with estimates of the frequency of occurrence of the initiating events and estimates of the cost of the end states, allow the computation of the "mean-annual-cost." By examination of these different end state costs, and by testing the sensitivity of these costs to different assumptions of probabilities in the branches of the trees, it is possible to identify those parameters which have the greatest effect on safety-related operating costs (the mean-annual-costs). These safety-related operating costs, in turn, are related to the design parameters which can be adjusted to alter the costs. Alteration of a design parameter also affects the fuel cycle costs, and the relative importance of that design parameter can be determined by a comparison of its effects on both the safety-related operating cost and the fuel cycle cost.

This section shows the development of the fault tree logic. This logic is summarized following a step-by-step description of the components. The fault tree results are discussed in Section 3, Conclusions and Recommendations. The detailed fault trees and detailed discussions are found in Appendices II, III and IV.

2.1 SELECTION OF FAULT TREES

A rigorous introduction to the concept of a fault tree has been provided by Schroeder¹. Basically, the fault tree is a graphic representation of all of the events which lead to a given undesirable end state. Each undesirable end state, say reactor shutdown to remove a fuel bundle which contains failed fuel pins, constitutes the tree trunk and the events leading to the undesirable end states constitute the tree roots.

After acceptable fault trees are drawn, the statistical parameters which describe each event in the roots of the trees can be estimated. Then, for a given reactor design, the fault trees can be used to compare the frequency of the different final undesirable end states. Given an estimate of the cost-per-event for the final states, a mean-annual-cost parameter can be calculated. The "best" combination of design and performance characteristics and plant operating philosophy

will be that which minimizes the mean-annual-cost. This approach is similar to that of Mean-Annual-Severity described by Doron and Albers²

This economic measure of the consequences of reactor faults permits a relative evaluation of all faults, not just those posing a public hazard. As noted by Walke³, the major implication of fuel failures in a BWR is in costs rather than public safety. The same effect is expected in a commercial LMFBR.

Because of the scarcity of data on fuel failures and the induced effects the calculated results will contain a high degree of uncertainty. However, by exercising the procedure outlined above it is possible to identify those areas where added information is most vital.

Extensive and detailed fault trees have been completed for twelve arbitrarily defined end states of the reactor. It was intended that the twelve states would effectively characterize the full spectrum of final states which result from fuel failure effects. Included are reactor shutdowns to remove fuel bundles which contain damaged fuel and either damaged or undamaged channel walls. Also included is continued reactor operation with different levels of fuel damage.

Similarly, twelve initiating events were arbitrarily defined, which apply to each of the twelve fault trees. The twelve events are intended to characterize the full spectrum of initiating events. The initiating events considered were "normal" fuel pin failures, channel inlet flow blockages, and large reactivity transients. Fuel pin failures were divided into three categories: (1) Crack Failure — fission gas is released through the clad and sodium may enter the pin, but no washout of the fuel and/or solid fission products occurs, (2) Large Area Failure — a significant amount of sodium contacts the fuel; (3) Burst Failure — fuel is rapidly expelled from the pin into the coolant. Preliminary values of frequency of occurrence were assigned to each of these initiating events.

The fault trees show nearly all of the intermediate steps required for the reactor to progress from one of the initiating events to one of the final states. Preliminary values were assigned to each of the probabilities of progressing from one state to the next successive state, based on the state-of-the-art as determined through an extensive literature survey.

All of the preliminary estimates were based on the reactor design and operating philosophy presented by GE in the Task II report of the 1000 MWe LMFBR Follow-On Study⁴. These estimates were combined, using the fault trees, to establish the frequency and to estimate the mean-annual-cost of each of the final states of the reactor.

The results were used to identify the most important sequences of events through which fuel failures affect the cost of the reactor and of reactor operation. Effort was then devoted to examining, more carefully, each of these important events, and identifying the development effort necessary to minimize its probability of occurrence.

Numerous examples of fault trees are available. As far as can be determined, they are all qualitative logic trees, and have not been quantitatively exercised. Of principal interest are the following:

- 1) Combustion Engineering⁵
 - shows the relationship of instrumentation response to fault progression
 - lacks detail necessary for quantitative analysis
- 2) FFTF⁶
 - detailed fault tree for the design basis accident (DBA), including separate fault tree for safety system
 - lacks fuel failure propagation detail
- 3) FFTF⁷
 - a logic fault tree showing the relationship to on-going experimental programs
- 4) ANL⁸
 - limited to pin-to-pin failure propagation under near normal operation.

Since no directly applicable fault trees for fuel failure effects are available, preliminary fault trees have been developed under this study.

2.2 FAULT TREE FINAL STATES

The twelve end states of the fault trees are intended to include all possible end states of the reactor which result after either random fuel failure has occurred, or loss of flow to one or more channels has occurred, or a large reactivity transient has occurred. However, the number of possible end states is infinite if all possible degrees of damage are considered. Consequently, an attempt was made to represent these continuous spectra of final states by a manageable number of discrete states. The criterion for defining one of the discrete states is that the range of events in that state have roughly similar costs-per-event and likelihood of detection. The chosen states are listed below with brief definitions where appropriate. The costs-per-event estimated for each end state is also presented. These costs are based on the reference reactor design and operating philosophy⁴. This list of states is repeated for easy reference on a pull-out page at the back of this volume.

S1 — The reactor is contaminated by fission gas leakage from a bundle with one failed fuel pin. This state will exist if either the reactor operators cannot locate the bundle or the operating philosophy dictates leaving the bundle in for continued irradiation.

S2 — The reactor is contaminated by fission gas leakage from a bundle with a few failed pins. ("Few" is an arbitrary number which is greater than one and less than "many", say two through seven.)

S3 — The reactor is contaminated by fission gas leakage from a bundle with many failed pins.

The cost-per-event of being in states S1, S2 or S3 was taken to be zero. The reactor, in this case, is designed to operate with short-holdup vented fuel pins. Therefore the consequences of added contamination from fission gas leakage through failed cladding is negligible.

S4 — The reactor is contaminated by fission product and fuel release from a bundle with one failed pin. (This failure type covers the range from small cracks which have opened up enough to admit sodium, to massive burst failures which completely disintegrate the pin.)

S5 — The reactor is contaminated by fission product and fuel release from a bundle with a few failed pins.

The cost-per-event of being in states S4 and S5 was taken to be zero. Although the buildup of fission products on the pumps and heat exchangers is serious enough to require facilities for remote handling and cleaning, such facilities are also required due to radioactive corrosion products (notably Mn-54). This conclusion was stated from a study of the FFTF by Gee⁹.

S6 — The reactor is contaminated by fission product and fuel release from a bundle with many failed pins.

Again, the consequences of the contamination are probably small. However, a significant loss of fuel from one bundle could have a slight effect on the reactivity life of the core. In the reference reactor, the core is expected to average 790 days (at 0.8 load factor) per 234 bundles. Therefore, if ten percent of the fuel was lost from one bundle, it might shorten the life of the core by 0.34 days. If the cost of replacement power is 2.2 mills/kwh then the penalty is:

\$18,500/event.

S7 — The reactor is shutdown to remove one bundle whose channel wall is undamaged. (This state will exist whenever the instrumentation in the reactor detects and locates a fuel bundle in which the number and type of fuel pin failures, or the flow rate, or the outlet temperature are not within a preset allowable range. Further, the operators must find that they can remove the bundle using the normal refueling procedures and equipment.)

It has been estimated⁴ that this can be accomplished in approximately 62 hours. The cost of the replacement power (at 2.2 mills/kwh) is then \$140,000/event. If this event occurred six months before the bundle was to be replaced normally, then the penalty for the reduced burnup is approximately \$7,000/event. The total cost is:

\$147,000/event

- S8 — The reactor is shutdown to remove a few bundles whose channel walls are undamaged.
- S9 — The reactor is shutdown to remove many bundles whose channel walls are undamaged.

The penalties are of the same nature as for S7. Let few = 2 and many = 7:

State	S8	S9
No. of Bundles	2	7
Refueling Time, hrs.	63	65
Down-Time Cost, \$	142,000	147,000
Lost Burnup Cost, \$	14,000	49,000
Cost Per Event, \$	<u>156,000</u>	<u>196,000</u>

- S10 — The reactor is shutdown to remove one bundle whose channel wall is damaged. (The signals from the instrumentation in this case may not be very different from those in case S7. However, the operators find that they cannot remove the bundle using the normal refueling procedures and equipment.)
- S11 — The reactor is shutdown to remove a few bundles whose channel walls are damaged.
- S12 — The reactor is shutdown to remove many bundles whose channel walls are damaged. (This state covers the range from seven bundles damaged through disassembly of the core.)

For these states the time required to replace the fuel is longer because the normal refueling procedures and equipment are inadequate. In addition, extra time is required to inspect the remainder of the core, the core structure, and the instrumentation. The following judgments were used:

	S10	S11	S12
No. of Bundles	1	2	7
Refueling Time, hrs.	68	72	81
Inspection Time	1 day	1 week	1 month
Total Down-time, hrs.	92	240	811
Downtime Cost, \$	207,000	530,000	1,790,000
Lost Burnup Cost, \$	7,000	14,000	49,000
Cost Per Event, \$	<u>214,000</u>	<u>544,000</u>	<u>1,839,000</u>

The fault trees can be used to calculate the average frequency of occurrence of each end state. The product of the above costs per event and the frequency yield an average cost per unit time.

2.3 FAULT TREE INITIATING EVENTS

The twelve fault trees resulting in the twelve final states described above share common initiating events. For the purposes of the fault tree an initiating event is one whose frequency of occurrence can be estimated, but the mechanisms which cause it to occur are not defined on the fault tree. Each of the events chosen as an initiating event can also appear elsewhere in the fault tree as one result of another initiating event. However, the frequency of that event, considered as a result, is a distinct quantity unrelated to its frequency as an initiating event.

The chosen initiating events are intended to include all possible randomly occurring initiating events. However, if all of the possible types, sizes, and locations of random fuel failures, and all of the possible levels of randomly occurring flow reductions, and all of the possible levels and durations of reactivity transients are considered, then the number of initiating events is infinite. Consequently, these continuous spectra of initiating events has been replaced by a manageable number of discrete events. The criterion for defining the range of one of the discrete initiating events is that the events represented have roughly similar immediate consequences and likelihood of detection.

To limit the size of the fault tree it was necessary to make many simplifying assumptions. For instance, the counting system for either the number of failed pins per bundle or the number of bundles with failed pins has only three values: one, few (2-6), or many (>6). In addition, the characterization of a channel wall (for the purpose of estimating the cost per event of removing a fuel bundle) has only two modes: damaged or undamaged.

The chosen randomly occurring initiating events are listed below, with brief definitions: This list is repeated for easy reference on a pull-out page at the end of this volume.

- I-1 — Crack type clad failure in one fuel pin. (This failure type covers the range from the smallest failures, say faulty end-plug welds, to the largest failures, say long hair-line cracks, which release fission gas from the pin and allow sodium to enter, but prevent washout of the fuel and/or solid fission products.)
- I-2 — Large area clad failure in one fuel pin. (This failure type includes any clad failure which allows sodium to flow into the fuel pin, and which allows fuel and non-gaseous fission products to leave the fuel pin, but does not result in the rapid expulsion of hot fuel into the sodium. Failure debris, such as fuel grains and pieces of cladding, may enter the coolant stream.)
- I-3 — Burst type clad failure in one fuel pin. (This failure type includes any pin failure which results in the rapid expulsion of fuel from the pin into the sodium. Failure debris is rapidly distributed in the channel.)

Note: These fuel pin failures are considered spontaneous in that they occur while the fuel is operating within the design limits of the fuel and clad.

- I-4 — Inlet flow blockage which causes only a reduction of flow through one fuel channel.
- I-5 — Inlet flow blockage which causes local boiling in one fuel channel but does not immediately lead to channel voiding.
- I-6 — Inlet flow blockage which causes immediate voiding of one fuel channel. (These three, I-4, I-5, and I-6 include only randomly occurring accidents such as Tetralin in the coolant or zirconium plates in the inlet plenum. Flow reductions due to fuel failure debris are *not* included.)
- I-7 — Loss of cooling to a few fuel channels (simultaneous inlet blockage of a few channels).
- I-8 — Loss of cooling to many fuel channels. (Includes pipe ruptures, loss of pumps, and simultaneous inlet blockage of many channels.)
- I-9 — Reactivity addition which causes crack type clad failures in many fuel bundles but does not damage the channel walls. (This event includes only randomly occurring accidents such as control rod blowout. Reactivity addition due to the voiding which follows channel blockage is *not* included.)
- I-10 — Reactivity addition which causes large area failures in many fuel bundles but does not damage the channel walls.
- I-11 — Reactivity addition which causes burst failures in many fuel bundles but does not damage the channel walls.

- 1-12 — Reactivity addition which causes channel wall damage in many fuel bundles. (The damage to the channel walls is such that the normal fuel handling procedures and equipment cannot be used. This event probably includes the DBA.)

2.4 RELATIONSHIP TO LMFBR DESIGN PHILOSOPHY

The reactor design and operating philosophy has a very significant effect on core safety. The philosophy dictates the type and quantity of instrumentation which will be used to detect abnormal core conditions. The formulation of the responses to given signals of abnormality also relies heavily on this parameter. In a similar fashion the fault tree analysis relied on a certain design and operating philosophy. To carry out this analysis it was necessary to evaluate the probabilities associated with detecting, locating, and removing various types of fuel failures or abnormal conditions. Given that the abnormality is detected, an assessment must be made as to whether or not the reactor is to be shut down to correct the situation. This assessment depends on the reactor operating philosophy. As indicated previously the present analysis is based on the philosophy applicable to the GE 1000 MWe LMFBR design⁴.

The fault tree formulation incorporated many "removal probabilities". These probabilities indicate the likelihood that a given abnormal condition or fuel failure will be detected and the reactor shut down to correct the situation and remove any damaged fuel.

Evaluating these removal probabilities requires that the conceivable accidents and failure conditions be analyzed to determine the type of detectable signals which may emanate from such occurrences. These signals can then be compared with the reactor design and operating philosophy to determine the appropriate response.

The limited scope of the present effort did not allow that factors such as the sequencing and timing associated with given signals or sets of signals be adequately accounted for in the fault tree analysis. These factors, if thoroughly analyzed, could require a different response to the same types of signals. This would require a different assessment of the removal probabilities, depending on the sequence and timing elements assumed. It was not possible to consider these important and complicated factors in the present analysis.

Table 2-1 presents a list of many possible accident and/or fuel failure signals and combinations of signals which are assumed to be detected by the core instrumentation, along with the response appropriate to the reference design and operating philosophy. The list of possible signals involves several assumptions. One example is that of incremental fission gas detection. The reference reactor design employed vented fuel which is continuously releasing fission gas. Detection of a failure in which fission gas is released through the failure assumes that this new gas will either have a distinguishing isotopic composition or that the increment itself is large enough to detect. The table also assumes that it will be possible to distinguish different types of noise. Thus boiling noise is assumed to be distinguishable from that which may be generated by the bursting of a fuel pin. These assumptions, however, are not critical to the types of responses made to the various signals.

The responses set forth in the table are dependent on the degree of abnormality called out by the instrumentation signals. Small deviations from normal conditions do not justify scrambling the reactor except when the signals involve flow or temperature perturbations from at least a few fuel assemblies. Large deviations from even a single channel, however, require a scram. The implications of these design and operating philosophy characteristics were factored into the fault tree removal probabilities, as were the balance of the items presented in Table 2-1.

The basic philosophy behind the indicated response of the safety system is that the signals shown as emanating from the faults are generated by developed and reliable instruments. Moreover, it is assumed that these instruments and instrument probes are designed to protect against supposed damaging events based upon the knowledge of these events in today's technology. For example, items 23 and 24 indicate that even with only one channel indicating abnormal flow or abnormal coolant temperature, if that abnormality is as much as 15%, the reactor will be scrambled. This presumes first that the sensors and logic are sufficiently reliable that a 15% abnormality in a single channel can justify a scram of the reactor. Furthermore, it recognizes the current fear that a relatively modest disturbance in terms of flow or temperature in a single channel may be the initiation of a fault which can rapidly propagate into more damaging conditions in adjacent channels or the reactor core.

It is possible that later research will show that the philosophy upon which this table is based is inappropriate. For example, it is quite possible that much of the safety research now being carried out will demonstrate that propagation of damage from minor flow or temperature disturbances does not occur. In this event, it would be inappropriate to

Table 2-1. Response to Signals of Abnormal Core Conditions Based on the Reference LMFBR Design and Operating Philosophy

	SIGNAL										RESPONSE*											
	Gross Cover Gas Incremental Fission Gas Release		Gross Coolant Delayed Neutron Detection		Gross Core Acoustic Noise		Abnormal Flow	Abnormal Coolant Temperature	Gross Core Boiling Noise		~ 15% Abnormality				~ 10% Abnormality				~ 5% Abnormality			
											Alarm	Set-Back	Programmed Shut Down	Scram	Alarm	Set-Back	Programmed Shut Down	Scram	Alarm	Set-Back	Programmed Shut Down	Scram
	Small	Large	Small	Large	Small	Large			Small	Large												
1.	X																					
2.		X									X											
3.	X		X																			
4.		X		X							X	X										
5.	X		X		X						X		X						X			
6.		X		X		X					X		X						X			
7.		X					X (One Chan.)	X (One Chan.)			X		X	X					X			
8.		X		X			X "	X "			X			X	X		X		X			
9.		X		X		X	X "	X "			X			X	X		X		X	X		
10.		X								X		X							X			
11.		X		X						X		X							X			
12.		X		X		X				X			X						X			
13.							X "	X "			X			X	X	X			X			
14.										X			X						X			
15.					X		X "			X				X	X	X			X			
16.							X (Few)	X (Few)			X			X	X			X	X			X
17.							X (Many)	X (Many)			X			X	X			X	X			X
18.		X		X		X	X (Few)	X (Few)			X			X	X			X	X			X
19.		X		X		X	X (Few)	X (Few)		X				X	X			X	X			X
20.		X		X		X	X (Many)	X (Many)		X				X	X			X	X			X
21.							X (Many)	X (Many)		X				X	X			X	X			X
22.		X		X		X	X (Many)	X (Many)		X				X	X			X	X			X
23.							X (One Chan.)				X			X	X	X			X			
24.							X (One Chan.)				X			X	X	X			X			

* Assumes reliable instrumentation readouts.

subject the reactor to the extreme thermal transient which occurs during a scram for this cause. Additionally, it is possible, for example, that some of the signals indicated in the table such as delayed neutron detection or boiling noise can be made to detect a localized failure with more speed and reliability than is supposed in the table. Alternatively, the flow and temperature signals may not be as reliable as the table anticipates. In these events the response of the reactor protective system would be designed differently than that shown for illustrative purposes in this table.

Alternative philosophies may lead to entirely different responses to many of the signals outlined in Table 2-1. Differences which would change the removal probabilities as used in the fault tree analysis could lead to different fault tree results.

The scope of the present effort did not allow that the effects of alternative philosophies be examined. These alternative approaches could be significant and, if possible, they should be factored into the fault tree analysis. In this way it may be possible to identify an optimum philosophy which will lead to the lowest mean-annual-costs for the plant.

2.5 NOMENCLATURE AND USAGE FOR FAULT TREES

As noted previously, a rigorous description of the concept of fault trees is presented by Schroeder¹. For assistance to the reader, however, a review of certain fundamental symbols and their use may be helpful.

Symbols Used in Fault Trees

The overall fault tree is constructed by defining certain initiating events, transition events and final states. The initiating events are labeled "I" on the trees and are subscripted with a number to identify the specific event. Also included adjacent to the event designation is the estimated annual frequency of occurrence of the event.

Example
of
Initiating Event

The transition events or probabilities are designated "Q" with subscripted numbers to identify specific events. The estimated probability of occurrence is shown adjacent to this event number.

Example
of
Transition Event

Initial Crack Failure Induces Burst
Failure in One or More Adjacent Pins
 $Q_{103} = 0.001$

The final states are designated "S" with subscripted numbers for identification. The annual frequency of occurrence is indicated and is computed from the initial event frequency and the transition probabilities.

Example
of
Final State

The Reactor is Shut Down to Remove One
Fuel Bundle Whose Channel Wall
is Damaged
 $S_{10} = 2.385$

It should be noted that the designation "S" is also used to identify intermediate events in the fault tree between transition probabilities and the final state. Again, these intermediate events are shown with the computed annual frequency of occurrence.

The transfer boxes are used to transfer from one location on the overall fault tree to another. Included in the triangular boxes are identifying symbols which relate to where the transfer occurs.

Example of
Transfer Box

In this case, the symbol means transfer from page M into transfer box M3 on page L.

2.6 FAULT TREE EXAMPLES

2.6.1 Overall Fault Tree

The overall fault trees show the probabilities which must be evaluated in order to calculate the frequencies of the final events. The fault trees which were prepared are simply an outline of the paths leading from one of the twelve initiating events to one of the twelve final states. Figure 2-1 is an example (one of seventeen) of the overall fault trees (see Appendix II) used in this study. This example shows the propagation of a burst type fuel failure to a state where many fuel pins have burst. The quantitative evaluation of the fault tree probabilities was programmed for computer solution.

2.6.2 Mechanistic Fault Trees

In order to identify the means by which the transition occurs, and in order to assign probabilities to these transitions, it is necessary to construct mechanistic fault trees which identify the specific steps and to estimate probabilities for the steps. This is done separately from the overall fault tree preparation. Individual mechanistic fault trees are prepared for each of the major transition probabilities. These mechanistic fault trees are then used to estimate the probabilities of the various Q's (transition probabilities).

The individual transition probabilities depend on the mechanisms that cause the transitions to occur. For example, there is a transition in the overall fault tree labeled Q103 that reads: "Initial Crack Failure in One Pin Induces Burst

Figure 2-1. Overall Fault Tree Example

Failure in One or More Adjacent Pins" shown on Figure 2-2. One of the several mechanisms for this event is gas blanketing of the adjacent pin, leading to over-temperature and clad weakening followed by clad rupture and the expulsion of fuel through the clad breach and into the coolant. A diagram including all such mechanisms defines the mechanistic fault tree for the event of a crack failure inducing a burst failure in an adjacent pin. For a given fuel design, fuel performance level, and plant operating philosophy, there are certain probabilities of these mechanisms occurring. From past work done in the area of fuel element performance, defected fuel behavior and thermal hydraulics, those probabilities can be estimated, given the conditions. The state-of-the-art can be evaluated by determining the confidence with which probabilities can be assigned to the mechanisms. Mechanistic fault trees and the state-of-the-art are discussed in Appendix III.

2.6.3 Identification of Dominant Cut-Sets

The dominant cut-set is that sequence of transitions, from an initial event to the final event, which makes the largest contribution to the mean-annual cost for a given fault tree. A state-of-the-art review was conducted to gather information on fuel failure effects. Preliminary values were then assigned to the probability of each transition shown on the large fault trees (where a transition is the change of the reactor from one state, say large area failure in one pin, to the next more serious state, say large area failure in a few pins). In assigning the probabilities, the GE 1000 MWe LMFBR design⁴ was assumed.

Figure 2-2. Example of a Mechanistic Fault Tree

In assigning the preliminary probabilities, an attempt was made to maintain a constant level of confidence. (Where level-of-confidence is the probability that the true value is less than the chosen value.) Consequently, for those transitions where the state-of-the-art review produced very little information, it was necessary to assign relatively high probabilities.

The fault trees were then used to calculate the mean-annual-cost of the reactor end states, and to determine the "dominant cut-sets" associated with the preliminary probability assignments.

The dominant cut-set is that sequence of transitions, from an initial event to the final event, which makes the largest contribution to the mean-annual cost for a given fault tree.

Associated with each transition probability on the dominant cut-sets is a complete "mechanistic fault tree". (Where a mechanistic fault tree is one which shows all of the conceivable sequences of physical mechanisms through which the specified transition could occur, e.g. a large area failure in one pin induces large area failure in a few adjacent pins.)

Each of the mechanistic fault trees also has a dominant cut-set, for a given reactor design and operating philosophy. This cut-set was chosen (for the GE 1000 MWe design) in the assignment of the transition probabilities discussed above.

These preliminary dominant cut-sets were instrumental in identifying key questions which were amenable to further analysis. These questions served to direct the investigative efforts which were needed to reduce the uncertainties associated with the initial probabilities. Several areas of uncertainty were investigated and the description of these investigations form Volume II of this report. In this way greater confidence was gained in the probability assignments. The results of these efforts were then recycled through the fault trees. Consequently, greater confidence was also gained in the identification of the dominant cut-sets and the dominant failure mechanisms.

An example of a dominant cut-set is shown on Figure 2-3, detailing the most likely events causing reactor contamination by fission gas from many failed pins. The most likely initiating fault is simply a spontaneous fuel crack failure. A dominant cut-set through a mechanistic fault tree is shown on Figure 2-4 for part of the above overall dominant cut-set, the propagation of a large area failure to its neighbors. The dominant mode of propagation is considered local overheating due to fuel-sodium reaction product swelling.

The above descriptions cover all vital elements of the fault tree logic. The combined elements are illustrated in Figure 2-5.

Continuing evaluation of the fault tree as new data becomes available assures that development funds can be directed to the highest priority area.

The events of interest and the mechanisms causing transitions between them are first listed in fault tree form. The frequencies and transition probabilities are then estimated from the current state-of-the-art knowledge. Calculation of the intermediate and end state probabilities shows the dominant cut sets, the paths through the trees with the highest frequencies. Where event frequencies and transition probabilities are very uncertain, a sensitivity study is conducted by re-evaluating the fault tree with different values. This exercise shows the leverage that the uncertain events and transition mechanisms might have on the frequencies and costs. This demonstrates the principle incentive for further safety and/or core design research and development. As new design and safety information becomes available from analysis and testing, the fault tree is updated and new priorities are established. This process would logically be continued until the cost differences produced by reducing uncertainties of event frequencies and transition probabilities can no longer justify the research costs.

**EVENTS WHICH RESULT IN REACTOR CONTAMINATION BY
FISSION GAS FROM A BUNDLE WITH MANY FAILED PINS**

Figure 2-3. Example of a Dominant Cut-Set

MECHANISMS THROUGH WHICH A LARGE AREA FAILURE IN ONE PIN
INDUCES CRACK FAILURES IN ONE OR MORE ADJACENT PINS
Q204

Figure 2-4. Example of a Mechanistic Dominant Cut-Set

Figure 2-5. Fault Tree Logic

2.7 ASSIGNMENT OF QUANTITATIVE PROBABILITIES

Based on the state-of-the-art review and on analyses performed in this study, a set of reference probabilities was established. These reference values are used as a basis for performing sensitivity studies on design variables. Table 2-2 lists the "Q" numbers of the transitions and Table 2-3 the frequency of initiating events in the overall fault trees (see Appendix II). Also shown are the pages on which the transition appears, the reference probability for that transition to occur and a shorthand description of the transition. The following symbols are used in the shorthand notation:

C	=	crack failure
L	=	large area failure
B	=	burst failure
1	=	one
F	=	few
M	=	many
→	=	causes
LOC	=	loss of cooling
Void	=	channel voiding
RW	=	rupture of channel wall
Boil	=	coolant boiling
RF	=	reduced coolant flow
RB	=	partial flow blockage
Rem Prob	=	probability that the bundle(s) will be removed
Remains	=	probability that the bundle remains in the reactor
Sim. SP	=	simultaneous spontaneous pin failures
sec	=	secondary failure
PT	=	power transient

For example:

"M-B → F-RW" reads: Many burst failures cause the rupture of a few channel walls.

Table 2-2. Quantitative Fault Tree Inputs

Q	Page*	Reference Probability	Description	Q	Page*	Reference Probability	Description
4	D	.0001	Rem Prob 1-C	51	H	.00001	1-B → M-RW
5	D	.001	Rem Prob 1-L	52	H	.0001	F-B → M-RW
6	D	.01	Rem Prob 1-B	53	H	.001	M-B → M-RW
7	D	.001	Rem Prob F-C	54	H	.0001	M-L → M-RW
8	D	.01	Rem Prob F-L	55	H	.00001	M-C → M-RW
9	D	.01	Rem Prob F-B	56	H	1	1 Void → M-RW
10	D	.001	Rem Prob 1-C + 1-L	60	A,M	1	1-C Remains
11	D	.01	Rem Prob 1-L + 1B	61	A,C,L	1	1-L Remains
12	D	.01	Rem Prob 1C + 1-B	62	A,C,I	1	1-B Remains
13	D	.01	Rem Prob M-C	63	B,M	1	F-C Remains
14	D	.1	Rem Prob M-L	64	B,L	1	F-L Remains
15	D	.1	Rem Prob M-B	65	B,I	1	F-B Remains
16	E	.01	Rem Prob F-C + F-L	66	B,C,K	1	1-C + 1L Remains
17	E	.01	Rem Prob F-L + F-B	67	B	1	1-L + 1B Remains
18	E	.01	Rem Prob V-C + F-B	68	B,C	1	1C + 1B Remains
19	E	1	Rem Prob RF + M-C	73	A,K	1	M-C Remains
20	E	1	Rem Prob RF + M-L	74	A,C,J	1	M-L Remains
21	E	1	Rem Prob RF + M-B	75	A,C	.01	M-B Remains
22	E	1	Rem Prob Boil + M-C	76	A,B,K	1	F-C + F-L Remains
23	E	1	Rem Prob Boil + M-L	78	A,B,J	1	F-C + F-B Remains
24	E	1	Rem Prob Boil + M-B	79	A,N	.001	RF + M-C Remains
25	E	1	Rem Prob 1 - RF	80	A,C,N	.001	RF + M-L Remains
26	E	1	Rem Prob 1 - Boil	81	A,C	.001	RF + M-B Remains
27	G	1	Rem Prob 1-B + RW	82	A,N	.01	Boil + M-C Remains
28	G	.001	1-B → RW	83	A,C,N	.01	Boil + M-L Remains
29	G	1	Rem Prob F-B + RW	84	A,C	.01	Boil + M-B Remains
30	G	.01	F-B → RW	101	F1,K,M	.1	C L, same pin
31	G	1	Rem Prob M-B + RW	102	F1,K,M	.01	C B, same pin
32	G	.1	m-B → RW	103	M	.001	C B
33	G	1	Rem Prob M-L + RW	104	M	.001	C L
34	G	.01	M-L → RW	105	M	.001	C C
35	G	.1	Rem Prob M-C + RW	106	M	.0001	Sim. SP - C
36	G	.0001	M-C → RW	111	M	1	1, F-B + 1, F-C Remains
37	G	.1	Void → 1-RW	112	M	.001	sec - C → C
38	H	1	Rem Prob F-RW	113	M	.001	sec - C → B
39	H	.0001	1-B → F-RW	114	M	.001	sec C → L
40	H	.001	F-B → F-RW	115	K	1	F-C → F-L same pins
41	H	.01	M-B → F-RW				
42	H	.001	M-L → F-RW				
43	H	.0001	M-C → F-RW				
44	H	1	1-Void → F-RW				
45	E	1	Rem Prob 1-Void				
50	H	1	Rem Prob M-RW				

*Page letters refer to Appendix II.

Table 2-2. Quantitative Fault Tree Inputs (Continued)

Q	Page*	Reference Probability	Description	Q	Page*	Reference Probability	Description
117	K	.01	F-C → F-B same pins	636	F1	1	Rem. Prob. M-L in F-Bundles
201	F1,K,L	.01	L → B same pin	637	F1	1	Rem. Prob. M-B in F-Bundles
202	L	.1	L → B				
203	L	.1	L → L				
204	L	.1	L → C	642	F2,P	1	M-C in M-Bundles Remains
205	L	.00001	Sim. SP - L	643	F2,P	.01	M-L in M-Bundles Remains
212	J	.1	sec - L → B	644	P	.001	M-B in M-Bundles Remains
213	L	.1	sec - L → L				
214	K	.1	sec - L → C				
215	K	.01	F - L → F - B same pins	650	F1	.001	1-B → M-C in F-Bundles
301	I	1	B → B	651	F1	.001	1-B → M-L in F-Bundles
302	I	1	B → L	652	F1	.001	1-B → M-B in F-Bundles
303	I	1	B → C	653	F2	.0001	1-B → M-C in M-Bundles
304	I	.00001	Sim. SP - B	654	F2	.0001	1-B → M-L in M-Bundles
311	I	1	sec - B → B	655	F2	.0001	1-B → M-B in M-Bundles
312	I	1	sec - B → L	656	F1	.01	F-B → M-C in F-Bundles
313	I	1	sec - B → C	657	F1	.01	F-B → M-L in F-Bundles
404	O	.001	1-RF Remains	658	F1	.01	F-B → M-B in F-Bundles
405	O	.001	1-Boil Remains	659	F2	.001	F-B → M-C in M-Bundles
406	O	1	PB+Boil → M-B	660	F2	.001	F-B → M-L in M-Bundles
407	O	1	PB+Boil → M-L	661	F2	.001	F-B → M-B in M-Bundles
408	O	1	PB+Boil → M-C	662	F1	.1	M-B → M-C in F-Bundles
416	O	1	PB → M-B	663	F1	.1	M-B → M-L in F-Bundles
417	O	1	PB → M-L	664	F1	.1	M-B → M-B in F-Bundles
418	O	1	PB → M-C	665	F2	.01	M-B → M-C in M-Bundles
501	K	1	1,F-L+1, F-B Remains	666	F2	.01	M-B → M-L in M-Bundles
506	J	1	F-L + F-B Remains	667	F2	.01	M-B → M-B in M-Bundles
600	F2,N	.01	M-C → M-B same pins				
610	F2,N	1	M-C → M-L same pins				
620	F2,N	.01	M-L → M-B same pins				
630	F1	1	Rem. Prob. F-LOC				
631	F2	1	Rem. Prob. M-LOC				
632	F2	.01	Rem. Prob. M-C in M-Bundles				
633	F2	1	Rem. Prob. M-L in M-Bundles				
634	F2	1	Rem. Prob. M-B in M-Bundles				
635	F1	.01	Rem. Prob. M-C in F-Bundles				

*Page letters refer to Appendix II.

Table 2-2. Quantitative Fault Tree Inputs (Continued)

Q	Page*	Reference Probability	Description	Q	Page*	Reference Probability	Description
701	O	.01	1-Void → M-C in F-Bundles	808	P	.1	F-LOC → M-C but no RW
702	O	.01	1-Void → M-L in F-Bundles	809	P	.1	F-LOC → M-L but no RW
703	O	.01	1-Void → M-B in F-Bundles	810	P	.1	F-LOC → M-B but no RW
801	P	1	F-LOC → M-RW	811	P	.01	1 Void → M-C but no RW
802	P	.1	F-LOC → F-RW	812	P	.01	1 Void → M-L but no RW
803	P	.1	M-LOC → M-RW	813	P	.01	1 Void → M-B but no RW
804	P	.1	F-LOC → SHUTDOWN				

*Page letters refer to Appendix II.

Table 2-3. Initiating Event Frequencies, Events Per Year

I	Page*	Reference Frequency	Description
1	M	110	Crack Failures
2	L	11	Large Area Failures
3	I	1	Burst Failures
4	O	.01	Flow Reduction
5	O	.01	Local Boiling
6	O	.01	Channel Voiding
7	P	.01	LOC - Few Channels
8	P	.01	LOC - Many Channels
9	P	.01	Small to Moderate Δp
10	P	.01	Moderate Δp
11	P	.01	Large Δp
12	P	.01	Very large Δp

*Page letters refer to Appendix II.

3.0 CONCLUSIONS AND RECOMMENDATIONS

3.1 INTRODUCTION

The evaluation of the LMFBR safety development program in relationship to core design requirements began with the following fundamental questions:

1. What developments can potentially influence core design, operation and instrumentation?
2. What developments will influence risk assessment?
3. What developments can influence licensing success?

The common problem in answering these three questions is the uncertainty associated with certain accident consequences. In particular, accident mechanisms can be postulated which give sequences of events progressing from fuel failure propagation through sodium boiling to clad and fuel meltdown and violent boiling. The primary need identified for the safety development program is therefore to determine whether the postulated mechanisms are real, and if real, whether they are significant from the design, licensing and risk assessment standpoint.

After this is established, the key problem areas will have been established and emphasis can then be placed on achieving solutions. Development work aimed at showing core design influences on accident consequences are in this second phase of work. Nevertheless, in the following discussions, secondary, design-oriented recommendations are made even though the primary credibility of a mechanism has yet to be established at a significant confidence level. In effect, it should be kept in mind that the recommendations for design-oriented development are conditional upon the credibility of the basic mechanisms involved.

Table 3-1 shows the significant recommendations of this report and also notes the relationship to existing programs. The highest priority items are discussed in section 1.2. Following the summary table are the detailed results, including the reference fault tree results and sensitivity studies with the fault tree which contributed to establishing a qualitative priority ranking of recommendations. The items considered top priority are flagged (†) for clarity.

3.2 FAULT TREE RESULTS

3.2.1 Reference Fault Tree Results

The fault tree analysis has been applied utilizing the reference GE 1000 MWe reactor design. The probability estimates employed in the analyses were derived from the state-of-the-art survey and the results of the individual investigations carried out during this study. All the probability assessments are judged to be quite conservative. This approach was necessary since relatively large uncertainties surround the phenomena involved. In addition the frequency of occurrence of each of the discrete initiating events defined for the fault trees had to be estimated. Here, again, much uncertainty exists. The reference frequencies used in the analysis are felt to be reasonable and conservative.

The reference values of each of initiating event frequencies are given below. The fault tree sensitivities to these frequency assignments are discussed later in this section.

- I-1 — Crack type clad failures in one fuel pin.

It was arbitrarily decided that the fuel performance will be such as to result in an average of one pin per bundle with a crack type clad failure at the end of life. For the reference design, this is 234 failures per 790 days or:

110 events/year

- I-2 — Large area clad failure in one fuel pin.

This is arbitrarily taken as one tenth of I-1:

11 events/year

Table 3-1. Recommendations Summary

Key Area	Information Needs	Recommendation	Relationship to Existing Programs
1. Fuel Failure Propagation (Pin-to-Pin)	Propagation potential and rate: a) to adjacent pins b) to succeeding rows		
a. Influence of design parameters and properties on propagation failure thresholds	Clad ability to withstand: - local spacer contact loads - local overheating - shock pulses - internal pressures - fuel-clad attack at high temperature	1) † Develop transient mechanical clad failure model including effect of spacer contact stresses 2) Perform in-pile tests of biaxial creep rupture of fuel pins 3) Perform ex-pile burst tests of to 1800°F, simulating high burnup properties, temperature transient, and strain rate. 4) † Perform ex-pile spacer contact clad failure test 5) † Perform in-pile fuel-clad attack correlation to 1800°F 6) Perform ex-pile tests of external shock loads on pins	Transient extension of PA10, Task P Extension of Pa10, Task F to failure condition Extension of PA10, Task L to high temperatures in liquid New tests Extension of Task B, E to high temperature New tests
b. Fuel Debris Blockage Potential	Character of fuel failure debris. Effect of P/d, spacer design and edge design on debris trapping. Capability of the central void to retain gas.	1) Perform in-pile test of central void pressure 2) Perform in-pile or heated ex-pile test of fuel column response to central void pressure 3) Perform post-irradiation characterization of debris size and location from all failed pins and pin geometry changes	(Requires gas seal between fuel and pressure tube) extension of Task B work In-pile — Task B Ex-pile heater — Task E thermal cond. rig. No new tests — continue surveillance of existing programs.

Table 3-1. Recommendations Summary (Continued)

Key Areas	Information Needs	Recommendation	Relationship to Existing Programs
		4) Perform ex-pile test of debris trapping as a function of spacers, P/d, and debris size and shape, etc.	Potential use of ANL transparent test section.
		5) † Perform in-pile multipin test of blockage propagation potential	Task B Series 10
c. Fuel Swelling Potential	Effect of reactor conditions on fuel swelling. Influence of P/d and spacer design in accommodating swelling.	1) † Characterize fuel-sodium reaction product swelling, and propagation potential vs. fuel O/M, P/d spacer design and initial failure size.	On-going program Task B, PA10
d. Fission Gas Pressure Propagation	Effect of pressure level on adjacent pins.	1) Characterize propagation potential due to fission gas pressure-burst tests of one pin in multipin array (note-based on ANL's EBR-II Mark E tests — include channel mockup around pins.)	On-going program Task B, PA10 Also ANL's FEFP
2. Sodium Boiling			
a. Boiling Threshold	Scoping of blockage geometries and/or fission gas release which initiate boiling.	1) Perform analysis of thermal perturbations due to blockages and fission gas release 2) Perform ex-pile tests to confirm analysis	On-going — ORNL program Task G-10, PA-10
b. Boiling Stability	Transient stability of local boiling within a heated tube bundle.	1) † Perform modeling and ex-pile testing as above.	On-going — ORNL program Task G-10, PA-10
c. Boiling Heat Transfer	Cooling effectiveness of both local and bundle boiling.	Perform modeling and testing of clad temperatures during boiling incidents. 1) ex-pile testing to high power levels and high temperature. 2) in-pile testing	Difficult testing - may be possible with Al UO ₂ heaters Task C-2 PA10
d. Sodium Hammer	The mechanical effects of sodium boiling; sodium hammer and vapor bubble collapse.	1) Perform ex-pile test of water hammer in a core mockup.	New program

Table 3-1. Recommendations Summary (Continued)

Key Areas	Information Needs	Recommendation	Relationship to Existing Programs
d. Continued		2) Perform ex-pile test of vapor bubble collapse damage potential by vapor injection into a pin array	Possibly related to ANL's gas injection tests.
3. Fuel/clad-sodium Thermal Interaction	<p>The range of pressures and work energies in prototypic LMFBR accidents</p> <p>The damage potential for these energies</p>	<p>1) Model the effect of gas and/or vapor on heat transfer and compressibility (emphasis on experiment correlation) during fuel/clad sodium thermal interaction.</p> <p>2) Extend modeling to include structural response if shown to be necessary (emphasis on experiment correlation)</p> <p>3) † Perform in-pile prototype tests of accident conditions to establish the range of pressures and energies to be expected. Both power transient & loss of flow tests. Emphasis, radial heat transfer and radial structural accommodation of core.</p>	<p>Extension of PNL, ANL modeling.</p> <p>Extension of PNL, ANL modeling</p> <p>Tasks C-1, C-2, PA10 and ANL TREAT tests</p> <ul style="list-style-type: none"> — GETR C-2 tests recommended for loss of flow — ANL's Mark II loop for power transients in TREAT
4. Channel Containment Capability	The need for and then the capability of channels to contain accident pressures and energies.	<p>Perform ex-pile tests of channel containment capability to test the analytical predictions.</p> <p>1) Pneumatic test to simulate vapor expansion and fission gas release</p> <p>2) Hydraulic tests to simulate sodium hammer</p> <p>3) Explosive tests to possible simulate violent boiling.</p>	New Programs

I-3 — Burst type clad failure in one fuel pin.

This is arbitrarily taken as one tenth of I-2:

1.1 events/year

In reality, the spontaneous fuel failure rate will be controlled by the reactor operating philosophy. The economic optimization of fuel lifetime will likely involve consideration of failure rates and consequences. With sufficient fuel performance data, the failure frequencies and characteristics will become sufficiently well known to permit control of the spontaneous, end-of-life frequency by refueling scheduling and by power and flow control where necessary. The random failure rate due to defects will of course not be affected. This permits, within the limits of random failures, the selection of failure frequency as a dependent variable.

For comparison purposes, initiating frequencies I-4 through I-12, flow blockages and power transients, are truly random, in that no failures are expected; they will be rare and abnormal occurrences for LMFBR's. In estimating their frequency, a procedure first outlined by Arnett¹⁰ was initially followed. The input for this procedure is the total amount of experience with the given component, the number of observed failures, and the desired confidence level that the real frequency of failure will be less than the estimated frequency. Fast reactor operating experience was studied in an attempt to assign frequencies to I-4 through I-12. In this study it was found that there was little statistical data available on the frequency of occurrence of events involving flow blockages and reactivity transients. The few events recorded are not considered to be applicable because of design improvements which greatly reduce their probability of occurrence. Thus it would be overly pessimistic to base the estimates of frequency of flow blockages and reactivity transients on past experience, and this approach was abandoned. It was decided to arbitrarily assign the same frequency to each initiating event (I-4 through I-12) until a more appropriate scheme is devised. A sensitivity study was conducted to insure that this approach was reasonable. The results of this study are presented later in this section.

I-4 — Inlet flow blockage which causes a reduction of flow through only one fuel channel.

0.01 events/year

I-5 — Inlet flow blockage which causes local boiling in one fuel channel but does not immediately lead to channel voiding.

0.01 events/year

I-6 — Inlet flow blockage which causes immediate voiding of one fuel channel.

0.01 events/year

I-7 — Loss of cooling to a few channels.

0.01 events/year

I-8 — Loss of cooling to many fuel channels.

0.01 events/year

I-9 — Reactivity addition which causes crack type (large area) (burst) clad failure in many fuel bundles but
(1-10)(1-11) does not damage the channel walls.

0.01 events/year

I-12 — Reactivity addition which causes channel wall damage in many fuel bundles.

0.01 events/year

Table 3-2 presents the results of the fault tree analysis for the reference case based on the GE 1000 MWe design. In section 2.2 it was argued that the cost per event of each of the first five final states (fission gas leakage from either one, few, or many fuel pins, and fuel and fission product release from either one or a few fuel pins) is negligible. The mean-annual-cost due to the remaining seven states is summarized in Table 3-2, as well as the frequency of occurrence of each of these final states.

Table 3-2. Calculated Frequency and Mean-Annual-Cost of Fuel Failure Effects and Abnormal Conditions In a Reactor Design Similar to the GE 1000 MWe LMFBR (GEAP-5678)

End State	10 ³ \$ Per Event	Events Per Year	10 ³ \$ Per Year
S6 Fuel Release From Many Pins	18	45	832
S7 Remove One Sound Bundle	147	3.5	519
S8 Remove Few Sound Bundles	156	5.4	841
S9 Remove Many Sound Bundles	196	0.6	112
S10 Remove One Damaged Bundle	214	2.4	510
S11 Remove Few Damaged Bundles	544	0.2	108
S12 Remove Many Damaged Bundles	1839	<0.1	91
		Total	3019

The values presented in Table 3-2 indicate that the largest costs are associated with end states wherein fuel pins are damaged but the channel walls remain intact. This result obtains from the estimate that these final states have higher frequencies than the more costly states S10 through S12. The greatest reduction in the mean-annual-costs, therefore, may be obtained by efforts which reduce the frequency of end states S6 through S9 rather than the frequency of the last three states.

The costs-per-event shown on Table 3-2 were presented in section 2.2. These values are optimistic because they do not include an extensive period of reactor downtime after each "incident" for the operators to convince the regulatory agencies of the safety of the reactor. This process apparently required over three years in the case of the Enrico Fermi Fast Breeder Reactor. Assuming longer shutdown periods after each event will result in higher costs-per-event, and will show an even stronger incentive to reduce the frequencies of the final events.

Table 3-3 shows a further breakdown of the reference case fault tree analysis. Here the cost contributions from each of the initiating events are tabulated. Contributions to both the costs of each end state and to the total costs are shown. This breakdown clearly indicates that spontaneous fuel pin failures have a much greater impact on costs than do initiating events such as flow blockages and reactivity transients. It is the high frequency of fuel pin failures, the conservative estimates of their effects and the probable difficulty of detection which cause these initiating events to dominate the cost picture.

Of these, the 'crack' type failure dominates because of its high frequency and the conservative assumption of probable progression to 'large area' and 'burst' type failures. These conclusions are obviously critically dependent on the values chosen for the frequencies of the initiating events and the transition probabilities. The judgments of these frequencies and probabilities should therefore be reviewed from time to time and by independent groups of knowledgeable people.

Sensitivity to Initiating Event Frequency

The previous discussion indicated that the initiating event frequencies used in the base case analysis were somewhat arbitrarily assigned. This is especially true of I-4 through I-12. The results of Table 3-3 show that the initiating event frequencies and/or transition probabilities are so low compared to fuel failure effects as to almost make events I-4 through I-12 insignificant.

In order to show the sensitivity of the fault tree analysis to the initiating frequencies of flow blockages and reactivity transients, the analysis was performed for a wide range of frequencies. As can be seen from Figure 3-1 the

Figure 3-1. Sensitivity of Mean Annual Cost to Changes in Initiating Event Frequency

**Table 3-3. Reference Case
Contribution from Each Initiating Event
(\$1000 per year)**

Initiating Event	Fuel Failure			Flow Reductions					Reactivity Transients				Total
Final State	<u>1 1</u>	<u>1 2</u>	<u>1 3</u>	<u>1 4</u>	<u>1 5</u>	<u>1 6</u>	<u>1 7</u>	<u>1 8</u>	<u>1 9</u>	<u>1 10</u>	<u>1 11</u>	<u>1 12</u>	
S6	712	65	54	0	0	0	0	0	2	0	0	0	832
S7	440	40	34	2	2	2	0	0	0	0	0	0	519
S8	718	65	58	0	0	0	0	0	0	0	0	0	842
S9	90	8	7	0	0	0	1	0	2	2	2	0	112
S10	434	39	35	0	0	2	0	0	0	0	0	0	510
S11	87	8	7	0	0	5	1	0	0	0	0	0	108
S12	29	3	2	0	0	18	18	2	0	0	0	18	91
Total	2511	227	198	2	2	28	20	2	4	2	2	18	3014

mean-annual-cost is not greatly affected until the frequencies exceed 0.1 event per reactor-year. This frequency means that, on the average, the nine initiating events would be expected to occur once, each, per 10 years. In other words, either a significant flow blockage or reactivity transient would occur about once every year in each reactor. This is an excessive rate to expect for these types of events.

The reference value for these frequencies has been taken as 0.01 events per reactor year.

It is seen that the fault tree analysis is not sensitive to decreases in this value, and is only slightly sensitive to an increase in this value by a factor of ten.

Also shown are the cost sensitivity to each of the fuel failure types, showing the dominant sensitivity of crack failures.

3.2.2 Public Safety Implications of Accident Chains

The results of the fault tree analysis have been quantified in terms of a mean-annual-cost parameter. This parameter is made up of the contributions from the costs associated with each of the undesirable reactor final states S1 through S12. Public safety considerations have not been used as a measure of any of these final states, at least not directly. In general, however, public safety is improved whenever the safety related mean annual costs are decreased. Lower costs are achieved by reducing the frequency of occurrence of the various undesirable reactor end states. This lower frequency also means a reduction in the number of times the public may potentially be exposed to hazardous conditions as a result of the reactor accident chains. The fault tree sensitivity analyses discussed in section 3.4 point to many design features which are quite influential in determining the safety through their effect on accident frequencies, and probabilities. Reduced hazards to the public, therefore, could be obtained in the same manner as cost reductions.

The reactor end state which can pose the greatest threat to public safety is S12. If the reactor has proceeded to this state it has undergone conditions severe enough to damage many channel walls. The core disassembly accident is also encompassed by this end state. Public safety may be endangered by this state because the types of accidents

leading to S12 are those which are most likely to result in violations of the reactor containment. Fortunately, the chain leading to such accidents is long, and each step can be combatted to the extent that the disassembly accident has an extremely low probability of occurrence.

Public safety can best be served by reducing the frequency of end state S12 as much as possible. For the base case conditions the fault tree analysis showed that the dominant cut set leading to S12 involves progressively worsening fuel pin failures which eventually propagate to a substantial core accident. Crack type failures in the fuel pins are the probable initiating events for this accident chain. Any steps taken to reduce the probability of fuel pin failures would lead to a reduced likelihood of the reactor proceeding to the point wherein many fuel assemblies are damaged. Such steps are reviewed in Section 3.4.

Sufficiently decreased probabilities ($\sim x 1/100$) of fuel pin failures eventually result in new dominant cut sets leading to end state S12. Three separate paths through the fault trees appear to be of equal likelihood in this case. The initiating events at the heads of these paths are as follows:

- I-6 — Inlet flow blockage which causes immediate voiding of one channel.
- I-7 — Loss of cooling to a few channels.
- I-12 — Reactivity addition which causes channel wall damage in many fuel assemblies.

The progression along these paths from the initiating events to the reactor end state appears difficult to impede by core design changes. This is obviously the case when a reactivity addition such as I12 initiates the accident. Public safety could best be served by designing and operating the reactor so as to reduce the frequency of occurrence of I-6, I-7, and I-12 to such low levels as to render them extremely improbable.

If the initiating event occurs, safety system performance determines the probability of these chains progressing. If the safety system is functional, it stops the progression before the public is endangered. Economic loss may still occur because of partial core damage even though the DBA is stopped.

As noted previously public safety considerations have not been factored directly into the costs associated with each of the defined reactor end states. The costs were based on the revenues lost when the reactor is shut down to correct the consequences of an accident chain. Reactor end states S6, S7, and S8 contributed the most to the total mean-annual-cost parameter, and end state S12 played a relatively minor role.

If public safety is endangered by any of the reactor end states then it may be reasonable to attach an additional cost to that state. These costs could be quantified in terms of the insurance coverage which must be obtained to cover damages to the public resulting from that accident condition. If insurance cannot be obtained for such accidents much greater costs must be attached to these consequential end states. Very large expenditures may be necessary to rectify damages to public health and property resulting from such accidents. Adding these costs to end state S12 would increase its prominence substantially. Such considerations may result in a shift in the priorities for needed development work. No attempt was made in this study to include quantitative public safety implications.

3.2.3 **Fault Tree Sensitivities for Identifying Priority Needs**

Establishing priorities for safety research is necessary in order that the most efficient and timely use be made of the resources available for this purpose. Fault tree analyses were used in this study to identify these priority areas. These analyses pointed to the accident chains which appear to contribute most substantially to the safety related mean annual costs of the reactor. By studying each failure mechanism and event in the dominant accident chains, it is possible to postulate means by which the initiating events can be prevented or else the progression along the chain can be stopped or significantly slowed. Alternately, the design may be modified such that the reactor can more readily accept the consequences of an accident chain or event while minimizing the hazards and economic penalties. Specific design changes can be pointed out which will most likely bring about the desired improvement in reactor safety. Research is then needed to assure that the design parameters in question do influence safety as predicted and to quantify the relationships involved. Many design parameters can possibly enhance reactor safety and reduce safety related costs. Sensitivity analyses are useful in identifying those parameters which are most influential in this respect.

One of the goals of this study was to identify those LMFBR fuel assembly design parameters which may have a significant influence on the core safety. The fault tree analysis was used to identify the major problem areas. The subsequent investigations of these problem areas served to point out the major design parameters of interest, and the nature of their influence on fuel failure probabilities.

The economic consequences of accident chains can be quite significant. This is evident from the estimates of the costs per event of the various reactor final states S1 through S12 as presented earlier. Any steps taken to reduce the frequency and probability of accidents and fuel failures will serve to reduce the number of times the reactor proceeds to one of these costly end states. These steps appear in the form of specific core design changes which enhance the safety characteristics of the core and reactor systems.

This section discusses the important safety related fuel assembly design parameters and the possible influence of these parameters on the safety-related costs of operating the reactor.

Core and reactor system design parameters must be optimized in such a way as to minimize the power generation costs of the plant. Costs associated with plant outages due to fuel failures and similar effects must be factored into this economic optimization. This generally involves a compromise of the performance to a certain extent. For example, reactor down-time might be minimized by employing a fuel with very thick cladding. Such a design choice would supposedly reduce the number of cladding ruptures expected during operation. The reactor employing this fuel design should enjoy a high availability factor. On the other hand, the use of the thick cladding increases the amount of steel in the core. Core neutronics suffer because of this, resulting in a reduced breeding ratio and the need for a high fuel enrichment. These factors increase the fuel cycle costs. All such effects must be properly accounted for in order to arrive at a balanced and optimized design.

An effort has been made during this study to quantify, in a preliminary way, the influence of several fuel assembly design parameters on the mean-annual-costs of operating an LMFBR. Estimates have been made of the accident and fuel failure transition probability changes associated with varying a given core design parameter. These revised probabilities were then recycled through the fault tree to estimate the new mean-annual-costs.

Table 3-4 presents a summary of the mean-annual-costs effect of several key fuel assembly design parameters. Other design variables such as specific power and coolant temperature which may possibly affect core safety could have been included in the table. Their exclusion was based on 1) the lack of reasonable measure of their quantitative impact on the safety related economics, 2) their relatively small influence, or 3) the knowledge that changing certain parameters enough to significantly influence core safety would lead to a sufficiently drastic reduction in the core performance as to render the design uneconomical and noncompetitive.

The base case core design and performance characteristics were based on the GE 1000 MWe Follow-On Study⁴. This design employed vented fuel pins. Because pressurized sealed pins are still candidates for LMFBR's, their safety related costs have been included in the table.

Table 3-4 was generated by modifying those transition probabilities influenced by the design changes indicated. The cost contribution from each of the possible reactor end states is shown in the table, as well as the total costs for each case.

Several cases were investigated for each of the fuel assembly design parameters tabulated. The changes made to the related base case probabilities are indicated by the "probability modifier." Modifiers both greater than and less than unity were used in some instances in order to determine the sensitivities about the reference point. In most cases relatively small decreases in the transition probabilities have a large impact on the plant mean annual costs. This implies that even small design improvements which can bring about the decreased failure probabilities will likely be worthwhile.

Table 3-4 reveals that, in most cases, the most costly reactor end states are those in which the reactor is shut down to remove fuel assemblies with damaged fuel but whose channel walls are undamaged. This result obtains from the fact that, although the end states in which channel walls are also damaged are more costly per event, their frequency is lower. The higher frequency of the undamaged wall events dominates in terms of producing the highest mean-annual-costs.

A comparison of the values presented in Table 3-4 reveals the relative importance of each of the design changes considered. The order of the priorities is given below. It is based on the cost savings which may potentially be realized by incorporating the indicated design features.

1. Stronger and/or improved fuel pin cladding. Such changes are assumed to result in a reduction of all probabilities of pin failure.
2. Annular fuel pellets rather than solid or powder fuel form. This configuration is assumed to reduce the probability of obtaining burst type fuel pin failures.
3. Increased pin diameter and/or pin spacing. The trapping of debris within the fuel pin lattice is reduced when

Table 3-4. Influence of LMFBR Fuel Assembly Design Parameters on the
Mean - Annual - Costs of Fuel Failure Effect

Mean Annual Costs 10^3 \$

Reactor Final State	10 ³ \$ Per Event	Design Area	Base Case	Fuel Pin (Stronger)			Cladding (Weaker)		Annular Pellets				Pin Spacing				Fuel Sodium Reaction Product Swelling				Channel Wall (Stronger) (Weaker)				Sealed Pins																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																		
		Probability Modifier	1.0	1	1	1	1.1	2.0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

such changes are made. In addition, the likelihood of adjacent pins being overheated by one which swells and restricts the coolant flow is reduced by having a larger pin spacing. Such improvements reduce the probability of propagation of certain types of pin failures.

4. Reduced fuel-sodium reaction product swelling. Steps taken to reduce this swelling, such as reduction of the fuel O/M ratio, will lower the probability that adjacent pins will be overheated because one pin has failed and this reaction has taken place.
5. Improved channel wall integrity. Such improvements may be made by employing better materials, a stronger configuration (thicker walls), gaps between channels to reduce overheating problems, or other such measures. These changes are assumed to reduce all probabilities of channel wall failures.

The potential cost savings discussed thus far are based solely on the mean-annual-costs associated with reactor outages and fuel failure effects. These cost savings must be measured against the possible deleterious effects on normal fuel cycle costs and/or plant capital costs resulting from the incorporation of these selected fuel assembly design features. Each suggested change must be examined thoroughly to assure that it will result in a net reduction in the overall plant operating costs.

The most sensitive parameter is fission gas accommodation. Sealed pins are expected to suffer higher probabilities of failure and failure propagation, with attendant risks of increasing safety related mean-annual-costs above these for the reference vented-fuel design.

An additional sensitivity study was made of failed fuel behavior. Secondary fuel failure propagation is defined as propagation beyond the immediate neighbors of a failed fuel pin. It is uncertain whether secondary fuel failure propagation will occur at all. The following table 3-5 shows the cost sensitivity to changes in the assumed probabilities of secondary fuel failure propagation.

Table 3-5 Fault Tree Sensitivity Study

Final State			Base Case Mean Annual Cost (in 1000's)	Secondary Failure Propagation Beyond Adjacent Pins			
				100x	10x	1/10x	1/100x
Continue Operation With Failures	— S6		832.4	2400.9	2400.1	85.8	11.2
Remove	one	— S7	518.7	1275.6	1267.3	148.9	111.9
Sound	few	— S8	841.5	862.0	862.0	102.5	28.6
Bundles	many	— S9	111.8	114.3	114.3	19.5	10.3
Remove	one	— S10	510.4	702.3	701.8	61.9	17.0
Unsound	few	— S11	107.8	156.4	156.2	18.4	9.5
Bundles	many	— S12	91.4	107.9	107.8	61.2	58.2
Total			3013.9	5619.4	5609.5	498.3	246.1

This shows that the safety related economics are particularly sensitive to reductions of secondary failure propagation. This sensitivity is due to the base case assumptions of secondary failure progressions being

- 0.01 for crack failures
- 0.1 for large area failures
- 1.0 for burst failures.

The maximum value for any transition probability is, of course, 1.0. Progressions from burst failures were assigned this value for the base case. Burst failures were found to be the principal damage mode leading to propagation across channel walls and to channel wall damage. End states S8 through S12 are unaffected by increases in secondary propagation probabilities, therefore, since the likelihood of progression from burst failures is already assigned a value of 1.0. On the other hand, these end states are directly affected by probability decreases.

Design features which might affect secondary fuel failure propagation include fuel form, spacer design and pin pitch to diameter ratio. These relationships have not been analyzed for lack of any behavioral models of fuel particle movements within a fuel bundle.

The following section details the conclusions and recommendations for design oriented safety research.

3.3 DESIGN DEVELOPMENT NEEDS

The following discussions treat in detail the conclusions and recommendations of this study which have potential bearing on the fuel assembly design of the LMFBR. If the two principal damage propagation mechanisms, fuel-sodium reaction product swelling and fuel debris blockage, are found to occur under reactor conditions, the fuel assembly design parameters identified could have a significant effect on costs of reactor operation. Potential cost changes affected by altering the design must be measured against the possible deleterious effects on fuel cycle costs and/or plant capital costs.

For lack of better data, this study assumed that both fuel-sodium reaction product swelling and fuel debris blockage do occur. The resulting cost sensitivity and design significance must be interpreted as being quite conservative.

The organization of this section is noted to be different than the summary of section one. The recommended priorities of development work in section one recognize the need to establish the credibility of the fuel swelling and debris blockage mechanisms. This section is organized to show the possible design implications of safety research. Each design parameter which may influence the progression of an accident is considered.

3.3.1 Improved Fuel Pin Clad Performance

Economic Incentives

Cladding performance may well be the most important fuel assembly design parameter affecting LMFBR safety related mean annual costs. This is clearly borne out in the values presented in Table 3-4. The dominance of this parameter results from the judgment that clad performance influences all probabilities of fuel pin failures.

Improved clad performance is intended to encompass all measures whereby fuel pin failures will be reduced by virtue of making the clad less susceptible to damaging influences. The list below suggests several areas where improvements are possible and possible means by which the benefits may be obtained.

Improvement	Means of Accomplishing
Greater strength, more resistance to forces exerted on cladding.	Thicker cladding, stronger materials.
Greater creep resistance.	Same as above.
More resistance to irradiation damage.	Improved materials.
Less susceptibility to effects of corrosion and chemical attack.	Improved materials, corrosion and chemical attack deterrents, thicker cladding.

Clad Thickness

By far the simplest and most direct method of improving clad performance is by utilizing thicker cladding. All else being constant, this measure should result in fewer fuel pin failures for any given reactor. This design parameter can have a significant effect on the fuel cycle costs. The added structure in the core necessitates a higher fuel enrichment. Breeding ratio may also be affected. The use of thicker cladding, therefore, would have a deleterious effect on the costs

of energy production. These consequences must be measured against the potential cost savings to be gained because of the reduced number of pin failures.

The base case probability assessments of the present analysis were intended to apply to the GE 1000 MWe design reported in reference 4. This design utilized a clad thickness of 10 mils. It is possible that more advanced designs may seek to employ even thinner walls. Pin failure probabilities would be expected to increase in this case. The sensitivity of the safety related mean-annual-costs to such changes in the probabilities were examined, as well as cases representative of designs employing thicker cladding.

Thicker cladding reduces the probability of pin failures, but the degree of reduction is difficult to predict with the present state of the art. Table 3-4 and the values tabulated below show that significant incentives exist to achieve even relatively minor improvements in cladding performance. On the other hand, the values also indicate that severe cost penalties may be associated with any degradation in the ability of the cladding to cope with the reactor environment and its associated hazards. Such a degradation may occur from an intentional reduction in the clad thickness. These analyses also show the extent to which the reference core costs may change if the probability assessments representing that case were overly optimistic. The table below gives the change in the safety related mean annual costs as a function of the change in the failure probabilities. Fuel cycle cost effects are not included.

	Base Case	Modified Clad Performance						
		(Example) (Thicker Clad)				(Example) (Thinner Clad)		
Probability Modifier	1.0	$\times \frac{1}{1.1}$	$\times \frac{1}{1.5}$	$\times \frac{1}{2}$	$\times \frac{1}{10}$	x1.1	x1.5	x2
Safety Related Annual Costs Change, 10^3 \$	0	-950	-2340	-2720	-2930	+ 616	+3994	+10,700
Energy Cost Change, mill/kwh	0	-.136	-.335	-.389	-.419	+.088	+.571	+1.525

Fuel cycle cost changes due to clad thickness variations may be assessed from trends such as those presented in GEAP-5710¹¹. That information indicates that core costs increase by about 0.015 mill/kwh for each additional mil of clad thickness. This cost rate applies to a core achieving an average burnup near 100,000 MWD/MT. The rate may be even higher since the costs presented in GEAP-5710 were generated using a fairly optimistic basis. Increasing the clad thickness from 10 mils to 15 mils will result in a fuel cycle cost increment of about 0.075 mill/kwh. The fault tree analysis indicated that such additional costs may potentially be offset if the change in the cladding results in as little as a 10 percent reduction in the fuel pin failure probabilities.

The relationship between thickness and failure probabilities is needed before the cost tradeoff involved can be properly evaluated.

Clad Inside Diameter Attack

Severe attack of the cladding inside diameter has been noted in several in-pile tests. The present state-of-the-art concerning these phenomena was reviewed in Section 4.1. The present belief is that this mechanism is strongly dependent on the clad inside diameter temperature. It is also supposedly caused by the presence of certain fission products. This phenomena must be better understood if it is to be controlled or eliminated. No cost relationships could be identified with this phenomenon and thus it is noted to be qualitatively similar to cladding thickness in its possible cost sensitivities.

Informational Requirements

What is required is a quantitative relationship between these cladding parameters (thickness, strength, ductility, corrosion resistance) and the fuel-pin lifetime failure frequency and resistance to propagation mechanisms. Both steady state operation and transient survival capabilities must be investigated.

Clad performance influences on failure probabilities due to the following potential failure mechanisms must be defined.

- Local overheating effects from gas blanketing, trapped debris, fuel swelling, or other local flow disturbances.
- Spacer contact loads.
- Shock type loadings, such as that from fuel-sodium interaction, sodium hammer, or sodium vapor bubble collapse.
- Pin Internal forces such as due to fuel swelling, gas pressure, or differential thermal expansion between clad and fuel.
- Chemical attack of the clad inside diameter.
- Erosion or corrosion of the clad inside diameter.

Each of these mechanisms which can contribute to cladding failure is in itself a phenomenon requiring further study and investigation.

Test Recommendations

The establishment of a mechanical failure criterion for the cladding has been identified as an important development goal. The effects of radiation-induced creep should be accounted for with in-pile, biaxial creep rupture tests of cladding materials. The effect of spacer contact forces on the cladding should be similarly investigated. Accommodation by the cladding of high localized stresses through thermal and irradiation-induced creep is an important consideration in determining the cladding failure limit.

The behavior of the cladding under abnormal spacer contact loading should be investigated out of pile in order to provide input to possible in-pile tests. Some of the tests should take place at elevated temperatures and with pressurized tubing to determine the mode of failure to be expected.

The tube-burst tests that have been reported in the Task L work of the P.A. 10 program are mainly conducted in air. The large deformations observed in the tubing indicate that some tests should be conducted under sodium (or a similar liquid) to determine if the inertia of the liquid is sufficient to prevent those deformations. The ability of cladding to resist shock type loadings, such as vapor bubble collapse, should be tested out-of-pile over a wide temperature range and with representative spacers.

Section 4.1 pointed out the importance of clad inside diameter attack in limiting the life of an overheated fuel pin. More work must be done to determine the dependencies of this phenomena on parameters such as temperature, clad stress levels, and availability of fuel and fission product elements which may cause the attack. Task E of the PA-10 program is seeking to provide this information, as well as developing means to control the attack. This work should continue with a high priority and should be extended to temperatures of interest in failure propagation (1800°F).

Knowledge of this potential failure mechanism is important to the design of LMFBR fuel because it helps to establish failure threshold limits. It must also be accounted for in selecting the clad materials and thickness.

A similar but apparently much less sensitive mechanism is clad sodium corrosion. The temperature range should be extended as noted above but on a lower priority than clad-fuel attack.

3.3.2 Fission Gas Pressure

LMFBR fuel pin designs may employ a wide range of internal pressures. Sealed designs will probably have the highest pressures in order to maintain the plenum length at reasonable levels. Intermediate pressures may be employed by venting schemes which attempt to reduce the activity level of the fission gases escaping from the pin. At the lower end of the scale is the nonpressurized vent design. Here the pin internal pressure is essentially that of the coolant at the location of the vent outlet.

Fuel pin lifetime and reliability is expected to decrease as internal pressures rise. The potential economic consequences of the attendant increase in failure probabilities was shown in Section 3.3.1. These consequences provide a significant incentive to investigate this design parameter thoroughly.

A vented fuel pin concept was selected for the reference core design⁴. Venting allows the use of a shorter pin length relative to what would be required for a sealed design with the same target burnup. The non-pressurized design should also result in a higher reliability since the cladding operates at lower stress levels. Such measures of performance provide significant incentives for vented designs.

The present study sought to define the safety related cost differences between sealed and vented pins. The reference pin design operated at the same pressure as the coolant at the vent outlet. Pressurized venting schemes have been proposed as a means of controlling the fission gas activity which escapes from the pin. These designs operate in a pressure range between that of the non-pressurized pin and that of a normally sealed pin. Because the cladding is stressed by the internal pressure, such mid-pressure schemes were categorized as a type of sealed pin for the purposes of the fault tree analysis.

Sealed pins increase the probability of pin failure propagation in several ways. First a breach in the clad will result in a greater quantity of gas being released from the pin. This release may cause gas blanketing and overheating of adjacent pins. Secondly, the higher internal pressure of sealed pins increases the likelihood of all types of pin failures.

The fault tree analysis demonstrated that the former propagation mode (increased gas release) plays only a minor role. Cases were studied wherein only the probabilities of failure propagation due to gas blanketing were increased. It was assumed for these cases that the design maintained low internal pressures, but that a larger inventory of gas was present. The tabulation given below shows that the safety related core costs rose only slightly in these cases relative to that of the reference vented pin design.

Conditions more typical of sealed pins wherein the internal pressure was assumed to be significant resulted in fairly dramatic increases in the safety related costs. Pin failure propagation probabilities were increased for these cases, as well as the frequency of the spontaneous pin failures. If the probabilities and the spontaneous event frequencies are higher than those of the reference vented design by only a factor of ten percent, the costs rise by about \$770,000/year. Greater failure likelihood results in much greater increases in the costs.

	Base Case	Sealed Pin Designs				
		Increased Gas Release		Increased Gas Pressure		
Probability Modifier	1.0	x10	x100	x1.1	x1.5	x2.0
Safety Related Annual Costs Change, 10 ³ \$	0	+208	+2874	+766	+4562	+12,271
Energy Cost Change, mill/kwh	0	+.030	+.410	+.109	+.651	+1.750

The reference design assumed in the fault tree analysis employed a fairly simple, non-pressurized, vented pin design. The selection of the vented concept was based on the economic incentives demonstrated in GEAP-5710¹¹. The fault tree results reinforce the correctness of the selection. The results may be viewed as an additional incentive to develop venting schemes which are simple and reliable, and which will meet the performance goals assumed for them during this analysis.

Information Needs

The following questions must be answered concerning pin internal pressure levels:

- What is the quantitative relationship between pin internal pressure and the expected pin lifetime?
- How does pin pressure affect the transient overpower or undercooling survival capabilities of the pin?
- How does the pin internal pressure affect the way in which the pin is likely to fail? How does it affect the most likely failure location?

- How well does the gas in the plenum region of the pin communicate with the fueled regions of the pin, including the central void?
- To what extent does the pressure level endanger adjacent pins and/or channel wall when a given pin fails?

Recommendations

A broad range of tests can be imagined which would determine failure thresholds for various perturbations as a function of fission gas pressure. These are logically a part of the clad performance tests mentioned above. Both steady state and transient testing to failure, in- and out-of-pile, appear necessary to understand the fundamental mechanisms. With regard to fuel failure propagation, local overheating and excess spacer contact forces are the dominant perturbations.

The mode and consequences of failure are more important in some respects than whether failure occurs. The direct effect of failure of a sealed pin is a sudden pressurization of the fuel bundle. Based on the preliminary results of ANL's EBR-II Mark E subassembly tests, it is recommended that the consequences of sealed pin failures on surrounding pins and the channels be investigated. The surrounding pins and channel should be severely coldworked to simulate the loss-of-ductility at high fluences. All pins should be pressurized to determine if failure propagation occurs with sealed pins.

The second major influence of fission gas pressure on failure consequences is the possibility of fuel debris ejection (burst failures). If the direct effects of bundle pressurization are negligible and sealed fuel pins fail in the 'bursting' mode, the results of this study show that a sealed pin design may suffer significant economic penalty due to fuel failure effects in comparison with vented designs. The general implications and test recommendations concerning burst failures are discussed in the following section on fuel form.

3.3.3 Fuel Form

Burst type fuel pin failures were defined for the fault tree analysis by the characteristic of forcibly ejecting fuel particles out of the ruptured clad. This type of failure was given a fairly high probability of leading to other failures in adjacent pins. One of the dominant mechanisms involved in the failure propagation was the overheating effects caused by the fuel debris released from the initial failure. This debris can become trapped in the fuel pin matrix, leading to the disruption of the flow locally and the subsequent overheating of the nearby pins. The fuel form, powder or pellet, and geometry, annular or solid pellets, along with fission gas pressure, are the most likely design parameters possibly affecting the probability of burst failures occurring.

It is possible that fuel particles may be released from a large area type fuel pin failure as well as a burst failure. The mode of release might be different, but the consequences would probably be similar. In this regard it is unimportant to be able to distinguish which type of failure has occurred. What is important, however, is to be able to define and understand the mechanisms which cause fuel particles to be released into the coolant stream. Several mechanisms can be postulated. One is the gas pressure which may be present in the central void of the fuel pellets. If this pressure is significant, it could very well propel fuel particles out of the ruptured clad. Vapor pressure from sodium which enters the failed clad may also provide the driving force. These mechanisms were explored analytically in Section 6.3. On the other hand, some fuel chunks may simply passively fall out of the breach in the clad. This may be especially likely if powdered fuel is employed or after the fuel pellets have cracked due to thermal cycling.

The driving force which expels fuel from the ruptured pin into the coolant is the gas pressure in the central void and fuel matrix. The central void pressure may possibly be much higher than that of the pin gas plenum or vent. This is due to the fact that during normal operation the path for gas flow between the fueled regions of the pin and the gas plenum or vent areas are quite restrictive. These restrictive flow paths result in poor communication between the distinct gas volumes present in an LMFBR pin. It is possible, therefore, that pins which are designed to operate with low plenum pressures may have high central void pressures.

The probability of any failure progressing to a burst failure can be substantially reduced by reducing or eliminating the central void gas pressure. One way in which this can be accomplished is to insure that the central void gas volume can communicate freely with the gas storage volume or vent regions of the pin. Annular pellets can be used for this purpose. Both the axial blanket pellets and the core fuel pellets would have to be fabricated with the annular configuration. Annular pellets would perform this function as long as the central void passageway would not be blocked

or closed up during the life of the pin. Irradiation experience to date indicates that the central void does not seal up, and that adequate performance can be expected.

Another benefit to be gained from the use of annular fuel pellets is that associated with the transient survival capabilities of the fuel pins. When the fuel pins experience either overpower or undercooling conditions some fuel melting may be expected. Because the fuel undergoes a significant volume increase upon melting and releases contained fission gas, large pressures may be generated in the region of the molten fuel. These pressures may be high enough to rupture the cladding. The pressures will not be generated, however, if void volume is available in the pin to accommodate the fuel volume increase and the released fission gas. A fabricated central void would provide the volume required.

Economic Incentives

Estimates of the changes in the mean annual costs due to the incorporation of annular pellets were made by exercising the fault trees. For the purposes of this analysis, all probabilities of transitions to burst failures were reduced. Spontaneous burst failure frequencies were also reduced.

The economic impact of reducing the probabilities of generating burst failures and spontaneous burst failure event frequencies were shown in Table 3-4 and repeated below. The plant mean-annual-costs associated with fuel failure effects, flow blockages, and reactivity transients were reduced substantially relative to the base case costs. If the probabilities can be lowered by a factor of two, it is predicted that these safety outage related costs will decrease by almost two million dollars annually. The driving force which causes the expulsion of fuel from a vented pin into the coolant may be entirely removed if the annular pellet configuration performs properly. If this is the case, the burst failure probabilities may be reduced on the order of a factor of ten or more. Over \$2,500,000/yr may be saved in this case.

The above discussion points out the potential economic benefit which may be obtained from the use of annular pellets rather than solid pellets. Such gains must be compared with the attendant economic changes associated with the normal energy production costs of the reactor.

Effect of Reduction of 'Burst' Failures

Probability Modifier	1.0 (base case)	1/1.1	1/1.5	1/2	1/10
Safety Related Mean Annual Cost Change (\$10 ³)	0	-459	-1442	-1935	-2645
Energy Cost Change mill/kwh	0	-0.07	-0.21	-0.28	-0.38

Annular pellets are slightly more complicated to fabricate than are solid pellets. It is likely that the annular configuration can be obtained with the appropriate dies in the pellet pressing process. The possibility exists however that the central hole will have to be drilled rather than formed during the pressing. This would add significantly to the pellet fabricating costs. On the other hand, the presence of the central hole may reduce the time required for sintering the pellets. The net incremental fabrication costs due to the use of annular pellets is expected to be quite small and essentially negligible.

The reference GE 1000 MWe LMFBR design (GEAP-5678) employed a fuel smeared density of 85% of theoretical. This same density can be achieved with annular pellets with an outside diameter of about 0.225 inch and an inside diameter up to 0.050 inch. With this configuration the actual pellet material would not have to be any higher than 94% of T.D. The axial blanket fuel has a higher smeared density than that of the core. Fabricating the blanket pellet with a central void would likely necessitate some reduction in the smeared density. The decrease will be small, however, if the inside diameter is kept small. For example, a pellet fabricated with an outside diameter of 0.225 inch and an inside diameter of 0.020 inch has only about 1% less fuel volume than a solid pellet of the same outside diameter. A one-percent loss of blanket material would have a very minute effect on the core performance and economics.

The effect of employing annular pellets in the core and axial blankets is expected to increase the fuel cycle costs by a very small amount, if at all. Fabrication costs may increase by as much as five percent. For typical LMFBR's this component of the fuel cycle costs is expected to range from about 0.20 to 0.45 mills/kwh. The use of annular pellets, therefore, might possibly increase the net fuel cycle costs on the order of 0.01 to 0.02 mills/kwh.

The above fuel cycle cost penalty is more than offset by the predicted annual savings of nearly \$2,000,000. For a 1000 MWe plant with a load factor of 0.8, this savings translates to an energy cost reduction of about 0.22 mills/kwh. This is a factor of ten greater than the possible increase in fuel cycle costs resulting from the use of annular pellets. On the basis of this analysis, it appears that the incorporation of annular fuel pellets into LMFBR fuel pin designs is worthy of consideration.

Informational Requirements

A better understanding of the mechanisms potentially leading to the release of fuel from a ruptured pin is necessary if the proper design steps are to be taken to minimize the frequency and consequences of its occurrence. If central void pressure is the major cause of fuel escaping from a pin then it is possible that the use of annular pellets in combination with low internal pin pressures could significantly reduce the likelihood of fuel expulsion. Passive fallout, on the other hand, may be very difficult to control. It is possible that fuel form (whether powder or pellet) could have some influence on this type of release.

Recommendations

The debris and geometry changes resulting from pin failures have the potential for causing failures in other pins. A systematic procedure for characterizing the amount and size distribution of the debris and the magnitude of pin geometry change should be implemented in connection with the post-irradiation examination of fuel pins.

At present, the build-up of fission gas in the central void to high pressures is hypothetical. The instrumentation of a test pin so that the pressure in the central void can be monitored is a method for the determination of the existence of this driving force for fuel expulsion from a failed fuel pin.

The in-pile test program of failed fuel, PA-10 Task B, is planning a test involving a burst failure. This should help resolve the current uncertainty of whether fuel debris is ejected when a pressurized pin fails. Continuing tests of this series should consider tests to failure with the alternate fuel designs of powder fuel and annular pellets for comparison with the reference solid pellets. The key question of whether these tests should be single pin tests to observe fuel debris after failure or whether multiple pin tests to observe directly the possible propagation of failure by debris blockage has not been addressed in this study. Intuitively, multiple-pin tests of propagation would find the answers faster and probably cheaper.

3.3.4 Fuel Pin Spacing and Spacer Design

Fuel pin spacing can influence pin-to-pin failure propagation in at least two ways:

1. For typical wire wrap LMFBR fuel assemblies increasing the pin spacing reduces the probability of debris being trapped between the pins.
2. Larger pin spacing reduces the likelihood that a distorted or swollen fuel pin will restrict coolant flow to the extent that adjacent fuel pins may be overheated.

The influence of pin spacing on these two problem areas may be estimated from the work reported in Section 6. In Section 6.2.1 a study was made of the fuel particle sizes which may be released from a fuel pin when its cladding fails. The particles entering the coolant stream will proceed along the flow path until they are trapped or escape the fuel bundle. The size of the particles which will be caught depends on the size of the trapping sites present in the fuel assembly. Section 6.2.2 explores flow passage dimensions as a function of pin diameter, pin pitch-to-diameter ratio, and fuel pin spacer type. For wire wrap spacers the flow path cross-sectional area increases with both larger pins and larger P/D's. For the grid spacer studied the area of concern varied only with pin diameter.

The assumption can be made that the size of fuel particles which enter the coolant stream from a ruptured pin will be fairly independent of pin diameter. If this is so, the likelihood of particles being trapped within the fuel pin lattice decreases as the trapping site size increases. Fewer local flow blockages should result, along with fewer overheated pins and, consequently, a reduced number of fuel pin failures.

The reference LMFBR employed a pin diameter of 0.25 inch and a P/D of 1.2. The work in Section 6 indicated that the potential for particle trapping may possibly decrease by a factor of three if the P/D is increased from 1.2 to 1.28. This can be interpreted to mean that the number of pin failures caused by debris trapping can be significantly reduced by such changes.

Fuel-sodium reaction product swelling is not well characterized at this time. It is likely, however, that the overheating problems caused by this phenomenon can be reduced by employing larger spacings between pins. For a given amount of swelling the flow disturbance will be less and the adjacent pins will undergo less of a temperature increase.

These arguments form the basis for reducing certain fuel pin failure probabilities when large pin spacings are employed. Quantifying the probability change due to the design modification is difficult. However, with the present knowledge it is not unreasonable to expect that a P/D change from 1.2 to on the order of 1.3 for the reference design would reduce the associated pin failure probabilities by a factor of two or so.

Economic Incentives

The safety related cost savings which may potentially be realized by employing larger pin spacings is tabulated below as a function of the reduction in the failure probabilities. The equivalent effect on energy generation costs are also shown.

Probability Modifier	Base Case	Increased Pin Spacing			
	1.0	$\times \frac{1}{1.1}$	$\times \frac{1}{1.5}$	$\times \frac{1}{2}$	$\times \frac{1}{10}$
Safety Related Mean Annual Cost Change (10^3 \$)	0	-263	-963	-1,444	-2,599
Energy Cost Change mills/kwh	0	-0.0376	-0.138	-0.206	-0.371

These values show that a reduction in the associated failure probabilities by a factor of two (probability modifier of 1/2) can result in a very substantial reduction in the plant operating costs.

As pointed out previously increased pin spacing can be achieved both by increasing the pin diameter (at the same P/D) and/or by increasing the pitch-to-diameter ratio. Changes in pin diameter, however, are less effective than P/D variations in changing the probability of pin failures due to particle trapping or other overheating effects. This being the case, it will be assumed that changes in pin spacing will result from changes in pin pitch rather than pin diameter.

The work reported in Section 6.2.2 indicated that particle trapping in a grid spacer fuel assembly design is considerably less sensitive to pin spacing than is trapping in a wire wrap design. Any reduced fuel pin failure probabilities due to the decreased likelihood of particle trapping and the resultant local flow blockages, therefore, are applicable only to the wire wrap design. However, both spacer configurations would benefit from a larger pin spacing due to the reduced likelihood of pin overheating from the fuel-sodium reaction product swelling.

Core performance may be influenced substantially by changes in the fuel pin spacing. The neutronics may deteriorate somewhat because of greater neutron leakage from the core. On the other hand a larger spacing between pins reduces the fuel assembly pressure drop. Plant efficiency rises because of the lower pumping energy requirements. Alternatively, longer cores may be used while adhering to the same pressure drop limitations.

Fuel cycle costs will vary somewhat with these changes in the core design and performance. These effects were studied in many of the LMFBR Follow-On studies. GEAP-5618^{1,2} reported a fairly shallow influence of pin spacing if the pressure drop was held constant by varying the core height. At a constant core height, however, the fuel cycle costs increased by about 0.1 mills/kwh in going from a 1.2 to a 1.3 pitch-to-diameter ratio. Pin diameter was held constant at 0.25 inch in these calculations. BAW-1328(V.5)^{1,3} showed about 0.05 mills/kwh cost increase for the same change in pin spacing at the same pin diameter. Both of these results were obtained while holding core height constant. Allowing core height to vary so as to maintain the same pressure drop limits will reduce the cost penalty to less than 0.05 mills/kwh. Proper optimization of the core design parameters at the new pin pitch would likely result in a very small,

and probably negligible, increase in the fuel cycle costs. Assuming that the conservative assumptions of debris blockage potential are valid, a comparison of the economic effects of employing larger pin spacing strongly suggests that this design feature will be economically advantageous.

Information Requirements

A quantitative assessment of the importance of pin spacing in its effect on core safety is presently not available. Research and testing must be done to remove the uncertainties which exist. These efforts should be directed at answering the following questions:

- What is the quantitative effect of fuel pin spacing on the likelihood of debris being trapped within the fuel pin matrix?
- To what extent does spacing affect the potential for fuel pin overheating due to bowing or distortion of a pin (including the effects of fuel-sodium reaction product formation)?
- How does spacing and spacer design affect the transport of debris beyond the immediate neighbors of a failed pin and hence affect secondary fuel failure propagation?

Concerning debris trapping, investigations must be directed toward an assessment of debris characteristics. The size and size distribution of fuel particles, clad, or spacer debris which may enter the coolant stream must be determined. This information must then be compared with the results of investigations aimed at defining the trapping potential within a given fuel pin arrangement.

The sites within a fuel assembly at which debris in the coolant stream may be trapped are largely associated with the spacer configuration. This was reviewed in Section 6.2.2. In addition, the type of spacer employed has been shown to affect the extent to which adjacent coolant streams mix. A high degree of mixing will serve to limit the consequences of a local flow blockage or disturbance. The selection of a spacer design, therefore, must factor in these considerations.

Two basic types of fuel pin spacers are presently being considered for LMFBR use. These are the wire wrap and the grid type spacers. Studies have indicated that wire wrap spacers should be less likely to trap debris (see Section 6.2). In addition, wire wraps appear to promote more coolant mixing than do grid spacers (see Section 6.1). Experimental evidence must be gained to identify the differences in the performance characteristics of these two spacer types.

Recommendations

Multiple pin (7 pin array) propagation tests are needed to determine the propagation potential beyond the immediate neighbors of a failed fuel pin. An edge pin could be intentionally ruptured (like PA10-Task B pins) and the integrity of the pins across the array could be monitored with fission gas pressure transducers to indicate if and when failure occurred. Successive tests with grid and wire wrap designs would show their relative potential for mitigating the possible effects of fuel pin failure.

The ORNL Fuel-Failure Mockup Program Plan describes a series of flow blockage experiments. The likelihood of debris trapping and the thermal effects of this trapping can be investigated in this program. The measurement of interchannel mixing in the tube bundles should also be investigated as a function of spacer design for correlation with analytical models. The ANL transparent test section or similar facility might be used to artificially inject fuel debris into the fuel bundle to note whether fuel particles are trapped and if so, where and to what extent. Successive tests with alternative pin spacings and spacer designs would establish the effect of those design parameters.

3.3.5 Fuel O/M Ratio and Sodium Purity

The fault tree analysis pointed out that one of the dominant mechanisms whereby fuel pin failures may propagate is from the formation of a low density fuel-sodium reaction product.* This product swells relative to the fuel

*Analysis (P.A. 53) completed since this work was done indicates that fuel-sodium reaction product swelling is limited by oxygen availability. A small clad defect apparently will admit sufficient sodium to form the reaction product and cause swelling. As soon as a significant size crack is formed (~ 15 mils), however, oxygen diffusion from the fuel to the sodium is sufficient to deplete the oxygen supply and to stop the reaction. Fuel swelling is thus limited and does not appear to be a serious problem.

volume involved. In this way it may serve to restrict coolant flow to pins adjacent to the one which has failed and hence to propagate the failure. Eliminating or reducing the formation of this reaction product would reduce the number of fuel pin failures to be expected from oxide-fueled LMFBR's.

The fuel-sodium reaction is not well understood at this time. Its formation is believed to be somewhat dependent on the amount of oxygen present. High oxygen concentrations appear to promote the reaction. This suggests that reducing the oxygen content of the system could serve to retard the formation of the fuel-sodium reaction product.

Low oxygen concentrations can be obtained by utilizing hypostoichiometric fuel and by maintaining coolant purity. Most LMFBR designs propose to restrict the oxygen level in the coolant to 10 ppm or less. This may be low enough to prevent the formation of a significant amount of reaction product.

Economic Incentives

A reduced amount of swelling primarily affects those transition probabilities in which a large area failure propagates to additional failures in adjacent pins. The effect of lowering these probabilities was shown in Table 3-4 and summarized below. Estimated cost savings range from \$250,000/year to more than \$2,000,000/year, depending on the degree to which the transition probabilities can be lowered by taking steps to prevent the formation of the fuel-sodium reaction product.

Effect of Reduced Propagation Due to Fuel-Sodium Reaction Product Swelling

	Base Case				
Probability Modifier	1.0	1/1.1	1/1.5	1/2	1/10
Safety Related Mean Annual Cost Change (\$10 ³)	0	-238	-820	-1931	-2094
Energy Cost Change mill/kwh	0	-0.03	-0.12	-0.28	-0.30

The assessment of any effects on plant capital costs or fuel cycle costs resulting from the steps taken to reduce swelling is difficult at this time. Not enough is known about the ways in which this failure propagation mechanism may be mitigated. Simply reducing the oxygen concentration in the system may be an adequate measure. If this is the case, no cost increments are foreseen. Fuel O/M ratios are fairly easily controlled, and the fuel preparation costs are not expected to increase if hypostoichiometric fuel (down to 1.93 O/M) is required. Fuel costs might be adversely affected if a tight tolerance on fuel O/M were necessary to prevent swelling. Similarly, present LMFBR designs call for low oxygen concentrations in the coolant for reasons other than the prevention of this swelling phenomena. The designs already require sodium purification equipment to maintain acceptable oxygen levels, and no additional equipment is needed.

The potential cost savings to be gained by controlling or preventing the fuel-sodium reaction provide a significant incentive to continue research in this area.

Information Requirements

The current uncertainty over whether fuel-sodium reaction product swelling will occur in a reactor environment must be resolved. If it is found to occur, the sensitivity to fuel O/M ratio must be determined to permit selection of this design parameter.

Recommendations

In-pile tests are needed in order to characterize and understand fuel-sodium reaction product swelling. This phenomenon is being investigated through a series (designated B9) of in-pile tests being conducted under the PA 10 program. In these tests, recent emphasis has been placed on control of the sodium purity in the capsule. This is necessary in all defected fuel testing because of the potentially large effect of coolant oxygen level on the reaction

product formation. If successful in controlling oxygen level, these tests will remove the uncertainty of the first tests where oxygen content was uncertain. If swelling is found under simulated reactor conditions, the sensitivity to fuel Q/M should be investigated. Of secondary priority would follow tests varying the type of failure (crack or large area). Lastly, an in-pile fuel-sodium reaction product swelling test of a diving bell vented pin would be needed to determine the possible influence of sodium circulation up from the rupture and out the vent and to determine if the vent and/or fuel column plugs, forcing gas and possibly fuel out the clad rupture.

3.3.6 Channel Containment Requirements

The ability of the channel wall to resist the hazards imposed upon it by the reactor environment is an important safety consideration. Many questions concerning this ability have not yet been answered.

Failure of the channel wall could have several consequences. Any opening in the wall will allow coolant to flow between the inside and the outside of the channel. The amount of flow depends largely on the size and characteristic of the opening, as well as the pressure differential existing from the inside to the outside of the ruptured wall. If this leakage flow is significant, the sections of the fuel pins downstream of the rupture will be under-cooled. This, in turn, could lead to further fuel pin failures and still other mechanisms for propagation of the failure to adjacent channels. The undercooled pins might melt, resulting possibly in at least a partial fuel slumping condition. In addition, the molten fuel might interact violently with the coolant, producing substantial shock waves or pressure pulses. Melt-through of the channel wall and the walls of adjacent assemblies is also possible.

Another potential consequence of a ruptured channel wall concerns the removal of the damaged bundle from the core. If the damage is such that the wall is distorted significantly, it may interfere with adjacent channels. Special equipment and techniques may be required to remove bundles in this condition.

This containment of propagation forces is dependent upon the response of both the pressurized channel and its neighbors. The significant failure mode of the adjacent channels has yet to be defined. Qualitatively, it is believed to be when significant fuel pin failures occur in adjacent bundle by crushing. That is, significant channel deformation (without necessarily breaking the channel) could constitute the measure of failure. This is especially true for sealed pin designs, where simultaneous failure of many pins could lead to voiding and meltdown of the adjacent bundle. The pin failure mechanism is by excess spacer contact forces, pointed out in previous sections as requiring development effort to establish the failure load.

Another deformation limit may be associated with excess force on control rod guide tubes. This was emphasized recently in the ANL EBR-II Mark E channel deformation tests.

Properly designing the channel to minimize both the frequency and consequences of failure require that additional design information be obtained. Analysis of channel deformation due to external and internal loads has shown the current hexagonal design to be a very flexible structural member. Evidence is growing that the consequence of fuel failure and loss of flow incidents may be significant channel deformation and possibly fuel failure propagation. Whether this will be an intolerable situation has yet to be determined. The frequency of initiating events and probability of progression are too uncertain at this time to warrant design recommendations. Alternatives, however, can be discussed.

Since the current understanding of fuel pin response to shock and pressure loads shows little cause for concern, the concept of a perforated channel gains credibility. Any pressure disturbance in the core is not constrained to a channel but simply passes through the core leaving the fuel pins relatively undisturbed. The principal uncertainties are:

1. The response of the relatively large and stiff control rod guide tubes to unimpeded pressure waves, and
2. The parallel channel flow response under boiling perturbations involving the entire core.

The opposite design solution is also credible, that of using strong circular channels capable of containing expected pressure loads. This solution possibly involves significant fuel cycle cost penalties due to the wasted space between circular channels ($\approx 9\%$ of the core).

This possible influence of safety related developments on channel design is very uncertain and has the potential for significant core design implications. This question should be pursued and placed in perspective as quickly as possible.

Economic Incentives

Variations in the probability of channel wall failure have a relatively small effect on safety related core costs. This is shown by the results of the fault tree analysis presented in Table 3-4 and repeated below. Events wherein any channel walls are damaged were judged to be quite costly. The probability of such events occurring, however, is low. This yields a low frequency of occurrence, with the result that the safety related mean-annual-costs associated with channel wall failures are of secondary importance.

Effect of Increased Channel Containment Capability

Probability modifier	1.0 (base)	1/2	1/10	2	10
Safety Related Mean Annual Cost Change (\$10 ³)	0	-344	-619	717	5873
Energy Cost Change mill/kwh	0	-0.05	-0.09	0.10	0.84

Channel wall failure probabilities may be influenced by several design parameters. Among these are variations in the channel wall thickness, the pressure differential across the channel wall, and the gap between channels. Making the walls thicker would make them more capable of coping with any loads the channel may be subjected to. Overheating problems can be minimized by employing larger gaps between adjacent fuel assemblies. In addition to these measures, an attempt can be made to reduce the pressures acting on walls by employing a pressure balancing scheme between the inside and the outside of the channel.

Employing all of the above techniques may reduce the probability of channel wall failure by a factor of two or less. It is unlikely that a factor of ten could be achieved without resorting to rather drastic measures. Attaining a factor of two reduction would yield a predicted cost savings on the order of \$350,000 relative to the base case mean-annual-costs or an equivalent $\sim .04$ mill/kwh.

The reference core design employed in the fault tree analysis utilized a nonfreestanding channel wall design.⁶ This allowed the use of rather thin channel walls with very little clearance between fuel assemblies. Present considerations of the volume increase of steels due to irradiation effects made certain features of the above design impractical. A substantial gap between adjacent channels may be necessary to accommodate the expected swelling. A freestanding channel wall design may also be more practical in this situation. Both of these measures degrade the core performance in that the neutronics suffer. Fuel cycle costs increase as a result.

The reference design LMFBR employs a channel wall thickness of 0.060 inch. If this were doubled to 0.120 inch in order to reduce the probability of channel wall failures, the steel volume fraction in the core would increase by slightly more than .04. This is about equivalent to increasing the clad thickness by four or five mils. From GEAP-5710¹¹ the fuel cycle costs would be expected to increase by about .05 to .10 mill/kwh, depending on the economic basis assumed.

Large gaps between adjacent fuel assemblies may be necessary in order to accommodate steel swelling. These gaps will reduce the probability of channel wall failure in that the gap coolant will serve to reduce the potential for channel wall overheating due to trapped fuel debris or distorted fuel pins. It is estimated that increasing the gap spacing from 0.030 inch to 0.25 inch will increase the fuel cycle costs by about 0.05 mill/kwh.

The extent to which the channel wall thickness and gap spacing would have to be changed to reduce the probability of channel failures by a factor of two is uncertain at this time. It appears, however, that such changes will result in fairly substantial increases in fuel cycle costs. These costs may not be offset by the potential savings in the safety related costs. The fault tree analysis indicates that relatively small returns may be expected from improvements in the channel wall integrity. Development efforts aimed strictly at improving its performance during abnormal conditions, therefore, must be assigned a relatively low priority.

Information Requirements

There is a relatively high priority need to establish the functional criteria for channels under accident conditions.

Are channels expected to contain accident pressures and energies?

If the answer is yes, there is then the need to establish the channel containment capabilities.

Recommendations

Presuming channels will be expected to contain certain accident conditions (or the information is needed for safety and licensing discussions) a series of tests is recommended.

Confidence in current analytical predictions is not intuitively good enough, requiring confirming empirical tests. The recommended tests are of three types:

- pneumatic tests to simulate sodium vapor expansion or fission gas release
- hydraulic tests to simulate sodium hammer type loads
- explosive tests to simulate postulated violent boiling type loads

Major test variables are pressure history (magnitude, duration and shape) and temperature history (magnitude, ramp rate).

Major channel variables are:

- wall thickness
- ductility (1-40%) simulating possible fluence effects
- perforated or solid channels

The tests should be run in a core mockup with fuel pins included. A water bath would be economically attractive but confirming tests at high temperature in sodium would be necessary if accurate results proved necessary. This luxury seems unnecessary at this point.

Measurements made on these tests should include:

- channel strain, deflections and rupture loads
- fuel pin strains, deflections and rupture loads

The propagation of fuel failures to adjacent bundles without necessarily rupturing the channels can be observed directly with the tests. If suitably representative fuel pins and spacers can be simulated, the failure point of pins in adjacent channels might be established.

The expected test results are generally the channel rupture loads (pressurized and adjacent channels) and fuel pin failure propagation loads.

In conjunction with violent boiling and reentry tests, these tests provide:

1. assurance that current channel designs are adequate to prevent propagation of local failures to adjacent fuel channels or to prevent damage to adjacent control channels.
2. or provides the basis for increasing channel wall thickness or interchannel spacing to provide the above assurance.
3. or shows that channel-to-channel propagation potential is large, requiring operating mode and instrumentation selection to reduce the probability of the initiating events.

3.4 SAFETY RESEARCH NEEDS

Most of the accident chains comprising the fault trees include some events which may not be significantly influenced by design features. These hypothesized failure propagation modes (involving sodium boiling and clad/fuel-sodium thermal interaction) could have a significant effect on the reactor safety related economics. Sensitivity analyses of these phenomena are useful for defining their relative importance. Those areas which are shown to be important then justify a fairly high research priority.

The results of the fault tree sensitivity analyses aimed at defining research priorities are discussed next.

3.4.1 Sensitivity to Failure Propagation Phenomena

The existence of failure propagation phenomena involving sodium boiling and clad/fuel-sodium thermal interaction (also referred to as violent boiling) may potentially have significant economic implications on the plant

operation. These areas were not included in the design sensitivity study because the mitigation of their effects may not necessarily be influenced by core design changes. Large uncertainties currently surround these phenomena and failure modes. The propagation probabilities assigned to the events involving these phenomena, therefore, are also quite uncertain. Exploring the appropriate cost sensitivities is useful in defining both the importance of these phenomena and failure modes and the research which is aimed at gaining a more adequate understanding of them. The areas of interest dealt with here are as follows:

1. The propagation modes involving sodium boiling and violent boiling which follow fuel failure propagation;
2. The propagation modes involving sodium boiling and violent boiling for all initiating events.

Sensitivity studies of these areas were made using the fault tree analysis. The appropriate transition probabilities were changed to model the desired effect. Initiating event frequencies, however, were held constant at the levels assumed for the reference case. The results of these analyses are presented in Table 3-6. The following discussion elaborates on the analyses.

1. Fuel failure propagation within and beyond the channel walls by sodium boiling and violent boiling mechanism. Of concern here are the transition probabilities whereby fuel failures can lead to channel wall failures and/or the failure of pins in adjacent fuel bundles. Table 3-6 indicates the mean annual costs increase by a factor of about six when the propagation likelihoods are increased by a factor of ten. Reducing the probabilities by a factor of ten cuts the safety related costs in half. All end states except S7, the removal of one sound bundle, are affected by these probability changes, particularly by increasing probabilities that postulated sodium boiling and violent boiling mechanisms are real. These results show the need to establish the credibility of these mechanisms.
2. Propagation modes possibly involving sodium boiling and violent boiling. This sensitivity analysis varies the transition probabilities mentioned for the above case, but it also included those probabilities involving loss of coolant accidents. The results of this addition are seen to be negligible. That is, the assumed frequencies of loss of cooling accidents, compared to the initial frequency and probability of propagation of fuel failures is too small to be significant.

Because of assumed frequency and cost of events, the dominant cut set for damaging a channel wall always involves fuel failure propagation. Cost penalties change almost in proportion to the probability of fuel failures progressing to fuel meltdown. These penalties are relatively independent of the probabilities associated with bundle loss of flow incidents progressing to channel failure because of the low assumed frequency. With our present understanding, it is to be expected that frequencies of fuel failure incidents will vastly overshadow flow blockage type accidents. Thus the present assumptions, and their consequences, are realistic.

3.4.2 Sodium Boiling Research

Several key questions were identified in this study which bear on sodium boiling research:

1. What conditions lead to local boiling?
2. Is local boiling stable?
3. What are the damage mechanisms due to boiling? Vapor bubble collapse? Sodium hammer?
4. What are the boiling burnout conditions for loss of flow or local blockage accidents? What is the time sequence of clad and fuel melting?
5. Are voiding and slumping reactivity rates significant?
6. What is the final condition of a melted fuel assembly?

There is basic uncertainty about the answers to these key questions. Possible core design significance can be found in answers to the first two questions. The pin pitch and core length probably will influence the boiling threshold and stability, with open lattices and short cores more tolerant of blockages. Instrumentation choices and operating modes might also be affected.

Sodium boiling damage mechanisms may influence the propagation rate of local disturbances. Large vapor bubble collapse damage potential could imply fuel pin rupture beyond the radius of significant thermal perturbation and hence speed the propagation rate of fuel damage and sodium boiling. Similarly, sodium hammer is a possible mechanism to cause rapid channel to channel damage propagation. The choice of fission gas accommodation appears the principal design variable possibly affecting the vapor bubble collapse and sodium hammer mechanisms.

Table 3-6. Fault Tree Sensitivity Study

Final State		Base Case Mean Annual Cost (in 1000's)	Fuel Failure Propagation				All Boiling Propagation Modes			
			100x	10x	1/10x	1/100x	100x	10x	1/10x	1/100x
continue operation with failures	- S6	832	4081	1127.8	802.9	799.9	4084.9	1129.6	802.7	799.7
	one - S7	519	518.7	518.7	518.7	518.7	518.7	518.7	518.7	518.7
	few - S8	842	10,232	8412.9	84.3	8.6	10,238.2	8414.8	84.1	8.4
remove sound bundles	many - S9	112	10,507.8	1056.9	17.3	7.8	10,520.7	1064.5	16.5	7.0
remove unsound bundles	one - S10	510	17,481.4	5084.6	53.0	7.2	17,481.4	5084.6	53.0	7.2
	few - S11	108	10,187.7	1024.2	16.2	7.0	10,192.6	1029.1	15.7	6.5
	many - S12	91	3498.9	401.2	43.9	39.1	3515.5	417.8	25.7	19.1
Total		3014	56,508	17,626.2	1536.2	1388.4	56,552.0	17,658.9	1516.3	1366.6

GEAP 13639

The last three questions concerned with the details of sodium boiling involve burnout, melting and fuel/clad dispersal. No prominent design parameters are obvious as affecting these latter stages of a boiling accident. An accident progressing to this stage is of concern from the standpoint of bundle-to-bundle propagation and reactivity effects (only after sufficient numbers of bundles are involved). The most likely bundle-to-bundle propagation mechanism is molten fuel/clad-sodium thermal interaction, hence the interest in the sodium boiling details (principally cooling rates beyond the DNB) as precursors to possible propagation.

Note is made of the significance of possible void fractions. Generally sodium boiling burnout would involve very high void fractions (>95%). This implies that the most likely conditions for violent boiling begin with <5% liquid sodium, supporting the recommendation for inclusion of the vapor phase in violent boiling models. This applies particularly to loss of cooling and local blockage accidents but not necessarily power transients, when the fission gas would be the principal gaseous phase, at least initially.

Sodium boiling hydrodynamics certainly will dominate the study of core meltdowns involving significant reactivity perturbations, whether from whole core accidents or bundle-to-bundle propagation.

The last question concerns the final state reached by a fuel assembly meltdown accident. The cooling rate during meltdown will likely control the final fuel location. The final disposition of fuel on a local bundle meltdown would hopefully be out of the core but still within the fuel assembly. In this regard, certain core design features may be influenced. This leads to possible configurations of the lower fuel assembly involving strainers, heat sinks and/or molten fuel catchers.

Recommendations

1. Boiling Threshold

This problem area is currently being studied by ORNL¹⁴. Plans are to use an existing sodium loop to simulate local blockages in a fuel bundle. Sufficient correlation with flow mixing models (COBRA II) will be necessary to extrapolate to the reactor situations of interest. The principal extrapolation will be the type of blockage as the most likely in-pile blockage is heat generating fuel and the ORNL tests will use non-heat generating blockages. The confidence level for attaining the required correlations is high because these tests and correlations are within the state-of-the-art technology. Pending results from this program, there are no further recommended programs in this area.

2. Boiling Stability

Out-of-pile local boiling tests in simulated fuel bundles are recommended to answer the question of boiling stability. Of prime concern is the test pressure as the question of stability should be sensitive to the vapor-liquid density ratio and most existing sodium test facilities are not capable of operating at the pressures of interest (20-50 psia)

ORNL's fuel element failure propagation loop facility (see above) will be operated in the boiling mode at the end of each single phase sodium test of the temperature perturbations due to local blockages. These boiling tests might provide answers to the question of local boiling stability.

The confidence in attaining the required test information is low because of the uncertain lifetime of conventional heaters at these temperatures. If only single, short boiling runs are achieved for each test section, the required confidence in predicting boiling stability may not be achieved.

In addition, the heater temperature limitation may not permit boiling in the pressure range of interest.

The development of high power, high temperature heaters at AI holds some promise for producing the required heater, particularly the UO_2 heater. The current testing involves three heaters for investigating boiling hydrodynamics at high power but relatively low pressure (8 psia) and in essentially steady state conditions. For these reasons, it is uncertain whether these hydrodynamic studies will provide the required insight necessary to answer the questions of local boiling stability.

Modeling of local boiling stability will require a new analytical tool. A transient code capable of multichannel calculations with flow blockages, cross flow mixing, boiling hydrodynamics and heat transfer and fuel and clad heat transfer should be developed to correlate the experimental results and scope reactor boiling incidences.

A large multi-pin array (19 or greater) appears necessary to explicitly demonstrate stability (or lack of stability). It appears credible that the current programs can answer the question of local boiling stability. It would seem to require the following:

- a 19 pin or larger test section
- operation at realistic pressures (20-50 psia)
- reusable heaters capable of clad temperatures near and perhaps above 2000°F.

The cost of these tests is clearly dependent upon the successful developing of reusable heaters. If such development were unsuccessful, an in-pile test would be the only alternative. Such a test could be conducted in ANL's proposed 19 pin ETR loop or in SEFOR. In this case, however, the question of boiling stability would be only a minor objective since the expense of an in-pile test would require better utilization of in-pile capabilities, namely study of fuel meltdown and possible violent boiling. The in-pile test is not specifically recommended to answer the question of boiling stability because it appears technically feasible to determine the answer at much less expense out-of-pile.

3. Sodium Hammer and Vapor Bubble Collapse

The review and analysis of sodium hammer indicates that it is a potential contributor to channel-to-channel failure propagation. The probability of sudden and complete condensation of the sodium vapor bubble when a liquid slug re-enters a voided bundle is considered to be very low, but the consequences are serious enough to warrant some attention during sodium boiling tests.

Sodium boiling test programs must make every effort to simulate the reactor environment and geometry to establish credibility in predicting sodium boiling consequences like sodium hammer. Pressure instrumentation must be capable of detecting the magnitude and duration of pressure pulses produced by bubble collapse and sodium hammer. Since intervening two-phase conditions may muffle these pressures, many transducers at various elevations appear necessary. For multi-pin boiling tests, the channel should be simulated, allowing a soft response to sodium hammer and permitting a direct correlation of channel damage potential due to boiling.

An out-of-pile test of channel containment capabilities for sodium hammer is recommended, using water hammer techniques. Reentry sodium hammer can be simulated with an evacuated channel and a rupture diaphragm in a water test. The fuel pins and surrounding fuel bundles can be simulated in a water tank to represent the core. This test was detailed in Section 7.2.

Analytical treatments of void collapse problems are also needed. Dynamic solutions, particularly in the radial direction, are necessary, not only to determine the magnitude of pressures due to sodium hammer, but to simultaneously determine the response of adjacent channels and fuel pins.

Likewise, vapor bubble collapse during local boiling has not been discounted as a possible contributing propagation mechanisms, but is considered a low priority research item, primarily because local boiling is likely to be hydrodynamically unstable. If local boiling around small blockages were found to be stable and undetectable, the vapor bubble collapse damage mechanisms would be elevated in relative importance.

Experimentally, evidence of vapor bubble collapse damage might be obtained from any test involving sodium boiling. These include the PA-10 C-2 series, and the ANL, PNL, and ORNL Laboratory boiling tests. One specific test at ANL, the fission gas blanketing test, could perhaps be modified to inject sodium vapor into a pin array rather than fission gas. This would permit a direct measurement of vapor bubble collapse damage

potential and subsequent design oriented studies if damage were found to be significant. Design features such as spacers, fission gas pressure, clad ductility and clad strength could be varied.

4. Boiling Heat Transfer

Considering questions 4-6, boiling burnout, voiding, slumping, and final fuel disposition, the key uncertainty concerns the cooling rates possible with sodium boiling. Correlations of transient sodium boiling hydrodynamics and heat transfer would be necessary in addition to a powerful transient accident code, to analytically predict accident consequences involving fuel and clad melting. Current accident codes being developed, SAS-1A and FREADM in particular, include some elements necessary for this prediction. However, the probability of their success as predictive accident codes is considered very low for the near term. As scoping tools to explore possible accident responses they should be adequate. It is recommended that these accident codes be used simultaneously to study reactor accidents and to correlate in-pile and out-of-pile experiments. The sodium boiling portions of these codes should be continually updated to incorporate the latest correlations of boiling hydrodynamics and heat transfer.

Out-of-Pile Tests

The generation of experimental sodium boiling heat transfer data out-of-pile suffers the limitations discussed in the previous section, namely high temperature heaters and multi-pin arrays. In addition, data is necessary past clad and fuel melting. The only proposal in this area is AI's UO₂ heater used to simulate a power transient, and this suffers from power shutdown on failure, whereas this isn't necessarily the case in reactors. Although not discussed in the literature, of greater interest is use of the high temperature AI heaters to simulate loss of flow accidents involving clad melting and slumping before fuel melting begins. This appears technically feasible and would begin to bridge the gap between state of the art boiling tests and in-pile loss of flow meltdown tests which are now planned. Extension of out-of-pile boiling tests into transient, high temperature tests involving melting is recommended. Radial and axial simulation of the reactor environment is recommended to make the tests directly applicable to reactor accident studies. If successful, the out-of-pile tests could scope a broad range of accident conditions of interest at less cost than the in-pile tests.

In-Pile Tests

In-pile tests have the potential of showing directly the consequences of certain reactor accidents. Integral tests involving all aspects of an accident are recommended. In this regard, no specific sodium boiling in-pile tests are recommended. Rather, the planned accident simulating in-pile transients should be designed and instrumented to extract the available data about specific phenomenon as well as the direct accident consequences. High temperature clad and fuel thermocouples would permit correlation of sodium boiling heat transfer rates. Care should be exercised to simulate the heat sinks available in a fuel bundle. A nonprototypical heat transfer path could severely complicate the interpretation of the results.

These in-pile tests, notably PA-10 Task C-2 loss of flow tests, will necessarily be concerned with boiling heat transfer and clad/fuel meltdown. These tests should provide a direct answer to the key questions identified. Being integral tests, it is recommended that an early C-2 test be run to complete clad/fuel meltdown to provide guidance to further in-pile and ex-pile tests and model development.

These tests provide, in addition, a wide spectrum of design oriented information, most involving sodium boiling heat transfer. The following are examples of the direct results obtainable from integral experiments:

- A. Tests establish failure thresholds for over temperature in a tube bundle permitting logical choice of an operating or design temperature limit (in conjunction with probability and risk assessment on what range of temperature transients must be accommodated) as a function of design parameters, for instance:

- clad thickness
- fission gas pressure
- spacer design
- bundle edge spacing
- burnup (clad corrosion and ductility)

- B. The consequences of loss of flow influence the channel design. If containment of the consequences of a loss of

flow (due to failure propagation, local blockage or gross bundle blockage) is necessary, tests will show the mechanisms of channel loading (i.e., sodium hammer and violent boiling) as well as integral channel containment capabilities as a function of channel design parameters:

- material
- degree of cold work
- thickness
- support methods (pad location)
- interchannel spacing

(note: this requires simulated channels and/or measurement of radial pressure loads currently included in the C-2 design.)

C. Tests will provide the criteria for selection of instrumentation to detect accident consequences and of operating mode (trip and scram set points) to minimize the effects of the accident. Tests will demonstrate the signal characteristics and response times of candidate instruments: detecting flow, temperature, noise, reactivity, gas and fission product release in the correct environment.

D. A separate use of integral loss of flow tests is for the design of accident accommodation features. If the consequences of loss of flow which cause some fuel damage must be accommodated, core design features may be influenced.

1. post-accident control of fuel location; geometry of the fuel bundle certainly affects the final location of failed fuel, particularly if once molten. Strainers above and below the core might prove necessary to retain particulate fuel in the bundle and prevent gross system contamination.

Molten fuel may tend to agglomerate and could drain out of a fuel bundle unless bundle geometry provided sufficient surface area for solidification and post-accident cooling.

2. If the above fuel containment designs proved ineffective, surrounding components, like the grid plate, vessel and piping may be subject to contamination, requiring cleaning or removal. Integral in-pile tests should determine the necessity of designing from this possibility.
3. Fuel bundle damage or distortion may result, requiring prior design of special handling tools to accomplish swift plant recovery.

E. The information on the time to melt clad and/or fuel in loss of flow or local blockage situations will be derived from the described tests, both in and out of pile. Core design parameters such as pin P/d or core length are expected to have relatively minor influence on the sodium boiling cooling rate which determines the time to melt. Of more importance is the establishment of the time regimes from realistic tests. This in turn helps determine the required instrumentation and operating modes.

F. The question of reactivity rates due to voiding and slumping are related to the question of the magnitude of the DBA and bundle-to-bundle propagation. Loss of safety system response is necessary to involve significant reactivity perturbations, making these very low probability events. Understanding the magnitude of the reactivity rates contributes to establishment of a containment criteria which is acceptable to licensing authorities and to risk assessment. Again it is not clear that design parameters will have a dominating or even significant influence on these reactivity rates. Rather, it is necessary to establish a maximum rate which can be defended with analysis and tests. The out- and in-pile test described under violent boiling provide the appropriate test data of voiding and slumping rates. It is not appropriate to specify the exact test program because in the final analysis, a judgment will be reached between the licensing authorities and contractors that the phenomena of importance to the DBA are sufficiently well understood.

Because the individual phenomena appear extremely difficult to predict, model and incorporate into a comprehensive safety analysis; an integral test program is recommended.

3.4.3 Clad/fuel-Sodium Thermal Interaction

Information Needs

The broad objectives of clad/fuel-sodium interaction studies are:

1. To establish realistic levels of thermal to work energy conversion efficiency in prototypical configurations.
2. To provide such analytical tools as are necessary in order to design for the minimization of failure propagation due to the clad/fuel-sodium interaction mechanism.
3. To provide analytical tools needed to accomplish DBA analysis as a basis for plant licensing.

The aggregate of currently available experimental and analytical data leaves some uncertainty as to whether or not the thermal interaction mechanism is significant in either the damage propagation or DBA context. Analytical estimates range from the admittedly conservative values of 10% associated with Hicks-Menzies analysis to efficiencies of 10^{-5} or less associated with other thermodynamic models based on liquid expansion. A similar range is suggested by experimental results. Laboratory tests such as those of TRW employing shock tubes show that large energy releases result from thermal interaction. More prototypical tests, most notably the S-Series tests of ANL, give stronger evidence of much lower efficiencies on the order of 10^{-5} . If the latter figure is typical then it is not likely that fuel-sodium thermal interactions are of significance in assembly to assembly damage propagation or in DBA situations.

The primary need at the present time is to establish under prototypical conditions the range of pressure generation and work energy which can be expected in an operating LMFBR. In order to provide data which can be extrapolated by the reactor designer, the influences of the following must be determined:

- effects of burnup (this include the effect of burnup on clad and channel wall ductility, the effect of fission gases in mitigating the pressure pulses generated during thermal interaction, and others),
- the effect of continued coolant flow during the failure of pins and subsequent expulsion of fuel – particularly the effect of channel voiding with expulsion of sodium into the plenum,
- the effect of fuel pin length,
- the effect of the number of pins,
- the effect of axial restraint.

There is also a need to develop better understanding of the basic mechanisms involved in the thermal interaction process. The important information needs in this area are:

- fragmentation data (including the rate of fragmentation of overheated fuel and clad particles and the extent of fragmentation, i.e., ultimate particle size distributions),
- heat transfer rates,
- rate of mixing of particles with sodium and extent of mixing,
- understanding of phenomena which limit the duration of the interaction (e.g., blanketing of heat transfer surfaces by non-condensable gas or sodium vapor),
- the effect of non-condensable gas in cushioning pressure generation.

These should not be interpreted as primary information goals. The need for this type of information is largely dependent on the outcome of an initial testing and analytical program aimed toward determining realistic pressure generation and work energy data in prototypical situations.

The need for sophisticated modeling techniques and extensive laboratory experimentation to develop data on and understanding of basic mechanisms is directly related to the magnitude of the thermal interaction problem. Clearly if the results of the ANL S-Series tests are prototypical, the only need is to establish enough basic information to correlate further integral test results and to allow extrapolation of them for the purpose of reactor design and DBA analysis. This does not imply a need for full understanding of such things as fragmentation and heat transfer rate. The pursuit of such detailed information should be carried out only after a realistic perspective has been gained by the acquisition of information based on integral type tests.

Recommended Programs

The immediate emphasis should be placed on a program involving a series of increasingly prototypical in-pile loss-of-flow tests. This program should be accompanied by an analytical program which includes sufficient model development and analysis to supplement the test program. It is important to realize that an analytical program of this type does not require thorough understanding of basic mechanisms. It is possible to utilize models having uncertainties about basic mechanisms or about the magnitudes of inputs which involve basic mechanisms. By normalizing the

analytical results to the test results, the model can then be applied in order to compare one test to another by varying the parameter of significance. This type of model is not required to produce accurately absolute results, but only to substantiate hypotheses regarding the experimental results. The tests should progress very soon to systems which include multiple pins, flowing sodium with space for expansion above the shield zone and realistic pin lengths. The test series should include both clean and irradiated fuel. The primary measured variable of these tests should be the pressure transient developed in the assembly.

In keeping with this recommendation the schedule of the GE C-2 series of forced convection capsule tests should be accelerated, and the scope of the program should be broadened to cover a wider range of experimental parameters (burnup, flow, temperature, pin length, etc.).

The C-1 series of overpower transient capsules should eventually be modified to include flowing sodium and more typical geometry (multiple pins, with capsule containment designed for prototypical radial heat transfer conditions). This is not essential in the context of failure propagation, but will provide necessary information of thermal interaction damage for DBA analysis. Strong consideration should be given to conducting these tests in SEFOR, which offers conditions much more prototypical of fast reactors than does TREAT.

With respect to analytical modeling the present emphasis should be on correlation of the data obtained from the test series. For this purpose the following recommendations are made:

1. Present analytical efforts should be directed principally toward the correlation of experimental results.
2. Further development should be based on the hydrodynamic model approach rather than on purely thermodynamic model concepts. The latter models are inherently based on conservative assumptions and do not relate to the dynamic response to fuel-coolant interaction since the rate effects determine the extent of pressure generation.
3. The first model refinement which should be accomplished is to incorporate the effects of non-condensable gases. These effects include limiting the heat transfer rate by blanketing or partly blanketing the heat transfer surfaces and effectively increasing the compressibility of the medium during the early part of the transient.
4. Examination of the S-Series test data suggests that multiple shock wave reflections are occurring within the test capsule. This is an important consideration in estimating the thermal to mechanical conversion efficiency since it is necessary to distinguish between reflected pulses and original pulses generated by successive pin failures. None of the currently used modeling techniques are able to account for wave reflections and the further sophistication of the models to accomplish this should be considered in the near future.
5. The structural accommodation of violent boiling pressures should be included, either explicitly or with appropriate empirical correlations of channel dynamic response.

The need for other laboratory tests directed toward the understanding and establishment of basic parameters is largely contingent on the knowledge gained from the prototypical test series. While it may not be practical to delay entirely this type of testing until the completion of the prototypical test series, the program should place a very low priority on this type of testing until pressure generation and thermal to mechanical conversion efficiencies have been established with greater certainty. At that time further heat transfer tests and fragmentation tests and related tests may or may not be warranted. If they are it should be realized that one of the most significant parameters necessary to predict both damage propagation and the response to DBA conditions is the extent of mixing of fuel and coolant, including the geometric detail. This particular piece of information will never be available and the best that can be hoped for is to treat it somehow in a parametric fashion both in design and in DBA analysis. The undesirability of this approach is clear because it appears at this time to result in unnecessarily conservative work energy estimates.

If early experimental programs lead to the conclusion that LMFBR design and licensing will depend in a significant way on the analytical modeling of fuel-coolant thermal interaction effects, then the further sophistication of analytical modeling techniques is indicated. A primary need will be to incorporate the interactions of the expanding fuel and coolant with the containing structure. Such interactions include the expansion of the container under stress, which significantly relieves the pressure load, and the multiple reflections of shock waves due to various material interfaces and free surfaces. The need for this level of sophistication is not established yet. Should it become so, consideration should be given to the use of two-dimensional or even three-dimensional stress propagation codes such as the hydrodynamic and plastic/elastic codes under development at ANL^{15, 16}). In order to apply these tools in the context of the fuel-coolant thermal interaction, it will be necessary to couple them with a heat transfer model providing the

energy input source. Given this source, the codes as presently conceived are capable of developing the initial shock wave and following it through the coolant and also internally in the containing structure (the latter leading to prediction of internal stresses). Additional refinements which may be needed are the inclusion of the effects of heat conduction and real fluid and metal viscosities. Both of these dissipating mechanisms may significantly reduce the conservatism of the model.

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APPENDIX I

REVIEW OF LMFBR DESIGN AND SAFETY PHILOSOPHY

The starting point for this study was a review of the recently completed LMFBR follow-on studies, references 17, 18, 19, 20, 21, in particular the design and safety philosophy. The purpose of the review was to determine the contractors' design and safety philosophies and how these philosophies affect the contractors' judgement of the importance of various sodium boiling phenomena and fuel failure effects. A brief summary of the philosophies and the major philosophy-dependent design features has been tabulated (Table I-1). Standard features have normally been omitted (e.g., under "Instrumentation," flux monitors and fission product detectors have not been included.)

Discussion of the Tabulated Results

The contractors show total agreement on the existence of a cost penalty in spoiling the core to attain a non-positive sodium void coefficient. There is less agreement on the necessity of designing for that coefficient. GE, AI and B&W presented "unspoiled" cores with positive void coefficients, while the Westinghouse core was modular with a near-zero void coefficient. CE took a stand between the two types and presented a flattened cylindrical core with a moderately positive void coefficient but with moderator added to enhance the Doppler. The designers of the "unspoiled" cores require that the overall prompt reactivity feedback be negative. Thus, the Doppler must override the sodium void coefficient.

The Westinghouse concept has uncanned fuel assemblies in order to mitigate the consequences of an inlet flow blockage with cross-flow from other assemblies. Babcock and Wilcox also considered uncanned fuel bundles but reached the conclusion that it is more important to localize void effects with a channel wall. The positive void coefficient of B&W's unspoiled core makes the prevention of void propagation more important than in the Westinghouse core.

The general approach to the safety analysis is based on the AEC's "General Design Criteria for Nuclear Power Plant Construction Permits", (10 CFR 50 Appendix A). These criteria are specifically for water reactors, so most of the contractors modified them for LMFBR application. GE is the only contractor that did not do this exercise. The GE stand was that the LMFBR concept is too early in its development stage to be made subject to specific design and safety criteria. B&W made a similar statement although they did consider the AEC criteria.

The approach to containment was also indicative of basic design philosophy. The containments provided by four contractors were based on DBA-generated pressures of 100 psig (B&W) down to negligible pressures (CE). GE did not provide a conventional, sealed secondary containment dome. They rely on the inherent containment characteristics of their pot-type design. This philosophy, however, is dependent on the development of instrumentation and protective system technology to make an accident of severe consequences highly improbable. It is also dependent on the development of quantitative methods to show that the product of the probability and the consequent cost of any accident is acceptable. This typifies their probabilistic approach to safety.

The core instrumentation is a sensitive function of design and safety philosophy. Thus, the four concepts with containment have very similar instrumentation, all with bundle exit thermocouples. GE, with no containment, has taken the additional step of adding flowmeters to every channel to reduce the probability of a flow blockage accident going undetected. The flow blockage at power is generally agreed upon to be one of the most severe credible accidents for an LMFBR.

The trends and divergences in designs discussed above do not necessarily lead to different R&D needs. There is general agreement on the need for work in fuel failure phenomena, sodium boiling phenomena, and instrumentation development. Table I-2 indicates the areas of investigation that the various contractors have considered in their Task IV Reports.

Dates are the contractors' estimates of when particular studies should be completed.

The following summaries of R&D goals in the areas of fuel failure effects and sodium boiling will demonstrate areas of emphasis for the various contractors.

- a. GE — Estimates of sodium voiding patterns and rates are needed (the core protective system will be sensitive to these estimates). Out-of-pile tests are recommended to study boiling caused by overheated but intact fuel pins. In-pile tests in GETR and TREAT will be used to study boiling caused directly by molten or solid fuel contact.

Table I-1

Contractor	Philosophy & Criteria	Core Design	Containment	Instrumentation	Priority Areas of R&D
W	Conservative approach to safety. Modified AEC criteria (from 10 CFR 50 Appendix A) used in design.	4-module core to attain near-zero sodium void coefficient. Open fuel assembly to mitigate flow blockage accident. Vented carbide fuel.	Based on DBA-generated pressure of 30 psig. Multi-purpose steel lined containment dome (also provides missile protection).	2 inlet T/C's on each module. 44 core exit T/C's for each module. 1 magnetic flowmeter per loop. 1 noise detector per module.	Fuel pin failure mode and propagation mechanisms. Sodium boiling, superheat; time sequence of voiding. Inlet flow blockage consequences.
CE	Compliance with AEC criteria as modified by AIF and CE. Moderately positive sodium void coefficient accepted if overall power coefficient is negative. Protected accidents will not lead to significant fuel melting or core voiding.	Flattened cylinder L/D - .22 Moderator added to enhance doppler.	Containment design based on DBA. DBA contained in primary system with negligible pressure buildup in containment building.	Mixed mean inlet T/C's. Individual outlet T/C's. Core inlet flowmeter.	Fuel failure and propagation thresholds and mechanisms. Bond loss effects. Sodium vaporization pressures. Pressure effects on structures. Acceptable fuel damage limits definition.
B&W	Compliance with modified AEC light water reactor criteria. Due to nature of Follow-On Study, basic safety requirements cannot be fully defined. Overall prompt power coefficient will be negative under all credible circumstances. Credible accident will not destroy the core.	Cylinder L/D - .29 Positive Na void coefficient. Channel walls used to localize sodium voiding effects.	Containment dome for DBA pressures up to 100 psig.	Mixed mean inlet T/C's. Individual outlet T/C's. Flow meters on eight channels to monitor flow distribution.	Relation of pin parameters to failure type and location. SS fuel performance limits. Transient failure limits. Subassembly and fuel pin behavior under severe conditions.
GE	Design controlled by economic considerations. Safety problems will be engineered out. Specific design and safety criteria were not considered because of the early stage of LMFBR development. Probabilistic approach to safety. Optimal combination of safety features.	Cylinder L/D - .30 Positive Na void coefficient. Vented oxide fuel.	No conventional containment dome provided. Containment is inherent in the pot-type system.	Mixed mean inlet T/C's. Individual outlet T/C's. Individual flow meters.	Sodium voiding patterns and rates. Fuel-sodium heat transfer. Boiling Na vapor pressure. Fuel failure thresholds. Failed pin behavior. Failure propagation.
AI	"Realistic" approach to design with a minimum delay in licensing. Modified AEC criteria for guidelines. Overall negative power coefficient. No significant core damage from credible accident.	Square cylinder. Fuel element interaction minimized.	Containment to maintain integrity under DBA. 2 containment barriers.	Individual outlet T/C's. Other information not available.	Fuel Failure propagation modes and rates (pins & channels). Release conservation of less of flow DBA. General Sodium boiling effects. Molten fuel-sodium interaction.

Table I-2. Areas of Safety Investigation: LMFBR Follow-On Studies

SUBJECT OF INVESTIGATION	GE	B&W	CE	WEST	AI
Sodium Superheat	X	X		X	X
Voiding Pattern and Rate	X	X		X	X
Boiling Pressures	X	X	1971	X	X
Condensation Pressures		X	1971	X	
Explosive Boiling (Fuel-Na Contact)	X	X	X	X	X
Sodium Bond Loss			1971		
Sodium Void Propagation		X		1974	X
Core Disassembly due to Sodium Vapor Pressure	X	X		X	X
Fuel Pin Failures in Steady State	X	1980	1971		X
in Transient	X	1975	1971	1972	X
Fuel Pin Failure Propagation					
in Steady State	X	X	1974		X
in Transient	X	X	1976	1975	X
S/A to S/A Failure Propagation	X	X	X		X

As can be seen by the completeness of this table, the contractors seem to be concerned with the same general areas. The major deficiency, however, is the ability to determine the level of importance of each subject to each contractor.

The rate of heat transfer between fuel and sodium and the magnitude of the resultant sodium vapor pressure during boiling are of primary interest. In the area of fuel failure effects, sufficient tests are needed to estimate failure thresholds for LMFBR fuel as a function of burnup, fuel smeared density, void deployment (gap size and annular pellets) and clad thickness. In transients not producing failure, the behavior of the fuel pin (esp. fuel movement) must be investigated. Behavior of the fuel pin during failure is also important in order to assess the probability of sodium voiding and failure propagation.

- b. CE – Transient and steady-state fuel failure thresholds and mechanisms are to be established for sodium-bonded carbide fuels. Steady state tests in thermal reactors will be used to define acceptable fuel damage limits from the safety viewpoint and to investigate the propagation of single pin failures under normal operating conditions. The bond loss mechanism will receive primary attention in these studies. In-pile transient tests will be used to study the behavior of fuel pins and subassemblies with severe fuel damage. Areas of prime interest in these studies include:

1. Phase changes; fuel, clad and coolant movement and the resulting effects on reactivity;
2. Pressures created by sodium vaporization and the effect of these pressures on structures;
3. Mechanisms and dynamics of interaction between blocked and unblocked adjacent channels; and
4. Instrument response to failure propagation.

The development of a system that will confidently detect flow plugging of individual fuel subassemblies is an important need.

- c. West. – Transient testing of fuel rods is needed to study the inherent safety characteristics of sodium-bonded carbide pins. Single pin and 3-pin clusters are to be tested to determine mode of failure, propagation mechanisms

and characteristics of rapid sodium boiling. Study is recommended on the effects of fuel slumping and collapse. Experiments in representative geometries, flow conditions and heat transfer surface preparation will be used to determine as closely as possible the mechanisms of the onset of boiling and degree of superheat. Time sequence of voiding also needs study. A better understanding of void formation and propagation will be obtained from an evaluation of the inlet flow blockage accident.

- d. B&W – An effort will be directed toward defining relationships between allowable fission gas volume, gas pressure, pin lattice geometry and pin failure types and locations for sealed-plenum pins. A development need is a sophisticated fuel pin-coolant thermal-hydraulic interaction model with out-of-pile testing to substantiate it or fix empirical constants. Steady state fast flux irradiations are required to determine the maximum allowable linear heat rate, fission gas release, and burnup. Failure limits will be defined by transient tests. An experimental program is needed to further understand the following: the behavior of a subassembly following meltdown, fuel movement and/or resultant pressure transients caused by a flow or reactivity transient; the ability of the subassembly walls to prevent damage propagation through melting or pressure transients; and liquid sodium superheat and bubble formation. B&W has "discounted any near-term hopes of incorporating into the analytical work any quantitative evaluation of the probabilities of cladding failure type, location or time (related to burnup)". Their approach is to design for the circumvention of potential problems in the area of fuel failure.

- e. AI – Three development areas of interest to this study were classified high priority as follows:

1. Analytical Studies
2. Sodium Boiling Studies
3. Sodium Vapor Explosion Studies

The key analytical study area is fuel pin failure propagation information needs, with the following components:

1. fuel pin failure modes
2. effects of fission gas release and partial flow blockage on the extent and rate of cladding failure and fuel melting
3. effects of multichannel heat transfer and coolant flow on failure propagation
4. effects of fuel assembly meltdown
5. channel-to-channel propagation effects

In addition, better understanding of the phenomenon involved in the loss of flow DBA was considered necessary.

The two experimental areas identified were sodium boiling and sodium vapor explosions, with a detailed out-of-pile test program recommended. Experimental correlation is recommended to place meaningful limits on:

- the rate of coolant voiding
- superheat for incipient boiling
- transient pressure rise in the coolant channel
- rate of propagation of 'boiling disease'
- magnitude of a potential sodium vapor explosion

APPENDIX II OVERALL FAULT TREES

The following figures illustrate the overall fault trees used in this study. These fault trees are simply an outline of the paths leading from one of the twelve initiating events to one of the twelve final states. No detailed mechanisms of fuel failure propagation or failed fuel detection are presented. Instead, the total probability of progression from one state of the reactor to another state is shown. The mechanistic fault trees are shown in Appendix III.

The numbers shown on the figures are the reference or base case probabilities and frequencies.

Figure II-1

Figure II-2

Figure 11-3

Figure II-4

Figure 11-5

Figure II-6

Figure II-7

Figure II-8

Figure II-9

Figure 11-10

Figure 11-11

Figure II-12

Figure II-13

Figure II-14

Figure II-15

Figure 11-16

Figure II-17

APPENDIX III

MECHANISTIC FAULT TREES AND STATE OF THE ART REVIEW

A state-of-the-art survey was conducted in order to gather information on fuel failure effects and to use the information in estimating the probabilities of the important mechanisms. The overall fault tree provides a convenient framework in which to conduct the state-of-the-art survey. A more detailed fault tree was produced for each transition probability described on the overall fault tree.

The individual transition probabilities depend on the mechanisms involved in the transitions. For example, in the overall fault tree there is a transition probability labeled Q103 that reads: "Initial Crack Failure in One Pin Induces Burst Failure in One or More Adjacent Pins" (Figure II-14). One of the several mechanisms for this event is gas blanketing of the adjacent pin, leading to over-temperature and clad weakening followed by clad rupture and the expulsion of fuel through the clad breach and into the coolant. A diagram including all such mechanisms defines the mechanistic fault tree for the event of a crack failure inducing a burst failure in an adjacent pin. For a given fuel design, fuel performance level, and plant operating philosophy, there are certain probabilities of these mechanisms occurring. From past work done in the area of fuel element performance, defected fuel behavior and thermal hydraulics, those probabilities can be estimated, given the conditions. The state-of-the-art can be evaluated by estimating the confidence with which probabilities can be assigned to the mechanisms.

Obviously there are areas in which little or no experimental evidence is available. In these cases, analytical models of fuel element performance are investigated and a probability can be estimated from these. Confidence in this value, however, is often low until experimental verification is available.

The mechanistic fault trees are presented individually or in convenient groupings, with each grouping followed by a discussion. The discussion consists of a brief explanation of each mechanism, an indication of the applicable reference material and a brief discussion of the important core design parameters.

There is a class of qualifiers in the overall fault tree that is not treated here individually. These qualifiers are determined by the operating philosophy and the state-of-the-art of instrumentation and control systems. For example, Q13 (Figure II-4 of Appendix II) is the probability that "A Sound Fuel Bundle Containing Many Pins With Crack Failures Will Be Detected, Located and the Decision Will Be Made to Remove the Bundle." If the operating philosophy is such that the risk of leaving that bundle in the core is acceptable, then Q13 will be a low probability. If the risk is not acceptable, provisions for detection, location and removal must be made, and Q13 will be high. There is a mean annual cost associated with both approaches. If the bundle remains, there is a probability that a more serious and costly state will result. If the philosophy requires removal of the bundle, added instrumentation must be provided in order to locate the failures, the reactor must be shut down in order to remove the bundle, and the utilization of that fuel bundle will be reduced.

Clearly, trade-off studies must be performed on the above approaches during the design of the reactor. In order to make a prudent decision, some knowledge of the risk involved with leaving failed pins in the reactor must be available. The mechanistic fault trees are tools for obtaining quantitative information of this nature.

The same symbols are used in the mechanistic fault trees as for the general fault trees.

Q101 – Crack Failure Progresses to Large Area Failure in the Same Pin (Figure III-1)

This transition probability is concerned with the behavior of a pin with a crack failure if it remains in the operating reactor. The distinction between crack and large area failures is that the latter releases fuel and/or solid fission products to the coolant. The progression from crack to large area failure can occur in either of two ways: 1) internal pressure can open up the crack, or 2) the cladding itself can be removed.

The crack can be opened up by normal fuel swelling due to burnup, by fuel-sodium reaction product swelling, by thermal ratcheting, and by fission gas buildup in the fuel matrix and in the central void. If the cracked pin remains in the reactor long enough there is a high probability that these mechanisms will eventually enlarge the crack. Therefore, these are chosen as the dominant mechanisms for this transition.

The crack size that is just large enough to admit significant amounts of sodium is quite important because of possible fuel-sodium reaction product swelling. From work done at LASL¹ on sodium leakage through stainless steel

Figure III-1 Q101 Crack Failure Progresses to Large Area Failure in the Same Pin (Also applies to Q115 and 610)

tubes, it appears that sodium will penetrate extremely small breaches of the cladding. The leakage is also sensitive to the temperature. At a temperature of 400°C (752°F), a helium leak rate of 10^{-4} atm cm³/sec* does not appear to permit significant sodium leakage. At 650°C (1202°F), however, a helium leak rate of only 10^{-6} atm cm³/sec will allow significant sodium leakage according to the LASL work. This indicates that sodium leakage into a small failure may differ significantly between the core inlet (cool) and outlet (hot).

A leak rate of 10^{-6} atm cm³/sec is very small. Fission gas in a 15 kw/ft, 30 inch long pin is produced at the rate of 40×10^{-6} atm cm³/sec. If it is very conservatively assumed that fission gas leaks at the same rate as helium does, the fission gas loss would be only 2.5% of the production in the pin. If the temperature is 1200°F, this minute leak will admit sodium. From this example, it may be qualitatively concluded that any failure that allows "significant" release of fission gas will also admit "significant" quantities of sodium to the pin. This argument is presented only to establish the potential for sodium ingress. It is realized that if the leak rate is smaller than the production rate, a continuously increasing positive pressure difference will exist between the inside of the cladding and the lower, ambient pressure of the coolant. It is conceivable that sodium could leak in against the pressure gradient. Sodium wets stainless steel extremely well at operating temperatures and could "creep" in along the edges of the crack. This is a hypothetical mechanism for in-leakage of sodium. The swelling due to fuel-sodium reaction product formation due to the sodium in-leakage is discussed in Section 6.1.

There is evidence to indicate that once a crack in the cladding occurs, it is likely to become larger. In the Russian reactor, BR-5, one of the goals was to study the performance of leaking fuel elements. After the PuO₂ fuel reached 5% burnup, the reactor was shut down because of gross fission-product activity in the coolant. An examination of the fuel elements revealed that 18 of the 81 subassemblies leaked badly. Some of these subassemblies were reinserted and

*atm cm³/sec = cubic centimeters at atmospheric pressure per second.

irradiated to 5.76% burnup. The subsequent examination of the failed elements showed cracks along most of their lengths². The indication is that the faults became progressively worse with burnup.

Test B9A in GETR consisted of two pins with purposely defected cladding. A one-inch long groove was fabricated in the cladding. The depth of the groove was about two thirds the thickness of the cladding. Three small holes were drilled through the remaining cladding in the groove. When the pins were removed after irradiation, their cladding was cracked and open nearly end to end³. It appears that the initial defect provided a point for stress concentrations and that internal pressures split the cladding open.

Most failed fuel pins exhibit large area failures as opposed to "crack" failures. Crack failures may have been observed in two different situations however. In EBR-II, test pin F4D suffered a small failure that released some fission gas into its capsule. No failure could be located at room temperature in the subsequent examination^{2,8}. Later, metallographic examination revealed no evidence that sodium had entered the pin^{2,9}. It is possible, however, that the failure occurred above the capsule sodium level. The sodium level is about 13 inches below the top of the test pin in the GE capsules at room temperature⁴. Encapsulated pins from EBR-II that exhibit large area failures include F2F, F20, F4F, FQM, F2D and E1H-33 (an F3B pin).

This case of a possible "crack" failure, compared to the much larger number of large area failures observed indicate that one of two things happens. Either initial crack failures readily progress to large area failures or most failures are initiated as large area failures. If the former conclusion is correct, then the rate at which the progression occurs is of interest. Although there is no way of predicting the rate of progression at the present time, it appears clear that it must occur in a time that is short compared to the life of the pin. This is because most failures are large area failures by the time the failed pins are examined. Most failures are believed to begin as crack failures, progressing to large area failures fairly rapidly.

A power transient can also increase the internal pressure of a pin. In a transient that causes fuel melting, pressures will increase due to the ten percent volume increase in the molten fuel and the increased gas pressure in the fuel pin. According to analysis⁴ and interpretation of early experimental results⁵, the fission gas pressure increase due only to the average fuel pin temperature increase is not sufficient to fail sound cladding. More recent studies of transient effects⁶ identify the internal pressure caused by sudden local release of contained fission gas from the molten fuel as being the primary mechanism for failure. A model developed by Stuart and Thomas⁷ considers the interaction of fuel expansion upon melting and internal gas pressure. In a transient, before fuel melting starts, internal pressure increases slowly due to the heating of the contained gases. When the fuel begins melting, its volume change rapidly increases the internal pressure as it compresses the gases. When the gases are compressed to a certain point, the yield strength of the cladding is exceeded. The fuel volume increase is then accommodated by cladding strain until rupture. The compressed gases then expand to some larger volume and lower pressure. This indicates that significant fuel melting is needed to produce sufficient internal pressures to fail sound cladding.

However, if the cladding has a crack failure, the defect will act as a stress riser. less resistance will be offered to internal pressures and failure could result from the fission gas pressure increase due to the fuel temperature rise in a less severe transient. This mechanism, however, has a low probability because the frequency of power transients that lead to significant pin overheating will probably be small.

Parameters such as fuel temperature, fuel density, burnup and cladding thickness will influence the resistance of the fuel element to degradation. A sealed pin will be slightly more likely to have crack failures progress to large area failures due to fission gas pressure. If significant fuel melting occurs, venting of the pin will not help significantly because the main effect is due to the fission gas and fill gas in the fuel matrix⁷. This is shown by the mode of failure of C4A⁸, C4B⁹ and C4J¹⁰, none of which had been pre-irradiated. They all appeared to have failed due to internal pressures resulting from fuel melting.

The means of clad removal include melting and erosion. Melting could be caused by self-blanketing of the pin due to fission gas release. This has not been observed in pins cooled with forced convection. Encapsulated pins F2F and F4F in EBR-II, however, suffered gross failures that appeared to be due to fission gas blanketing from an initial (small) failure or from a gas bubble trapped in the bottom of the capsule during the assembly and filling procedures¹¹. In an LMFBF core, however, the high coolant velocity will make such a mechanism highly unlikely. The erosion of the clad around the crack is a long term effect that will have a low probability of admitting sodium to the fuel before the other

mechanisms do. Evidence of this mechanism does not appear in the reviewed literature. Coolant velocity, clad temperature, and clad material will influence the erosion rate.

Q-102 Crack Failure Progresses to Burst Failure in the Same Pin (Figure III-2)

The transition here is similar to that described in the previous fault tree except that the fuel must be forcibly expelled from the pin. This fault tree includes the mechanism of rapid sodium vaporization within the pin. It is hypothesized that, upon failure, the coolant could enter a pin and penetrate the fuel by passing into radial and circumferential cracks. The saturation vapor pressure rapidly increases in the steep thermal gradient. This increased pressure may expel fuel material from the pin. This mechanism, however, is quite speculative. More observations are necessary to confirm or disprove its existence. Pin B3C may have, in fact, followed the cut-set which includes the power transient. The pin was at overpower conditions (approximately 10% molten) at the time of failure^{1,2}.

Another mechanism on fault tree Q-102 depends on the gas blanketing of the cracked pin by its own fission gas. If this blanketing is sustained long enough to produce high clad temperatures, re-contact of the sodium on the hot cladding could lead to sodium boiling and vapor bubble collapse leading to pin failure. These effects are discussed in Sections 7.1 and 7.2. Evidence of this mechanism has not been presented in the references. The probability of this mechanism occurring is very low for the following two reasons:

- 1 A pressure history of the B3B failure^{1,3} indicates a very long time (hours) for the pin to blow down. This means the gas mass release rate was small and significant blanketing could not occur
- 2 Studies at ANL^{1,4} indicate that gas released in water does not form a coherent, insulating void in heated channels cooled by forced convection and actually spray cools the surface opposite the gas release point. Similar results are expected in the forthcoming sodium tests.

The dominant cut-set of this fault tree is similar to that for Q101. Naturally occurring internal pressures increase the stresses in the region of the crack. When the crack region starts to open up, fission gas from the central void provides a sustained driving force that continues to open the clad and, in the process of escaping, drives fuel out of the pin. Thus, sealed pins may be more likely to fail in this manner than vented pins because, on the average, the central void pressure is slightly higher in a sealed pin. This may be true if the occurrence of a crack failure and its progression to a burst failure take place during the same reactor run. If there is a shutdown, the central void in a sealed pin will be vented to the plenum and the gas can escape through the failure. From then on it will act like a vented pin. Fission gas pressure in the central void is discussed in Section 6.3.

Q-103 (104, 105) – Initial Crack Failure Induces Burst (Large Area, Crack) Failure in an Adjacent Pin (Figure III-3)

The three fault trees considered here are very similar. The variations appear in the final conditions; those that define the type of induced failure. Fuel expulsion and fuel washout are the dividing lines between burst, large area and crack failures.

It is assumed that the only effect from a crack failure is the local undercooling due to the fission gas release. Available analyses indicate that fission gas will probably not cause melting of the clad of an adjacent pin. However, the pessimistic assumption used in those analyses lead to large local clad temperature increases. If these increases are non-uniform circumferentially, thermal bowing may aggravate the overheating. The analysis in Section 6 indicates that this bowing will not be a serious problem.

From the above considerations, fission gas release can cause overheating due to blanketing. As the cladding of the adjacent pin increases in temperature its ultimate strength decreases rapidly. For example, if the internal pressure is 1000 psi, the hoop stress on a 0.250 inch outside diameter, 0.010 inch thick cladding is 11,500 psi. The ultimate strength for irradiated type 316 stainless steel is 11,500 psi when the temperature is in the range of 1600°F^{1,5}. A THTD run shows that, if a 120° segment of the cladding is blanketed, the cladding will reach 1600°F in 0.5 second. In calculations of fission gas release rate in the choked flow regime, flow from an 18 mil diameter orifice can blanket a 120° segment for 0.8 second. Clad failure could therefore be expected in sealed pins with an internal pressure of 870 psi.

There are two pessimistic assumptions used in those calculations that are severe in light of recent work. First, perfect communication between the gas plenum and the failure was assumed (a free path down the central void). Pressure measurement of the B3B blowdown^{1,6} indicated a significant flow resistance is likely (slow pressure decrease

Figure III-2 Q102 Crack Failure Progresses to Burst Failure in the Same Pin (Also applies to Q117, 600)

in the pin after failure). Second, the fission gas was assumed to be released as a coherent bubble that perfectly insulates the fuel pin surfaces. ANL work offers strong evidence against this release behavior.

The experimental work at ANL^{14,17} was conducted in a water loop in which simulated fission gas impinges on a heated wall of the rectangular channel. Even with high heat fluxes (0.5 and 1.0×10^6 Btu/hr-ft²), the surface directly opposite the gas jet was cooled possibly as a result of improved heat transfer due to spray cooling. A short distance upstream from the point of gas impingement (about 3/4 inch) wall temperatures were observed to increase. For a case of heat flux = 0.50×10^6 Btu/hr-ft², initial flow velocity - 20 fps, gas pressure - 1000 psi and an orifice diameter of 0.050 inches, the measured temperature rise was 130°F and the flow decreased to 9-10 fps. A temperature increase was calculated from the flow transient data and it agreed well with the 130°F observed. The temperature at the point of impingement should not decrease as much with sodium as with water. The high conductivity of sodium makes it less sensitive to the added cooling due to spray cooling.

ANL studies will consider three regimes of fission gas release:

1. Rapid release through a break in the cladding. This could cause expulsion of coolant from the channel but would be completed very quickly.
2. Slow release - a high velocity gas jet impinging on an adjacent fuel element may persist long enough to cause overheating;
3. Slower release that might cause overheating due to local flow reduction brought about by flow characteristics of the two phase coolant-gas mixture¹⁸.

The mechanism of gas blanketing and sodium recontact on the hot cladding was discussed under Q-102. The pressure forces created by boiling or bubble collapse could lead to failure of the adjacent pin. The probability of this mechanism occurring is small for the same reasons discussed under Q-102.

The probability of fission product cladding attack causing cladding failure is very low because:

- Therefore the total probability of damage to adjacent pins due to crack failure in a given pin is very small.

Q-201 is the probability that a pin, operating with a large area failure, will suddenly expel fuel material into the coolant channel. Some hypothetical mechanisms have been defined for this occurrence. These mechanisms are discussed and analyzed in Section 6.3.

This transition probability has been assigned a high value because of the uncertainty associated with fuel-sodium reaction product swelling. This phenomenon is discussed in Section 6.1.

Large area failures may also cause overheating of other pins by producing debris that can be trapped in small flow passages, thus restricting the coolant flow. This is discussed in Section 6.2.

Fuel-sodium thermal interactions causing boiling or acoustic shock waves in the coolant are discussed in Section 7.

Figure III-4 Q201 Large Area Failure Progresses to a Burst Failure in the Same Pin (Also applies to Q215, 620)

Q-301, (302, 303) – Initial Burst Failure Induces Burst (Large Area, Crack) Failure in One or More Adjacent Pins (Figure III-6)

The distinguishing characteristic of burst failures is their forceful expulsion of fuel material into the coolant channel. After the expulsion has taken place, the failure acts like a large area failure. For this reason, the fault trees for burst failures are the same as for large area failures with the expulsion effects superimposed. Since the fault trees for large area failures were discussed in the previous section, only the expulsion effects will be considered here.

The contact of hot fuel and liquid sodium can lead to rapid sodium expansion and boiling. Boiling phenomena are covered in Section 7. The contact may lead to immediate generation of pressure forces capable of failing adjacent pins or leading to channel voiding. An analysis of EBR-II driver fuel failure led to the conclusion that fuel-sodium thermal interactions have a very low probability for propagation of faults in fuel elements¹⁹. According to Babcock and Wilcox, it is generally accepted that the pressures involved in void formation and growth will not be great enough to damage the structures because they are limited to the saturation pressure of the liquid at its interface with the bubble. However, it is probable that a higher pressure will arise when the vapor bubble condenses in the subcooled liquid surrounding the bubble.²⁰

Sodium reentry after voiding can create damaging forces. Effects of the void collapse and sodium hammer associated with reentry are discussed in Section 7 with the conclusion that sodium hammer is a potentially damaging mechanism. In addition, an inertial reentry effect has also been postulated as follows. The pressure drop across a bundle is considered to be constant. The frictional pressure drop in the sodium is proportional to the wetted length of the channel. If part of the channel is voided, that wetted length is reduced. Ignoring the friction of the sodium vapor, the

velocity of the liquid sodium increases inversely as the square root of the wetted length ($\Delta P \propto V^2 L$). Thus, if a substantial fraction of the channel is voided, the reentering sodium may have high velocity. When it enters the pin bundle, large forces may be imposed on the pins. Forces due to momentum change and perhaps excessive vibration may result in pin failure. Reference to this mechanism in the reviewed literature was not found. A brief study of this mechanism in Section 7.2 indicates that the probability of damage is very low.

Figure III-5 Q202 (203, 204) Initial Large Area Failure Induces a Burst (Large Area, Crack) Failure in an Adjacent Pin (Also applies to Q212, 213, 214)

Q406 (407,408) – Partial Blockage with Reduced Flow Causes Burst (Large Area, Crack) Failures in Many Pins (Figure III-7)

The effects on fuel pin life of undercooling caused by reduced flow is discussed in Section 4.1. This general degradation of the cladding is considered here to be the most important effect of a flow reduction that does not result in sodium boiling.

Q34, (42, 54) – Many Pins with Large Area Failures Cause Rupture of the Channel Wall in Only One (Few, Many) Bundle (Figure III-8)

Q34 and Q42 are similar except in magnitude. Q54 adds the mechanism of a reactivity increase. Reactivity effects are only applied to many-bundle cases because the long neutron mean free path in fast reactors precludes localized reactivity differences. The probabilities of "one" and "few" large area failures causing channel wall damage were considered to be negligible in this exercise and are excluded from the fault tree.

There are four cut-sets or possible paths shown on the fault trees for Q34 and Q42. When the sodium first contacts the hot fuel, rapid expansion may produce a shock wave that damages channel walls. Sub-cooled sodium boiling and

Figure III-6 Q301 (302, 303) Initial Burst Failure Induces Burst (Large Area, Crack) Failure in Adjacent Pin (Applies also to Q311, 312, 313)

bubble collapse may damage the walls. Self-blanketing prior to sodium-fuel contact may increase the fuel surface temperature thus enhancing thermal interaction. The likelihood of this mechanism occurring is small, for the reasons discussed under Q202, 203, 204.

If the rapid vaporization produces large inertial effects, channel voiding may result. Reentry effects include thermal interaction of sodium with hot fuel and clad, large void collapse pressures, and inertial effects if the reentry is at high speed. Channel wall rupture could conceivably result from any of these. These were discussed in the previous section.

Overheating of the channel wall due to eroded fuel contact or flow blockage by failure debris (see Section 5.2) could conceivably weaken the channel wall enough for the pressure differential and thermal stresses to rupture it. Melt-through is also possible but the sodium-filled gap that is present in the 1000 MWe Follow-On Study designs^{21, 22, 23} makes this very unlikely. In order for this mechanism to affect more than one bundle, the channel wall of the first bundle must be ruptured and the fuel debris must contact the adjacent channel wall.

The most likely mechanism for Q34 and Q42 is, therefore, local overheating of the channel walls due to fuel and cladding debris. ORNL is planning tests to mock-up flow blockage due to fuel pin swelling or bowing and the release of failure debris²⁴. The ANL Fuel Element Failure Propagation Program² includes an experiment to study the effects of mixing hot fuel and relatively cold sodium. Hot, powdered UO_2 will be injected into flowing sodium in the sodium test loop. The accumulation of UO_2 will be monitored by means of an X-ray unit.

A reactivity increase could result from channel voiding or fuel relocation. In the GE 1000 MWe Follow-On

Figure III-7 0416 (417, 418) Partial Blockage with Reduced Flow Causes Burst
(Large Area, Crack) Failures in Many Pins

Design²⁵, the worst reactivity additions that could result from these faults are \$0.18 for fuel compaction and \$0.06 for voiding. This appears unlikely to increase the heat flux to a point where many channels are voided. If this does occur, however, fuel pin overheating and sodium reentry effects may cause severe core damage. Reactivity effects will be produced by each voiding, and if many channels void at the same time, more serious consequences will result. If the initial reactivity addition is of significant magnitude, it will have a very high probability of actuating the safety system.

Q36 (43, 55) – Many Pins with Crack Failures Cause Rupture of the Channel Wall of One (Few, Many) Bundle (Figure III-9)

The only way in which crack failures could cause channel wall damage is through channel or subchannel voiding due to released fission gas. If the entire channel is voided, all of the reentry and reactivity effects previously discussed apply. If individual pins are blanketed due to the gas release of many pins, high temperatures could result. Reentry of the sodium into these hot subchannels could result in damaging shock waves.

The effect of fission gas release depends on the rate of release. If the measured gas release rate of pin B3B¹⁶ is indeed representative of gas release rates from pin failures in the fueled region, then there is a very small probability of gas release causing voiding.

On the other hand, if all the pins in the bundle failed simultaneously in the plenum region (assuming sealed pins) the pressures may reverse the flow of sodium. Analytical and experimental studies are necessary to determine the possible consequences of this type of gas release. ORNL plans large scale gas release experiments in bundles of up to 217 pins²⁴. The probability of this mechanism occurring cannot be determined from the existing data.

Figure III-8 Q34 (42, 54) Many Pins with Large Area Failures Cause Rupture of the Channel Walls of One (Few, Many) Bundle

Figure III-9 Q36 (43, 55) Many Pins with Crack Failures Cause Rupture of the Channel Wall of One (Few, Many) Bundle

The probability of channel wall damage due to crack failures in the fuel pins is considered to be very low because:

1. There is a low probability that many crack failures will occur simultaneously in a bundle, and
2. The available information shows that the gas release rate from a failure in the fueled region of the pin will probably be too low to cause voiding.

Venting the fuel pins would further reduce the probability of this mechanism occurring.

Q28 (30, 32), Q39 (40, 41) and Q51 (52, 53) One (Few, Many) Burst Fuel Pin Causes Rupture of the Channel Wall in One (Few, Many) Fuel Bundle (Figure III-10)

A channel wall may be ruptured by mechanical forces and by overheating. Mechanical forces can result from thermal interaction of the sodium and hot fuel from a burst failure. Briefly, when the burst failures occur, there is rapid heat transfer from the expelled fuel to the coolant. This may either boil the sodium or produce an acoustic wave due to rapid thermal expansion of the liquid sodium, depending on the effectiveness of the heat transfer. If boiling occurs, sodium vapor expansion may void the channel and the subsequent reentry effects may fail the channel wall. If the initial sodium heating produces acoustic energy, the channel wall may be failed by the shock wave without channel voiding. These mechanisms are discussed and analyzed in Sections 5 and 7

Overheating of the channel wall may result from failure debris blocking coolant flow next to the wall, resulting in the pins adjacent to the wall overheating, bowing and contacting the wall. The fuel debris itself may become trapped against the wall and overheat it as discussed in Section 5.2 The overheated wall loses strength. If this loss is sufficient, the pressure differential between the channel and the gap space may rupture the wall. Fuel on the channel wall may also melt through but analyses²⁶ show that this would take on the order of fifteen seconds; possibly enough for detection of fuel and fission products and shutdown if the instrumentation were in the safety circuit and programmed correctly. With a well-designed electronics system and a favorable detector site, a delayed neutron monitoring system can give a failure indication within about five seconds from the actual moment of failure²⁷

Further analyses and testing are required in order to assign probabilities to these mechanisms. The disposition of burst failure debris is important (fuel particle size, clad fragment size, etc.). The channel plugging tests planned by ORNL²⁴ will provide this type of information.

Q600-617 – One (Few, Many) Burst Failures in One Bundle Causes Many Burst (Large Area, Crack) Failures in Few (Many) Sound Bundles (figure III-11)

Two mechanisms have been hypothesized for these eighteen transition probabilities. One is a high-energy shock wave, the other is multiple channel voiding.

If the rapid sodium expansion resulting from contact with hot fuel from burst failures produces an acoustic shock wave with wavelength, λ , and the thickness of the channel wall is $n\lambda/2$, where n is any integer, most of the energy will be transmitted through the wall. If the magnitude of the shock wave is not attenuated too much, it could cause pin failure in other bundles after it passes through the channel walls. Insufficient data are available at the present time to assign a probability to this mechanism.

If the burst failure causes sodium boiling that voids the channel, the possibility exists that the void will expand into the inlet plenum and cause voiding of other channels. Subsequent reentry forces, as discussed previously could cause widespread pin failure.

The voiding of many channels simultaneously could also cause reactivity effects. A power transient that causes widespread pin overheating is a mechanism that could cause the transition. At \$0.06 per voided channel²⁵, it would require a large number of simultaneously voided channels to cause a dangerous power transient. Again, insufficient information is available to assign probabilities to the mechanism of multiple-channel voiding and consequent reactivity effects.

Figure III-10 Q28 (30, 32), Q39 (40, 41) and Q51 (52, 53)
One (Few, Many) Burst Fuel Pin Causes Rupture of the Channel Wall in One (Few, Many) Bundle

Sodium Boiling Mechanisms

The important mechanisms for accident progression and propagation once sodium boiling is initiated are discussed in this section. These mechanisms are divided again, into strictly sodium boiling phenomenon and those associated with fuel/clad-sodium thermal interaction (violent boiling). These mechanisms are generally postulated to occur in the same time scale as the safety system response, therefore they are considered "rapid" modes of propagation.

Figure III-11 Q650-667 One (Few, Many) Burst Failure in One Bundle Causes Many Burst (Large Area, Crack) Failures in Few (Many) Sound Bundles

Sodium Boiling

Sodium boiling is considered initiated by two basic faults, local fuel/clad debris blockage within the fuel pin array (Figures III-5 – III-6) and loss-of-flow to the fuel bundle due to inlet blockage or loss of inlet plenum pressure (Figure III-12). Details of the review of sodium boiling are found in Section 7.1 and 7.2. Table III-1 summarizes the qualitative results of this review with regard to the fault tree input.

Fuel/Clad-Sodium Thermal Interactions

The possibility exists for the generation of damaging pressures due to the sudden mixing of molten clad and fuel and sodium. Initiating events are power transients, core loss-of-flow incidents, bundle inlet blockages and progression of local debris blockages inside the fuel pin array that can cause channel voiding (see Figures III-13 and III-14). The thermal interaction can occur anytime from the initiation of melting until the clad and fuel are resolidified (either by reactor scram or movement out of the neutron flux to a sufficient heat sink). The details of this violent boiling phenomenon are thus linked directly to the sodium boiling phenomenon discussed above but have traditionally been separated, presumably to emphasize the possible violence of the interaction. Table III-2 summarizes the review of Section 7.3.

Figure III-12 Q406 (407, 408) Partial Blockage with Coolant Boiling Causes Burst
(Large Area, Crack) Failures in Many Pins

Table III-1. Qualitative Fault Tree Input for Sodium Boiling

Sodium Boiling Phenomenon	State-of-the-Art	Probability of Occurrence
1. Local boiling stability (typically around a local blockage caused by fuel/clad debris in the fuel bundle)	No analysis or tests applicable	Qualitatively unstable. Unknown whether local boiling progresses violently or smoothly.
2. Vapor bubble collapse damage potential (local sub-cooled boiling leading to bubble collapse on adjacent cold and undamaged fuel pins)	No direct test of damage potential. Analysis and tests of bubble collapse (water and sodium) suggest high pressure and energies are available. Deposition of energy on pins is uncertain.	Damage potential by fuel pin bending or cavitation not considered serious. Local clad bending and effect of fuel support could not be evaluated, but considered potentially serious for collapse directly on a fuel pin. Not pursued because local boiling considered thermally unstable.
3. Boiling heat transfer and burnout (determine rate of clad and fuel temperature increase following boiling initiation)	Data suggests a hydrodynamic limit on fuel bundle cooling due to 2ϕ pressure drop and possible critical flow. Forced convection boiling heat transfer adequate for low (<10%) quality but burnout occurring at higher qualities. No data on local boiling burnout.	Burnout leading to meltdown considered <i>likely</i> once boiling envelops the entire bundle. Local boiling burnout very uncertain.
4. Pressurization due to boiling initiation	In-reactor superheats expected to be low (<150°F) based on extensive out-of-pile tests.	Pressurization of channels on initial boiling is not considered significant at the likely superheats.
5. Forces due to sodium reentry (slug impact on tube bundle)	Analysis and experiments show high reentry velocities possible (up to 70 fps) in slug flow.	Damage potential very small for assumed velocities.
6. Sodium hammer	Tests in single heated tubes show sodium hammer to be real. Analysis shows it to be potentially very damaging to the channel.	Qualitatively, the flow regime (dispersed) to be expected in a voiding tube bundle, the complex geometry and heat transfer and the presence of fission gas are likely to preclude a clean sodium slug reentry and the resulting sodium hammer. Uncertainty is large, however, leaving cause for concern.

Figure III-13 Q701 (702, 703), 811 (812, 813) Channel Voiding Causes Many Crack (Large Area, Burst) Failures in a Few (Many) Sound Bundles

Table III-2. Qualitative Fault Tree Input for Violent Boiling

Violent Boiling Mechanisms	State-of-the-Art	Probability of Occurrence
1 Power Transient Dispersion of Fuel into Sodium	Traditionally receiving the major emphasis; until recently (ANL S-5 test) there was no experimental evidence of violent boiling in-pile. ANL S-5 results are not yet analyzed. Analytically, a wide range of energy release is calculated.	Very likely to cause significant damage due to core pressurization. Uncertainties are very large and damage potential not yet quantified. Possibility of mitigation by presence of fission gas noted.
2. Loss-of-Flow Meltdowns	Just now receiving attention (note PA-10, Task C-2). No applicable results. Analytically very uncertain.	Very likely to cause significant damage and propagation but also very uncertain. As above, the presence of fission gas may mitigate the damage potential.

Figure III-14 Q37 (44, 56) Channel Voiding Causes Immediate Failure of the Channel Walls of One (Few, Many) Bundles

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APPENDIX IV FAULT TREE DOMINANT CUT-SETS

The overall fault trees shown in Appendix II were used to calculate the mean-annual-cost and to determine the "dominant cut-sets." (Where the dominant cut-set is that sequence of transitions, from an initial event to the final state, which makes the largest contribution to the mean-annual-cost for a given fault tree.)

Associated with each transition probability on the dominant cut-sets is a complete "mechanistic fault tree." (Where a mechanistic fault tree is one which shows all of the conceivable sequences of physical mechanisms through which the given transition, say large area failure in one pin induces large area failure in a few adjacent pins, could occur.) The complete mechanistic fault trees are presented in Appendix III.

Each of the mechanistic fault trees also has a dominant cut-set for a given reactor design and operating philosophy. This cut-set was chosen (for the GE 1000 MWe design) in order to assign the transition probabilities for input into the overall fault trees.

The following pages show the dominant cut-sets identified in this study. Parts of the cut-sets are repetitious and refer back to the first figure showing that cut-set. Twelve dominant cut-sets are shown for the overall fault trees, one for each final state identified. Dominant cut-sets of the mechanistic fault trees are shown after they first occur.

EVENTS WHICH RESULT IN REACTOR CONTAMINATION BY FISSION GAS FROM A BUNDLE WITH ONE FAILED FUEL PIN

Figure IV-1

MECHANISMS THROUGH WHICH CRACK FAILURE IN ONE PIN PROGRESSES
TO LARGE AREA FAILURE IN THE SAME PIN (Q-115, Q-101, Q-610)

A crack failure in one pin has progressed to a large area failure in the same pin.

Material is *not* forced from the pin into the coolant.

Cladding breach becomes large enough to permit fuel leaching by the coolant and/or fuel washout.

Fuel-sodium reaction product swelling causes localized cladding stresses that cause the cracked clad to open up.

Pellet ratcheting due to thermal cycling causes the crack to open up slowly.

The reactor continues to operate with a fuel bundle which contains one pin with a crack failure.

Figure IV-2

**EVENTS WHICH RESULT IN REACTOR CONTAMINATION BY FISSION GAS
FROM A SOUND BUNDLE WITH A FEW FAILED PINS**

Figure IV-3

MECHANISMS THROUGH WHICH A LARGE AREA FAILURE IN ONE PIN INDUCES CRACK FAILURES IN ONE OR MORE ADJACENT PINS (Q-204)

Figure IV-4

EVENTS WHICH RESULT IN REACTOR CONTAMINATION BY FISSION GAS FROM A BUNDLE WITH MANY FAILED PINS

Figure IV-5

MECHANISMS THROUGH WHICH LARGE AREA FAILURE IN ONE PIN
INDUCES BURST FAILURE IN AN ADJACENT PIN (Q-202)

Figure IV-6

MECHANISMS THROUGH WHICH BURST FAILURES INDUCE
CRACK FAILURES IN ADJACENT PINS (Q-313)

Figure IV-7

EVENTS WHICH RESULT IN REACTOR CONTAMINATION BY FUEL AND FISSION
PRODUCT RELEASE FROM A BUNDLE WITH ONE FAILED PIN

S-4 The reactor is contaminated by fuel and fission product release from a bundle with one failed pin.

Figure IV-8

EVENTS WHICH RESULT IN REACTOR CONTAMINATION BY FUEL AND FISSION
PRODUCT RELEASE FROM A BUNDLE WITH A FEW FAILED PINS

S-5 - The reactor is contaminated by fuel and fission product release from a bundle with a few failed fuel pins.

Q-64 - A sound fuel bundle containing a few pins with large area failures remains in the reactor.

Q-115 - Crack failures in a few pins progress to large area failures in the same pins.

Q-66 - A sound fuel bundle containing one crack failure and one large area failure remains in the reactor.

Figure IV-9

EVENTS WHICH RESULT IN REACTOR CONTAMINATION BY FISSION PRODUCT
AND FUEL RELEASE FROM A BUNDLE WITH MANY FAILED FUEL PINS

-
- S-6 — The reactor is contaminated by fission product and fuel release from a bundle with many failed fuel pins.
- Q506 — A sound fuel bundle containing a few large area and a few burst failures remains in the reactor.
- Q-115 — Crack failures in a few pins progress to large area failures in the same pins.
- Q-78 — A sound fuel bundle containing a few pins with large area failures and a few pins with crack failures remains in the reactor.
- Q-313 — Secondary burst failure induces crack failures in one or more adjacent pins.
- Q-501 — A sound fuel bundle containing one or more burst failures and one large area failure remains in the reactor.
- Q-202 — Initial large area failure induces burst failure in one or more adjacent pins.
- Q-61 — A sound fuel bundle containing one pin with a large area failure remains in the reactor.
- Q-101 — Crack failure progresses to large area failure in the same pin.
- Q-60 — A fuel bundle containing one pin with a crack failure remains in the reactor.
- I-1 — Spontaneous crack type clad failure.

Figure IV-10

EVENTS WHICH RESULT IN REACTOR SHUTDOWN TO REMOVE ONE FUEL BUNDLE WITH UNDAMAGED WALL

Figure IV-11

MECHANISMS THROUGH WHICH A BURST FAILURE INDUCES A BURST FAILURE IN ONE OR MORE ADJACENT PINS (Q-311)

Figure IV-12

EVENTS WHICH RESULT IN REACTOR SHUTDOWN TO REMOVE A FEW BUNDLES WITH UNDAMAGED WALLS

Figure IV-13

**MECHANISMS THROUGH WHICH FUEL PIN BURST FAILURES IN ONE BUNDLE
CAUSE FUEL PIN FAILURES IN OTHER BUNDLES, WITHOUT
DAMAGING THE CHANNEL WALLS (Q-662)**

Burst failure in one or more pins of one bundle has caused many fuel pin failures in other sound bundles.

Figure IV-14

EVENTS WHICH RESULT IN REACTOR SHUTDOWN TO REMOVE MANY BUNDLES WITH UNDAMAGED WALLS

Figure IV-15

EVENTS WHICH RESULT IN REACTOR SHUTDOWN TO REMOVE ONE BUNDLE WITH DAMAGED WALL

Figure IV-16

MECHANISMS THROUGH WHICH MANY BURST FAILURES CAUSE CHANNEL WALL FAILURE (Q-32, -41, -53)

Figure IV-17

EVENTS WHICH RESULT IN REACTOR SHUTDOWN TO REMOVE A FEW BUNDLES WITH DAMAGED WALLS

Figure IV-18

**EVENTS WHICH RESULT IN REACTOR SHUTDOWN
TO REMOVE MANY BUNDLES WITH DAMAGED WALLS**

Figure IV-19

NOMENCLATURE AND DEFINITIONS

I. FAULT TREE END STATES		Cost per Event (10 ³ \$)
S-1	Single Crack Failure	0
S-2	A Few Crack Failures	0
S-3	Many Crack Failures	0
S-4	Single Pin Failure Releasing Fuel	0
S-5	A Few Pin Failures Releasing Fuel	0
S-6	Many Pin Failures Releasing Fuel	18.5
S-7	Reactor Is Down To Remove A Single Sound Bundle	147.0
S-8	Reactor Is Down To Remove A Few Sound Bundles	156.0
S-9	Reactor Is Down To Remove Many Sound Bundles	196.0
S-10	Reactor Is Down To Remove A Single Damaged Bundle	214.0
S-11	Reactor Is Down To Remove A Few Damaged Bundles	544.0
S-12	Reactor Is Down To Remove Many Damaged Bundles	1839.0

Frequency,
Events per Year

II. INITIATING EVENTS		
I-1	Crack Failure Of One Pin	110
I-2	Large Area Failure Of One Pin	11
I-3	Burst Failure Of One Pin	1
I-4	Bundle Inlet Blockage Causing Flow Reduction	0.01
I-5	Bundle Inlet Blockage Causing Local Boiling	0.01
I-6	Bundle Inlet Blockage Causing Voiding	0.01
I-7	Loss Of Cooling To A Few Bundles	0.01
I-8	Loss Of Cooling To Many Bundles	0.01
I-9	Reactivity Addition Causing Crack Failures In Many Bundles	0.01
I-10	Reactivity Addition Causing Large Area Failures in Many Bundles	0.01
I-11	Reactivity Addition Causing Burst Failures in Many Bundles	0.01
I-12	Reactivity Addition Causing Channel Wall Damage In Many Bundles	0.01

NOMENCLATURE AND DEFINITIONS (Continued)

III. FAULT TREE NOMENCLATURE AND SYMBOLS

Symbols Used in Fault Trees

BREEDER REACTOR DEPARTMENT
SUNNYVALE, CALIFORNIA 94086

Expanding the World's Energy Reserves

GENERAL **ELECTRIC**