CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

VOLUME 15

PROJECT MANAGEMENT CORPORATION

TABLE OF CONTENTS

Section		Page
1.0	INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT	1.1-1
1.1	INTRODUCTION	1.1-1
1.1.1	General Information	1.1-2
1.1.2	Overview of Safety Design Approach	1.1-3
1.1.3	Applicability of Regulatory Guides	1.1-5
1.2	GENERAL PLANT DESCRIPTION	1.2-1
1.2.1	Site	1.2-1
1.2.2	Engineered Safety Features	1.2-2
1.2.3	Reactor, Heat Transport and Related Systems	1.2-2
1.2.4	Steam Generator - Turbine and Related Systems	1.2-3
1.2.5	Offsite and Onsite Power	1.2-5
1.2.6	Instrumentation, Control and Protection	1.2-6
1.2.7	Auxiliary Systems	1.2-7
1.2.8	Refueling System	1.2-8
1.2.9	Radwaste Disposal System	1.2-9
1.2.10	Reactor Confinement/Containment System	1.2-9
1.2.11	Major Structures	1.2-10
1.3	COMPARISON TABLES	1.3-1
1.3.1	Comparisons with Similar Designs	1.3-1
1.3.2	Detailed Comparison with Fast Flux Test Facility	1.3-2
1.4	IDENTIFICATION OF PROJECT PARTICIPANTS	1.4-1
1.4.1	Functions, Responsibilities and Authorities of Project Participants	1.4-2
1.4.2	Description of Organizations	1.4-3
1.4.3	Interrelationships with Contractors and Suppliers	1.4-21a
1.4.4	General Qualification Requirement of CRBRP Project Participants	1.4-22
1.5	REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION	1.5-1
1.5.1	Information Concerning the Adequacy of a New Design	1.5-2
1.5.2	Information Concerning Margin of Conservatism of Proven Design	1.5-28
1.5.3	References	1.5-47

TABLE OF CONTENTS (Cont'd.)

Section		Page
1.6	MATERIAL INCORPORATED BY REFERENCE	1.6-1
1.6.1	Introduction	1.6-1
1.6.2	References	1.6-1
	Appendix 1-A Flow Diagram Symbols	1.A-1
2.0	SITE CHARACTERISTICS	2.1-1
2.1	GEOGRAPHY AND DEMOGRAPHY	2.1-1
2.1.1	Site Location and Layout	2.1-1
2.1.2	Site Description	2.1-2
2.1.3	Population and Population Distribution	2.1-4
2.1.4	Uses of Adjacent Lands and Waters	2.1-8
2.2	NEARBY INDUSTRIAL, TRANSPORTATION AND MILITARY FACILITIES	2.2-1
2.2.1	Locations, Routes, and Descriptions	2.2-1
2.2.2	Evaluations	2.2-3
2.2.3	New Facility/Land Use Requirements	2.2-4c
2.3	METEOROLOGY	2.3-1
2.3.1	Regional Climatology	2.3-1
2.3.2	Local Meteorology	2.3-4
2.3.3	On-site Meteorological Monitoring Program	2.3-9
2.3.4	Short-Term (Accident) Diffusion Estimates	2.3-9
2.3.5	Long-Term (Average) Diffusion Estimates	2.3-13
2.4	HYDROLOGIC ENGINEERING	2.4-1
2.4.1	Hydrologic Description	2.4-1
2.4.2	Floods	2.4-6
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	2.4-10
2.4.4	Potential Dam Failures (Seismically and Otherwise Induced)	2.4-21
2.4.7	Ice Flooding	2.4-31
2.4.8	Cooling Water Canals and Reservoirs	2.4-31a
2.4.9	Channel Diversions	2.4-32
2.4.10	Flooding Protection Requirements	2.4-32
2.4.11	Low Water Considerations	2.4-33
2.4.12	Environmental Acceptance of Effluents	2.4-42
2.4.13	Groundwater	2.4-44
2.4.14	Technical Specification and Emergency Operation Requirement	2.4-55

TABLE OF CONTENTS (Cont'd.)

Section		Page
2.5	GEOLOGY AND SEISMOLOGY	2.5-1
2.5.1	Basic Geologic and Seismic Information	2.5-1
2.5.2	Vibratory Ground Motion	2.5-20
2.5.3	Surface Faulting	2.5-27
2.5.4	Stability of Subsurface Materials	2.5-32
2.5.5	Slope Stability	2.5-48a
	Appendix 2-A Field Investigative Procedures	2A-1
	Appendix 2-B Laboratory Test Procedures	2B-1
	Appendix 2-C Report of Test Grouting Program	2C-1
	Appendix 2-D Report of Engineering Properties for Crushed Stone Materials from Commercial Suppliers	2D-1
	Appendix 2-E Extracts from U.S. Atomic Energy Commission AEC Manual	2E-1
	Supplement 1 to Chapter 2 Deleted	
	Supplement 2 to Chapter 2 Question and Responses Related to Chapter Two Information and Critical For NRC Docketing of CRBRP Environmental Report	1
3.0	DESIGN CRITERIA - STRUCTURES, COMPONENTS EQUIPMENT AND SYSTEMS	3.1-1
3.1	CONFORMANCE WITH GENERAL DESIGN CRITERIA	3.1-1
3.1.1	Introduction and Scope	3.1-1
3.1.2	Definitions and Explanations	3.1-2
3.1.3	Conformance with CRBRP General Design Criteria	3.1-8
3.2	CLASSIFICATIONS OF STRUCTURES, SYSTEMS, AND COMPONENTS	3.2-1
3.2.1	Seismic Classifications	3.2-1
3.2.2	Safety Classifications	3.2-2
3.3	WIND AND TORNADO LOADINGS	3.3-1
3.3.1	Wind Loadings	3.3-1
3.3.2	Tornado Loadings	3.3-2
3.4	WATER LEVEL (FLOOD) DESIGN	3.4-1
3.4.1	Flood Protection	3.4-1
3.4.2	Analysis Procedures	3.4-1a

TABLE OF CONTENTS (Cont'd.)

Section		Page
3.5	MISSILE PROTECTION	3.5-1
3.5.1	Missile Barrier and Loadings	3.5-4
3.5.2	Missile Selection	3.5-4a
3.5.3	Selected Missiles	3.5-7
3.5.4	Barrier Design Procedures	3.5-10
3.5.5	Missile Barrier Features	3.5-13c
3.6	PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING	3.6-1
3.6.1	Systems in Which Pipe Breaks are Postulated	3.6-1
3.6.2	Pipe Break Criteria	3.6-2
3.6.3	Design Loading Combinations	3.6-2
3.6.4	Dynamic Analysis	3.6-3
3.6.5	Protective Measures	3.6-8
3.7	SEISMIC DESIGN	3.7-1
3.7.1	Seismic Input	3.7-1
3.7.2	Seismic System Analysis	3.7-4a
3.7.3	Seismic Subsystem Analysis	3.7-11
3.7.4	Seismic Instrumentation Program	3.7-16
3.7.5	Seismic Design Control	3.7-20
	Appendix to Section 3.7 Seismic Design Criteria	3.7-A.1
3.8	DESIGN OF CATEGORY I STRUCTURES	3.8-1
3.8.1	Concrete Containment (Not Applicable)	3.8-1
3.8.2	Steel Containment System	3.8-1
3.8.3	Concrete and Structural Steel Internal Structures of Steel Containment	3.8-8
3.8.4	Other Seismic Category I Structures	3.8-22a
3.8.5	Foundation and Concrete Supports	3.8-35
	Appendix 3.8A Buckling Stress Criteria	3.8A-1
	Appendix 3.8-B Cell Liner Design Criteria	3.8-B.1
	Appendix 3.8-C Catch Pan and Fire Suppression Deck Design Criteria	3.8-C.1
3.9	MECHANICAL SYSTEMS AND COMPONENTS	3.9-1
3.9.1	Dynamic System Analysis and Testing	3.9-1
3.9.2	ASME Code Class 2 and 3 Components	3.9-3a
3.9.3	Components Not Covered by ASME Code	3.9-5
3.10	SEISMIC DESIGN OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT	3.10-1
3.10.1	Seismic Design Criteria	3.10-1

TABLE OF CONTENTS (Cont'd.)

Section		Page
3.10.2	Analysis, Testing Procedures and Restraint Measures	3.10-3
3.11	ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT	3.11-1
3.11.1	Equipment Identification	3.11-1
3.11.2	Qualification Test and Analysis	3.11-1
3.11.3	Qualification Test Results	3.11-1
3.11.4	Loss of Ventilation	3.11-2
3.11.5	Special Considerations	3.11-2
3A.0	SUPPLEMENTARY INFORMATION ON SEISMIC CATEGORY I STRUCTURES	3A.1-1
3A.1	Inner Cell System	3A.1-1
3A.2	Head Access Area	3A.2-1
3A.3	Control Building	3A.3-1
3A.4	Reactor Service Building (RSB)	3A.4-1
3A.5	Steam Generator Building	3A.5-1
3A.6	Diesel Generator Building	3A.6-1
3A.7	Deleted	
3A.8	Cell Liner Systems	3A.8-1
4.0	REACTOR	4.1-1
4.1	SUMMARY DESCRIPTION	4.1-1
4.1.1	Lower Internals	4.1-1
4.1.2	Upper Internals	4.1-3
4.1.3	Core Restraint	4.1-4
4.1.4	Fuel Blanket and Removable Radial Shield Regions	4.1-4
4.1.5	Design and Performance Characteristics	4.1-9
4.1.6	Loading Conditions and Analysis Techniques	4.1-9
4.1.7	Computer Codes	4.1-10
4.2	MECHANICAL DESIGN	4.2-1
4.2.1	Fuel and Blanket Design	4.2-1
4.2.2	Reactor Vessels Internals	4.2-118
4.2.3	Reactivity Control Systems	4.2-228
4.3	NUCLEAR DESIGN	4.3-1
4.3.1	Design Bases	4.3-1
4.3.2	Description	4.3-3
4.3.3	Analytical Methods	4.3-69
4.3.4	Changes	

TABLE OF CONTENTS (Cont'd.)

Section		Page
4.4	THERMAL AND HYDRAULIC DESIGN	4.4-1
4.4.1	Design Bases	4.4-1
4.4.2	Description	4.4-4
4.4.3	Evaluation	4.4-45
4.4.4	Testing and Verification	4.4-75
4.4.5	Core Instrumentation	4.4-80
5.0	HEAT TRANSPORT AND CONNECTED SYSTEMS	5.1-1
5.1	SUMMARY DESCRIPTION	5.1-1a
5.1.1	Reactor Vessel, Closure Head, and Guard Vessel	5.1-1a
5.1.2	Primary Heat Transport System	5.1-2
5.1.3	Intermediate Heat Transport System	5.1-5
5.1.4	Steam Generator System	5.1-7
5.1.5	Residual Heat Removal System	5.1-8
5.1.6	Auxiliary Liquid Metal System	5.1-9
5.1.7	Features for Heat Transport System Safety	5.1-10
5.1.8	Physical Arrangement	5.1-11
5.2	REACTOR VESSEL, CLOSURE HEAD, AND GUARD VESSEL	5.2-1
5.2.1	Design Basis	5.2-1
5.2.2	Design Parameters	5.2-4b
5.2.3	Special Processes for Fabrication and Inspection	5.2-7
5.2.4	Features for Improved Reliability	5.2-8
5.2.5	Quality Assurance Surveillance	5.2-10d
5.2.6	Materials and Inspections	5.2-11
5.2.7	Packing, Packaging, and Storage	5.2-11a
	Appendix 5.2.A Modifications to the High Temperature Design Rules for Austenitic Stainless Steel	5.2A-1
5.3	PRIMARY HEAT TRANSPORT SYSTEM (PHTS)	5.3-1
5.3.1	Design Bases	5.3-1
5.3.2	Design Description	5.3-9
5.3.3	Design Evaluation	5.3-33
5.3.4	Tests and Inspections	5.3-72
5.4	INTERMEDIATE HEAT TRANSPORT SYSTEM (IHTS)	5.4-1
5.4.1	Design Basis	5.4-1
5.4.2	Design Description	5.4-6
5.4.3	Design Evaluation	5.4-12

TABLE OF CONTENTS (Cont'd.)

Section		Page
5.5	STEAM GENERATOR SYSTEM (SGS)	5.5-1
5.5.1	Design Bases	5.5-1
5.5.2	Design Description	5.5-5
5.5.3	Design Evaluation	5.5-17
5.6	RESIDUAL HEAT REMOVAL SYSTEMS	5.6-1
5.6.1	Steam Generator Auxiliary Heat Removal System (SGAHRS)	5.6-1b
5.6.2	Direct Heat Removal Service (DHRS)	5.6-20
5.7	OVERALL HEAT TRANSPORT SYSTEM EVALUATION	5.7-1
5.7.1	Startup and Shutdown	5.7-1
5.7.2	Load Following Characteristics	5.7-2
5.7.3	Transient Effects	5.7-2a
5.7.4	Evaluation of Thermal Hydraulic Characteristics and Plant Design Heat Transport System Design Transient Summary	5.7-6
6.0	ENGINEERED SAFETY FEATURES	6.1-1
6.1	GENERAL	6.1-1
6.2	CONTAINMENT SYSTEMS	6.2-1
6.2.1	Confinement/Containment Functional Design	6.2-1
6.2.2	Containment Heat Removal	6.2-9
6.2.3	Containment Air Purification and Cleanup System	6.2-9
6.2.4	Containment Isolation Systems	6.2-10
6.2.5	Annulus Filtration System	6.2-14
6.2.6	Reactor Service Building (RSB) Filtration System	6.2-16
6.2.7	Steam Generator Building Aerosol Release Mitigation System Functional Design	6.2-17
6.3	HABITABILITY SYSTEMS	6.3-1
6.3.1	Habitability System Functional Design	6.3-1
6.4	CELL LINER SYSTEM	6.4-1
6.4.1	Design Base	6.4-1
6.4.2	System Design	6.4-1
6.4.3	Design Evaluation	6.4-1
6.4.4	Tests and Inspections	6.4-1
6.4.5	Instrumentation Requirements	6.4-1

TABLE OF CONTENTS (Cont'd.)

Section		Page
6.5	CATCH PAN	6.5-1
6.5.1	Design Base	6.5-1
6.5.2	System Design Description and Evaluation	6.5-1
6.5.3	Tests and Inspections	6.5-1
6.5.4	Instrumentation Requirements	6.5-1
7.0	INSTRUMENTATION AND CONTROLS	7.1-1
7.1	INTRODUCTION	7.1-1
7.1.1	Identification of Safety Related Instrumentation and Control Systems	7.1-1
7.1.2	Identification of Safety Criteria	7.1-1
7.2	REACTOR SHUTDOWN SYSTEM	7.2-1
7.2.1	Description	7.2-1
7.2.2	Analysis	7.2-13
7.3	ENGINEERED SAFETY FEATURE INSTRUMENTATION AND CONTROL	7.3-1
7.3.1	Containment Isolation System	7.3-1
7.3.2	Analysis	7.3-3
7.4	INSTRUMENTATION AND CONTROL SYSTEMS REQUIRED FOR SAFE SHUTDOWN	7.4-1
7.4.1	Steam Generator Auxiliary Heat Removal Instrumentation and Control Systems	7.4-1
7.4.2	Outlet Steam Isolation Instrumentation and Control System	7.4-6
7.4.3	Remote Shutdown System	7.4-8a
7.5	INSTRUMENTATION AND MONITORING SYSTEM	7.5-1
7.5.1	Flux Monitoring System	7.5-1
7.5.2	Heat Transport Instrumentation System	7.5-5
7.5.3	Reactor and Vessel Instrumentation	7.5-13
7.5.4	Fuel Failure Monitoring System	7.5-14
7.5.5	Leak Detection Systems	7.5-18
7.5.6	Sodium-Water Reaction Pressure Relief System (SWRPRS) Instrumentation and Controls	7.5-30
7.5.7	Containment Hydrogen Monitoring	7.5-33b
7.5.8	Containment Vessel Temperature Monitoring	7.5-33b
7.5.9	Containment Pressure Monitoring	7.5-33b
7.5.10	Containment Atmosphere Temperature	7.5-33c
7.5.11	Post Accident Monitoring	7.5-33c

Amend. 68
May 1982

TABLE OF CONTENTS (Cont'd.)

Section		Page
7.6	OTHER INSTRUMENTATION AND CONTROL SYSTEMS REQUIRED FOR SAFETY	7.6-1
7.6.1	Plant Service Water and Chilled Water Instrumentation and Control Systems	7.6-1
7.6.2	Deleted	
7.6.3	Direct Heat Removal Service (DHRS) Instrumentation and Control System	7.6-3
7.6.4	Heating, Ventilating, and Air Conditioning Instrumentation and Control System	7.6-3e
7.6.5	SGB Flooding Protection Subsystem	7.6-3f
7.7	INSTRUMENTATION AND CONTROL SYSTEMS NOT REQUIRED FOR SAFETY	7.7-1
7.7.1	Plant Control System Description	7.7-1
7.7.2	Design Analysis	7.7-16
7.8	PLANT DATA HANDLING AND DISPLAY SYSTEM	7.8-1
7.8.1	Design Description	7.8-1
7.8.2	Design Analysis	7.8-2
7.9	OPERATING CONTROL STATIONS	7.9-1
7.9.1	Design Basis	7.9-1
7.9.2	Control Room	7.9-1
7.9.3	Local Control Stations	7.9-6
7.9.4	Communications	7.9-6
7.9.5	Design Evaluation	7.9-6
8.0	ELECTRIC POWER	8.1-1
8.1	INTRODUCTION	8.1-1
8.1.1	Utility Grid and Interconnections	8.1-1
8.1.2	Plant Electrical Power System	8.1-1
8.1.3	Criteria and Standards	8.1-3
8.2	OFFSITE POWER SYSTEM	8.2-1
8.2.1	Description	8.2-1
8.2.2	Analysis	8.2-4
8.3	ON-SITE POWER SYSTEMS	8.3-1
8.3.1	AC Power Systems	8.3-1
8.3.2	DC Power System	8.3-44

TABLE OF CONTENTS (Cont'd.)

Section		Page
9.0	AUXILIARY SYSTEMS	9.1-1
9.1	FUEL STORAGE AND HANDLING	9.1-1
9.1.1	New Fuel Storage	9.1-3
9.1.2	Spent Fuel Storage	9.1-5
9.1.3	Spent Fuel Cooling and Cleanup System	9.1-20
9.1.4	Fuel Handling System	9.1-33
9.2	NUCLEAR ISLAND GENERAL PURPOSE MAINTENANCE SYSTEM	9.2-1
9.2.1	Design Basis	9.2-1
9.2.2	System Description	9.2-1
9.2.3	Safety Evaluation	9.2-3
9.2-4	Tests and Inspections	9.2-3
9.2-5	Instrumentation Applications	9.2-4
9.3	AUXILIARY LIQUID METAL SYSTEM	9.3-1
9.3.1	Sodium and NaK Receiving System	9.3-1a
9.3.2	Primary Na Storage and Processing	9.3-2
9.3.3	EVS Sodium Processing	9.3-9a
9.3.4	Primary Cold Trap NaK Cooling System	9.3-10
9.3.5	Intermediate Na Processing System	9.3-12
9.4	PIPING AND EQUIPMENT ELECTRICAL HEATING	9.4-1
9.4.1	Design Bases	9.4-1
9.4.2	Systems Description	9.4-2
9.4.3	Safety Evaluation	9.4-3
9.4.4	Tests and Inspections	9.4-3b
9.4.5	Instrumentation Application	9.4-3b
9.5	INERT GAS RECEIVING AND PROCESSING SYSTEM	9.5-1
9.5.1	Argon Distribution Subsystem	9.5-2
9.5.2	Nitrogen Distribution System	9.5-6
9.5.3	Safety Evaluation	9.5-10
9.5.4	Tests and Inspections	9.5-12
9.5.5	Instrumentation Requirements	9.5-12
9.6	HEATING, VENTILATING AND AIR CONDITIONING SYSTEM	9.6-1
9.6.1	Control Building HVAC System	9.6-1
9.6.2	Reactor Containment Building	9.6-12
9.6.3	Reactor Service Building HVAC System	9.6-25
9.6.4	Turbine Generator Building HVAC System	9.6-37
9.6.5	Diesel Generator Building HVAC System	9.6-40
9.6.6	Steam Generator Building HVAC System	9.6-45

TABLE OF CONTENTS (Cont'd.)

Section		Page
9.7	CHILLED WATER SYSTEMS	9.7-1
9.7.1	Normal Chilled Water System	9.7-1
9.7.2	Emergency Chilled Water System	9.7-4
9.7.3	Prevention of Sodium or NaK/Water Interactions	9.7-9
9.7.4	Secondary Coolant Loops (SCL)	9.7-12
9.8	IMPURITY MONITORING AND ANALYSIS SYSTEM	9.8-1
9.8.1	Design Basis	9.8-1
9.8.2	Design Description	9.8-2
9.8.3	Design Evaluation	9.8-5
9.8.4	Tests and Inspection	9.8-7
9.8.5	Instrumentation Requirements	9.8-8
9.9	SERVICE WATER SYSTEMS	9.9-1
9.9.1	Normal Plant Service Water System	9.9-1
9.9.2	Emergency Plant Service Water System	9.9-2
9.9.3	Secondary Service Closed Cooling Water System	9.9-4
9.9.5	River Water Service	9.9-11
9.10	COMPRESSED GAS SYSTEM	9.10-1
9.10.1	Service Air and Instrument Air Systems	9.10-1
9.10.2	Hydrogen System	9.10-3a
9.10.3	Carbon Dioxide System	9.10-4
9.11	COMMUNICATIONS SYSTEM	9.11-1
9.11.1	Design Bases	9.11-1
9.11.2	Description	9.11-3
9.12	LIGHTING SYSTEMS	9.12-1
9.12.1	Normal Lighting System	9.12-1
9.12.2	Standby Lighting Systems	9.12-2
9.12.3	Emergency Lighting System	9.12-3
9.12.4	Design Evaluation	9.12-4
9.13	PLANT FIRE PROTECTION SYSTEM	9.13-1
9.13.1	Non-Sodium Fire Protection System	9.13-1
9.13.2	Sodium Fire Protection System (SFPS)	9.13-13
9.13A	Overall Fire Protection Requirements -- CRBRP Design Compared with APCSB 9.5-1 & ASB 9.5-1	9.13A-1
9.14	DIESEL GENERATOR AUXILIARY SYSTEM	9.14-1
9.14.1	Fuel Oil Storage and Transfer System	9.14-1

Amend. 68
May 1982

TABLE OF CONTENTS (Cont'd.)

Section		Page
9.14.2	Cooling Water System	9.14-2
9.14.3	Starting Air Systems	9.14-4
9.14.4	Lubrication System	9.14-5
9.15	EQUIPMENT AND FLOOR DRAINAGE SYSTEM	9.15-1
9.15.1	Design Bases	9.15-1
9.15.2	System Description	9.15-1
9.15.3	Safety Evaluation	9.15-2
9.15.4	Tests and Inspections	9.15-2
9.15.5	Instrumentation Application	9.15-2
9.16	RECIRCULATION GAS COOLING SYSTEM	9.16-1
9.16.1	Design Basis	9.16-1
9.16.2	System Description	9.16-1
9.16.3	Safety Evaluation	9.16-6
9.16.4	Tests and Inspection	9.16-7
9.16.5	Instrumentation and Control	9.16-7
10.0	STEAM AND POWER CONVERSION SYSTEM	10.1-1
10.1	SUMMARY DESCRIPTION	10.1-1
10.2	TURBINE GENERATOR	10.2-1
10.2.1	Design Bases	10.2-1
10.2.2	Description	10.2-1a
10.2.3	Turbine Missiles	10.2-5
10.2.4	Evaluation	10.2-9
10.3	MAIN STEAM SUPPLY SYSTEM	10.3-1
10.3.1	Design Bases	10.3-1
10.3.2	Description	10.3-1
10.3.3	Evaluation	10.3-2
10.3.4	Inspection and Testing Requirements	10.3-2
10.3.5	Water Chemistry	10.3-3
10.4	OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEM	10.4-1
10.4.1	Condenser	10.4-1
10.4.2	Condenser Air Removal System	10.4-2
10.4.3	Turbine Gland Sealing System	10.4.3
10.4.4	Turbine Bypass System	10.4-4
10.4.5	Circulating Water System	10.4-5
10.4.6	Condensate Cleanup System	10.4-7

TABLE OF CONTENTS (Cont'd.)

Section		Page
10.4.7	Condensate and Feedwater Systems	10.4-9
10.4.8	Steam Generator Blowdown System	10.4-14
11.0	RADIOACTIVE WASTE MANAGEMENT	11.1-1
11.1	SOURCE TERMS	11.1-1
11.1.1	Modes of Radioactive Waste Production	11.1-1
11.1.2	Activation Product Source Strength Models	11.1-2
11.1.3	Fission Product and Plutonium Release Models	11.1-5
11.1.4	Tritium Production Sources	11.1-7
11.1.5	Summary of Design Bases for Deposition of Radioactivity in Primary Sodium on Reactor and Primary Heat Transfer Surfaces and Within Reactor Auxiliary Systems	11.1-7
11.1.6	Leakage Rates	11.1-10
11.2	LIQUID WASTE SYSTEM	11.2-1
11.2.1	Design Objectives	11.2-1
11.2.2	System Description	11.2-2
11.2.3	System Design	11.2-4
11.2.4	Operating Procedures and Performance Tests	11.2-5
11.2.5	Estimated Releases	11.2-6
11.2.6	Release Points	11.2-6
11.2.7	Dilution Factors	11.2-7
11.2.8	Estimated Doses	11.2-8
	Appendix 11.2A Dose Models: Liquid Effluents	11.2A-1
11.3	GASEOUS WASTE SYSTEM	11.3-1
11.3.1	Design Base	11.3-1
11.3.2	System Description	11.3-1
11.3.3	System Design	11.3-10
11.3.4	Operating Procedures and Performance Tests	11.3-11a
11.3.5	Estimated Releases	11.3-14
11.3.6	Release Points	11.3-15
11.3.7	Dilution Factors	11.3-17
11.3.8	Dose Estimates	11.3-17
	Appendix 11.3A Dose Models: Gaseous Effluents	11.3A-1
11.4	PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEM	11.4-1
11.4.1	Design Objectives	11.4-1
11.4.2	Continuous Monitoring/Sampling	11.4-2
11.4.3	Sampling	11.4-3

TABLE OF CONTENTS (Cont'd.)

Section		Page
11.5	SOLID WASTE SYSTEM	11.5-1
11.5.1	Design Objectives	11.5-1
11.5.2	System Inputs	11.5-1
11.5.3	Equipment Description	11.5-1
11.5.4	Expected Volumes	11.5-3
11.5.5	Packaging	11.5-4
11.5.6	Storage Facilities	11.5-4
11.5.7	Shipment	11.5-4
11.6	OFFSITE RADIOLOGICAL MONITORING PROGRAM	11.6-1
11.6.1	Expected Background	11.6-1
11.6.2	Critical Pathways to Man	11.6-2
11.6.3	Sampling Media, Locations and Frequencies	11.6-4
11.6.4	Analytical Sensitivity	11.6-4
11.6.5	Data Analysis and Presentation	11.6-4
11.6.6	Program Statistical Sensitivity	11.6-5
12.0	RADIATION PROTECTION	12.1-1
12.1	SHIELDING	12.1-1
12.1.1	Design Objectives	12.1-1
12.1.2	Design Description	12.1-3
12.1.3	Source Terms	12.1-13
12.1.4	Area Radiation Monitoring	12.1-23
12.1.5	Estimates of Exposure	12.1-24
	Appendix to Section 12.1	12.1A-1
12.2	VENTILATION	12.2-1
12.2.1	Design Objectives	12.2-1
12.2.2	Design Description	12.2-1
12.2.3	Source Terms	12.2-3
12.2.4	Airborne Radioactivity Monitoring	12.2-3
12.2.5	Inhalation Doses	12.2-5
12.3	HEALTH PHYSICS PROGRAM	12.3-1
12.3.1	Program Objectives	12.3-1
12.3.2	Facilities and Equipment	12.3-3
12.3.3	Personnel Dosimetry	12.3-6
12.3.4	Estimated Occupancy Times	12.3-7
	Appendix 12A - Information Related to ALARA for Occupational Radiation Exposures	12A-1

TABLE OF CONTENTS (Cont'd.)

Section		Page
13.0	CONDUCT OF OPERATIONS	13.1-1
13.1	ORGANIZATIONAL STRUCTURE OF THE APPLICANT	13.1-1
13.1.1	Project Organization	13.1-1
13.1.2	Operating Organization	13.1-5
13.1.3	Qualification Requirements for Nuclear Plant Personnel	13.1-12
13.2	TRAINING PROGRAM	13.2-1
13.2.1	Program Description	13.2-1
13.2.2	Retraining Program	13.2-6
13.2.3	Replacement Training	13.2-6
13.2.4	Records	13.2-6
13.3	EMERGENCY PLANNING	13.3-1
13.3.1	General	13.3-1
13.3.2	Emergency Organization	13.3-2
13.3.3	Coordination with Offsite Groups	13.3-5
13.3.4	Emergency Action Levels	13.3-6
13.3.5	Protective Measures	13.3-7
13.3.6	Review and Updating	13.3-7
13.3.7	Medical Support	13.3-7
13.3.8	Exercises and Drills	13.3-8
13.3.9	Training	13.3-8
13.3.10	Recovery and Reentry	13.3-9
13.3.11	Implementation	13.3-9
	Appendix 13.3A	13.3A-1
13.4	REVIEW AND AUDIT	13.4-1
13.4.1	Review and Audit - Construction	13.4-1
13.4.2	Review and Audit - Test and Operation	13.4-1
13.5	PLANT PROCEDURES	13.5-1
13.5.1	General	13.5-1
13.5.2	Normal Operating Instructions	13.5-1
13.5.3	Abnormal Operating Instructions	13.5-2
13.5.4	Emergency Operating Instructions	13.5-2
13.5.5	Maintenance Instructions	13.5-3

Amend. 68
May 1982

TABLE OF CONTENTS (Cont'd.)

Section		Page
13.5.6	Surveillance Instructions	13.5-4
13.5.7	Technical Instructions	13.5-4
13.5.8	Sections Instruction Letters	13.5-4
13.5.9	Site Emergency Plans	13.5-4
13.5.10	Radiation Control Instructions	13.5-4
13.6	PLANT RECORDS	13.6-1
13.6.1	Plant History	13.6-1
13.6.2	Operating Records	13.6-1
13.6.3	Event Records	13.6-1
13.7	RADIOLOGICAL SECURITY	13.7-1
13.7.1	Organization and Personnel	13.7-1
13.7.2	Plant Design	13.7-3
13.7.3	Security Plan	13.7-6
14.0	INITIAL TESTS AND OPERATION	14.1-1
14.1	DESCRIPTION OF TEST PROGRAMS	14.1-1
14.1.1	Preoperational Test Programs	14.1-2
14.1.2	Startup Test Program	14.1-2
14.1.3	Administration of Test Program	14.1-3
14.1.4	Test Objectives of First-of-a-Kind Principal Design Features	14.1-6
14.2	AUGMENTATION OF OPERATOR'S STAFF FOR INITIAL TESTS AND OPERATION	14.2-1
15.0	ACCIDENT ANALYSES	15.1-1
15.1	INTRODUCTION	15.1-1
15.1.1	Design Approach to Safety	15.1-1
15.1.2	Requirements and Criteria for Assessment of Fuel and Blanket Rod Transient Performance	15.1-50
15.1.3	Control Rod Shutdown Rate and Plant Protection System Trip Settings	15.1-93
15.1.4	Effect of Design Changes on Analyses of Accident Events	15.1-105
15.2	REACTIVITY INSERTION DESIGN EVENTS - INTRODUCTION	15.2-1
15.2.1	Anticipated Events	15.2-5
15.2.2	Unlikely Events	15.2-34
15.2.3	Extremely Unlikely Events	15.2-51

TABLE OF CONTENTS (Cont'd.)

Section		Page
15.3	UNDERCOOLING DESIGN EVENTS - INTRODUCTION	15.3-1
15.3.1	Anticipated Events	15.3-6
15.3.2	Unlikely Events	15.3-29
15.3.3	Extremely Unlikely Events	15.3-38
15.4	LOCAL FAILURE EVENTS - INTRODUCTION	15.4-1
15.4.1	Fuel Assembly	15.4-2
15.4.2	Control Assemblies	15.4-42
15.4.3	Radial Blanket Assembly	15.4-51
15.5	FUEL HANDLING AND STORAGE EVENTS - INTRODUCTION	15.5-1
15.5.1	Anticipated Events (None)	15.5-4
15.5.2	Unlikely Events	15.5-4
15.5.3	Extremely Unlikely Events	15.5-23
15.6	SODIUM SPILLS - INTRODUCTION	15.6-1
15.6.1	Extremely Unlikely Events	15.6-4
15.7	OTHER EVENTS - INTRODUCTION	15.7-1
15.7.1	Anticipated Events	15.7-3
15.7.2	Unlikely Events	15.7-9
15.7.3	Extremely Unlikely Events	15.7-18
15.A	Appendix 15.A - Radiological Source Term for Assessment of Site Suitability	15.A-1
16.0	TECHNICAL SPECIFICATIONS	16.1-1
16.1	DEFINITIONS	16.1-1
16.1.1	Reactor Operating Condition	16.1-1
16.1.2	Reactor Core	16.1-2
16.1.3	Plant Protection System Instrumentation	16.1-3
16.1.4	Safety Limit	16.1-5
16.1.5	Limiting Safety System Setting (LSSS)	16.1-5
16.1.6	Limiting Conditions for Operation (LCO)	16.1-6
16.1.7	Surveillance Requirements	16.1-6
16.1.8	Containment Integrity	16.1-6
16.1.9	Abnormal Occurrence	16.1-6

TABLE OF CONTENTS (Cont'd.)

Section		Page
16.2	SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS	16.2-1
16.2.1	Safety Limit, Reactor Core	16.2-1
16.2.2	Limiting Safety System Settings	16.2-1
16.3	LIMITING CONDITIONS FOR OPERATION	16.3-1
16.3.1	Reactor Operating Conditions	16.3-1
16.3.2	Primary Heat Transport System (PHTS)	16.3-2
16.3.3	Intermediate Heat Transport Coolant System	16.3-6
16.3.4	Steam Generation System (SGS)	16.3-7
16.3.5	Auxiliary Liquid Metal System	16.3-12
16.3.6	Inert Gas System Cover Gas Purification System	16.3-13
16.3.7	Auxiliary Cooling System	16.3-14
16.3.8	Containment Integrity	16.3-21
16.3.9	Auxiliary Electrical System	16.3-21
16.3.10	Refueling	16.3-24
16.3.11	Effluent Release	16.3-27
16.3.12	Reactivity and Control Rod Limits	16.3-31
16.3.13	Plant Protection System	16.3-34
16.4	SURVEILLANCE REQUIREMENTS	16.4-1
16.4.1	Operational Safety Review	16.4-1
16.4.2	Reactor Coolant System Surveillance	16.4-1
16.4.3	Containment Tests	16.4-3
16.4.4	HVAC and Radioactive Effluents	16.4-6
16.4.5	Emergency Power System Periodic Tests	16.4-10
16.4.6	Inert Gas System	16.4-13
16.4.7	Reactivity Anomalies	16.4-13
16.4.8	Pressure and Leakage Rate Test of RAPS Cold Box Cell	16.4-15
16.4.9	Pressure and Leakage Rate Test of RAPS Noble Gas Storage Vessel Cell	16.4-15a
16.5	DESIGN FEATURES	16.5-1
16.5.1	Site	16.5-1
16.5.2	Confinement/Containment	16.5-1
16.5.3	Reactor	16.5-2
16.5.4	Heat Transport System	16.5-5
16.5.5	Fuel Storage	16.5-7
16.6	ADMINISTRATIVE CONTROLS	16.6-1
16.6.1	Organization	16.6-1
16.6.2	Review and Audit	16.6-1
16.6.3	Instructions	16.6-4
16.6.4	Actions to be Taken in the Event of Reportable Occurrence in Plant Operation	16.6-6

Amend. 68
May 1982

TABLE OF CONTENTS (Cont'd.)

Section		Page
16.6.5	Action to be Taken in the Event a Safety Limit Is Exceeded	16.6-6
16.6.6	Station Operating Records	16.6-6
16.6.7	Reporting Requirements	16.6-7
16.6.8	Minimum Staffing	16.6-8
17.0	QUALITY ASSURANCE - INTRODUCTION	17.0-1
17.0.1	Scope	17.0-1
17.0.2	Quality Philosophy	17.0-1
17.0.3	Participants	17.0-2
17.0.4	Project Phase Approach	17.0-3
17.0.5	Applicability	17.0-3
17.1	QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION	17.1-1
17.1.1	Organization	17.1-1
17.1.2	Quality Assurance Program	17.1-2
17.1.3	References Referred to in the Text	17.1-6
17.1.4	Acronyms Used in Chapter 17 Text and Appendices	17.1-6a

TABLE OF CONTENTS (Cont'd.)

Section		Page
Appendix 17A	A Description of the Owner Quality Assurance Program	17A-1
Appendix 17B	A Description of the Fuel Supplier Quality Assurance Program	17B-1
Appendix 17C	A Description of the Balance of Plant Supply Quality Assurance Program	17C-1
Appendix 17D	A Description of the ARD Lead Reactor Manufacturer Quality Assurance Program	17D-1
Appendix 17E	A Description of the Architect-Engineer Quality Assurance Program	17E-1
Appendix 17F	A Description of the Constructor Quality Assurance Program	17F-1
Appendix 17G	RDT Standard F2-2, 1973, Quality Assurance Program Requirements	17G-1
Appendix 17H	A Description of the ARD Reactor Manufacturer Quality Assurance Program	17H-1
Appendix 17I	A Description of the GE-ARSD-RM Quality Assurance Program	17I-1
Appendix 17J	A Description of the ESG-RM Quality Assurance Program	17J-1
Appendix A	Computer Codes	A-1
Appendix B	General Plant Transient Data	B-1
Appendix C	Safety Related Reliability Program	C.1-1
Appendix D	Deleted	
Appendix E	Deleted	
Appendix F	Deleted	
Appendix G	Plan for Inservice and Preservice Inspections	G-1
Appendix H	Post TMI Requirements	H-1

APPENDIX G

CLINCH RIVER BREEDER REACTOR PLANT

PLAN FOR INSERVICE AND PRESERVICE INSPECTIONS

Amend. 65
Feb. 1982

APPENDIX G

TABLE OF CONTENTS

	PAGE
TABLE OF CONTENTS	G-i
CRBRP - PLAN FOR INSERVICE INSPECTION	G-ii
PART I - CRBRP INSERVICE AND PRESERVICE INSPECTION REQUIREMENTS FOR LIQUID METAL AND COVER GAS SYSTEMS	G-iii
PART II - CRBRP INSERVICE AND PRESERVICE INSPECTION REQUIREMENTS FOR WATER/STEAM SYSTEMS	G-57
PART III - CRBRP INSERVICE AND PRESERVICE INSPECTION REQUIREMENTS FOR REACTOR CONTAINMENT VESSEL, CELL LINERS, AND SODIUM FIRE SURPRESSION FEATURES.	G-58

CRBRP PLAN FOR INSERVICE AND PRESERVICE INSPECTION

INTRODUCTION

Clinch River Breeder Reactor Plant General Design Criterion 30, "Inspection of Reactor Coolant Boundary", requires that components which are part of the Reactor Coolant Boundary shall be designed to permit (1) periodic inspection and testing of areas and features important to safety to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program. These components are also ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Code Class 1 and 2 components.

In addition, General Design Criterion 33, "Inspection and Surveillance of Intermediate Coolant Boundary"; Criterion 36, "Inspection of Reactor Residual Heat Extraction Systems"; Criterion 39, "Inspection of Additional Cooling Systems"; Criterion 44, "Provisions for Containment Testing and Inspection"; and Criterion 50, "Inspection of Containment Atmosphere Cleanup Systems" require that the subject systems be designed to permit appropriate periodic inspection of important component parts to assure system integrity and capability. These components are ASME Code Class 2 and 3.

To meet these requirements, the CRBRP Inservice and Preservice Inspection Plan will be based upon ASME Code Section XI, Division 3, "Rules for Inspection and Testing of Components of Liquid Metal Cooled Plants," and Code Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components." Code Division 3 requirements, which apply to liquid metal and cover gas systems, are addressed in Part I of this Plan; Division 1 requirements, for water/steam systems, are discussd in Part II. Part III addresses the inservice inspection plans for the Reactor Containment Vessel, cell liners and sodium fire surpression features.

Federal codes and regulations are currently being reviewed for applicability to CRBRP inservice inspection. The inservice inspection plan will be revised as required upon completion of that review.

PSAR APPENDIX G
PART I
TABLE OF CONTENTS

SECTION		PAGE
	Table of Contents	G-iii
	List of Tables	G-iv
	List of Addenda	G-v
	List of Applicable Documents	G-v
1.0	Purpose	G-1
2.0	Summary	G-2
2.1	Sodium and Cover Gas Systems	G-2
3.0	Scope	G-3
4.0	General Approach	G-4
4.1	Basis	G-4
5.0	Sodium and Cover Gas Systems	G-7
6.0	Inservice Pump and Valve Tests	G-28
6.1	Inservice Pump Testing	G-28
6.2	Inservice Valve Testing	G-28
7.0	Component Supports and Snubbers	G-29
8.0	Preservice Inspection Requirements	G-33
9.0	Inspection Intervals	G-34
10.0	Examination Methods	G-35
10.1	Continuous Monitoring - Class 1 Components Requirements for Sodium Leak Detection	G-36
10.2	Continuous Monitoring - Class 2 and 3 Components Requirements for Sodium Leak Detection	G-36
10.3	Radioactive Cover Gas Leakage Monitoring - Class 1, 2 and 3	G-37
10.4	Sodium to Sodium Leak Detection	G-37

LIST OF TABLES

		PAGE
TABLE 5-1 -	CRBRP SODIUM AND COVER GAS SYSTEMS - CLASS 1	
	Liquid Metal Retaining Welds in Vessels Protected by Guard Vessels	G-9
	Liquid Metal Retaining Welds in Vessels Not Protected by Guard Vessels	G-10
	Cover Gas Retaining Components	G-11
	Dissimilar Metal Welds	G-12
	Bolting	G-13
	Integral Attachments for Vessels	G-14
	Liquid Metal Retaining Welds in Piping Protected by Guard Pipe or Tank	G-15
	Liquid Metal Piping Not Protected by Guard Pipe or Tank	G-16
	Integral Attachments for Piping and Valves	G-17
	Liquid Metal Retaining Valves	G-18
	Reactor Internal Components	G-19
TABLE 5-2	CRBRP SODIUM AND COVER GAS SYSTEMS - CLASS 2	
	Liquid Metal Retaining Welds in Vessels	G-20
	Welds in Guard Tanks and Guard Pipes	G-21
	Integral Attachments	G-22
	Bolting	G-23
	Dissimilar Metal Welds	G-24
	Liquid Metal Piping and Valves	G-25
	Steam Generator Tubing	G-26
TABLE 5-3	CRBRP SODIUM AND COVER GAS SYSTEMS - CLASS 3	
	Class 3 Components	G-27

TABLE 7-1 CRBRP COMPONENT SUPPORTS FOR SODIUM AND COVER GAS SYSTEMS

Plate and Shell Type Supports	G-30
Linear Type Supports	G-31
Component Standard Supports	G-32

LIST OF ADDENDA

ADDENDA		PAGE
A	Justifications for ASME Code Exceptions	G-38

LIST OF APPLICABLE DOCUMENTS

The below listed documents form a part of Part I of this Appendix to the extent specified herein.

- ASME Boiler and Pressure Vessel Code, Section XI Division 1, 1980 Edition.
- ASME Boiler and Pressure Vessel Code, Section XI Division 3, Winter 1981 Addenda.

NOTE: Section XI Division 3 is not currently recognized in 10 CFR 50.55a.

1.0 Purpose

This document specifies the overall requirements for Inservice and Preservice Inspection to be performed on sodium and cover gas systems and components in the Clinch River Breeder Reactor Plant (CRBRP). The Inservice and Preservice Inspection requirements given herein are to assure protection of public health and safety and are applicable to those Class 1, Class 2 and Class 3 systems and components constructed in accordance with ASME Boiler and Pressure Vessel Code Section III.

2.0 Summary

2.1 Sodium and Cover Gas Systems

The objective of inservice inspection of sodium and cover gas systems and components in CRBRP is to provide a continuing assurance of safe operation. Components constructed to ASME rules are considered to be safe for initial operation within the events and conditions specified in the Design Specification. During operation, very small liquid metal or cover gas leaks, should they develop, are easy to detect. Therefore, emphasis is placed upon leak detection and visual examinations as the primary techniques for inservice inspections of sodium and cover gas systems. These monitoring and visual examination programs will provide the level of inspection necessary to insure the continuing safety of the plant. In addition, design features have been provided to permit access for performance of volumetric examination of selected PHTS welds using the high temperature ultrasonic technology currently under development. The CRBRP requirements for Class 1, Class 2 and Class 3 sodium and cover gas systems inservice inspection are based on ASME Code Section XI, Division 3 as modified and interpreted by this document for CRBRP use. Since ASME Section XI Division 3 does not presently define sodium and cover gas pump and valve tests, these tests will be conducted in accordance with ASME Section XI, Division 1, modified as necessary to accommodate the design features of the sodium/cover gas components.

NOTE: In this and all other cases, the appropriate revision of the ASME Code will be as defined in the list of Applicable Documents.

3.0 Scope

The inservice inspection requirements presented in this document shall apply to the sodium and cover gas systems or portions thereof which are constructed in accordance with ASME Code Section III Class 1, 2 and 3.

4.0 General Approach

The general approach of inservice inspection for CRBRP is based on ensuring, through continuous monitoring, periodic examination, and testing that the system structural and leak tight integrity is maintained for those systems and components whose failure could adversely affect core reactivity control or core cooling, or result in an uncontrolled release of radioactivity to the environment.

Continuous monitoring of sodium boundaries is provided by a leak detection system capable of detecting a very small amount of sodium leakage. Continuous monitoring for leakage from cover gas boundaries is provided by radioactive gas detectors. Periodic visual examinations and tests are performed at specified intervals to examine components for signs of degradation.

4.1 Basis

The basis underlying the general approach to inservice inspection for CRBRP liquid metal systems is that leakage, if it occurs, will be detected long before such leakage could progress to a rupture or break or result in an unsafe condition. Factors on which the "leak before break" premise is based are as follows:*

(1) The principal structural material of major components and piping is austenitic stainless steel, which possesses good toughness and ductility over the temperature range of CRBRP.

(2) All liquid metal and cover gas retaining welds are subject to volumetric examination and surface examination at the time of construction, which ensures that no initial defect of significant size is present before the operation.

(3) The vapor pressure of the sodium coolant is low, and the operating pressures in the various sections of the Primary Heat Transport System are less than 200 psi, thus constituting a relatively small source of stored energy. Sufficient coolant inventory is provided and the system elevations are arranged to assure a safe reactor coolant level after any breach of the coolant boundary.

(4) The cover gas pressure within the reactor vessel enclosure is less than 1.0 psig. The closure head (22 inches in thickness), risers, nozzles, and associated welds are not stressed to any significant extent by the internal pressure or by normal operational loads, and hence rapid failure or crack propagation is not credible. It is further noted that complete loss of cover gas pressure, if it were to occur, would not have

---

*For details, refer to "CRBRP, Integrity of Primary and Intermediate Heat Transport System Piping in Containment," WARD-D-0185, dated September 1977. (Reference 2, PSAR Section 1.6)

an adverse effect on reactor coolant level, core cooling capability, or nuclear reactivity control, nor would it result in an unacceptable offsite radiological hazard.

(5) A crack size about equal to the diameter of the pipe is calculated to be necessary for the potential for rapid failure to exist in the primary sodium piping. Fatigue tests on elbows show that crack growth, if it occurs, is predominantly through-wall. Hence, detectable leakage through cracks would occur long before the crack length had propograted to a critical size. The calculational model used for crack size determination was developed from results of tests on intentionally flawed elbows.

(6) Diverse redundant leak detection capability is provided by aerosol-type leak detectors, radiation monitors, smoke detectors, continuity type detectors, and level sensors. Tests have shown that a sodium leak rate of 100 gms/hr or greater can be detected.

(7) Other factors contributing to safety and inspectability in CRBRP are (1) guard vessels, (2) elevated primary heat transport system piping, and (3) component and system arrangement.

- Guard vessels are provided as "second vessels", around the reactor vessel and attached sodium inlet and outlet piping, the intermediate heat exchangers, and the primary sodium pumps. The guard vessels are designed such that an adequate level of sodium is maintained in the reactor vessel to ensure core cooling should a leak develop in any components within guard vessels.

- The elevation of the primary sodium piping outside of the guard vessels is such that, should a leak occur in this piping, the level of sodium above the reactor core will not drop below the level necessary to ensure adequate core cooling.

- The spacing between components and between components and building structures is adequate for installation of examination equipment, maintenance and removal and laydown of insulation heaters, and leak detection equipment.

- Insulation assemblies are designed to be removed, where appropriate, for inservice inspection. Insulation requirements for insulation provide for a minimum of chlorides and other corrosive materials. Insulation for sodium system components and piping is designed to provide an annulus surrounding the component to assure circulation by natural convection for uniform heating and leak detection.

5.0 Sodium and Cover Gas Systems

Inservice inspection of ASME Code Class 1, Class 2 and Class 3 sodium and cover gas components in CRBRP shall be in accordance with the intent of ASME Section XI, Division 3 as defined in Tables 5-1, 5-2 and 5-3. The examination categories, identification of parts to be examined, examination methods, extent and frequency of examinations are contained in these tables. Justifications for relief from specific ASME Code requirements are contained in Addenda A of this document. The inspection intervals for examinations shall comply with Program B of the ASME Code as shown in Section 9.0 of this document.

Liquid metal to gas leak detection equipment used for continuous monitoring of Class 1 piping and components shall be diverse redundant, i.e., a given leak shall be detectable by at least two channels, each of which employs a different detection principle. Such equipment shall be installed to monitor each component and liquid metal appurtenance of the system for liquid metal to gas leakage.

For Class 2 piping and components, liquid metal to gas leakage continuous monitoring devices shall be installed on, under, or near each component and liquid metal appurtenance of the system. For both Class 1 and Class 2 systems these devices shall have sensitivities and response times to alert operating personnel for corrective action to be taken to prevent (1) substantial weakening of a component due to corrosion resulting from small leaks and (2) damage to other components and cells resulting from large leaks.

Throughout this document, references to IMA, IMB, IMC, etc., will be specific paragraphs contained in ASME Code Section XI, Division 3.

The information included in the requirements tables in this document should be interpreted to mean as follows:

Code Category provides a direct reference to the categorization of examinations used in ASME Section XI Division 3.

Item No. provides a direct reference to the specific examination requirements defined in ASME Section XI Division 1 and Division 3.

B - Class 1, C - Class 2, D - Class 3, F - Supports

Parts Examined or Components Subject to Test and Examination describes the component, support, or weld that is to be examined.

Examination Method identifies the method of examination required for the system component, support or weld.

Extent of Examination for Each Inspection Interval or Extent of Examination and Inspection Interval identifies the degree of each examination that must be completed within each of the three CRBRP ten year inspection intervals. Requirements are for each interval unless indicated otherwise.

NOTE: These Tables identify the examination requirements from ASME Section X, Division 3 in all instances. In those instances where CRBPR requirements are different from the code, these differences are identified and the justification for the variance is referenced. In those instances where CRBRP has specified requirements not contained in the code, they are identified as such in the Tables.

TABLE 5-1

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY B-A	LIQUID METAL RETAINING WELDS IN VESSELS PROTECTED BY GUARD VESSELS		CODE CLASS 1
ITEM NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
B1.10	Reactor Vessel Longitudinal and circumferential shell welds; meridional and circumferential bottom head welds; nozzle-to-vessel welds; nozzle-to-pipe welds	Visual, VTM-2	33% of all welds(1)
B1.11	Same as B1.10	Continuous Monitoring(2)	All welds
B1.20	Primary Pump Tanks All liquid metal retaining welds - including nozzle-to-tank and nozzle-to-pipe welds	Visual, VTM-2	33% of all welds(1)
B1.21	Same as B1.20	Continuous Monitoring(2)	All welds
B1.30	Intermediate Heat Exchanger All liquid metal retaining welds in shell; nozzle-to-shell and nozzle-to-pipe welds	Visual, VTM-2	33% of all welds(1)
B1.31	Same as B1.30	Continuous Monitoring(2)(3)	All welds
B1.X(4)	IHX tubesheet-to-shell welds; tubesheet-to-head welds	Continuous Monitoring	All welds

(1) Different sample each interval to cover essentially 100% by end of 3rd interval.
(2) As described in IMA-2241.
(3) As described in IMB-5300.
(4) This requirement is not defined in ASME Setion XI Division 3.

TABLE 5-1

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY B-B	LIQUID METAL RETAINING WELDS IN VESSELS NOT PROTECTED BY GUARD VESSELS		CODE CLASS 1
ITEM NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
B2.10	All liquid metal retaining welds, including nozzle-to-tank welds and nozzle-to-pipe welds	Visual, VTM-2	33% of all welds(1)
B2.11	Same as B2.10	Continuous Monitoring(2)	All welds

(1) Different sample each interval to cover essentially 100% by end of 3rd interval.
(2) As described in IMA 2241.

TABLE 5-1

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY B-C		COVER GAS RETAINING COMPONENTS	CODE CLASS 1
ITEM[3] NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Reactor Cover (Rotating Plugs and Deck)[1]		
B3.10	Non-structural Seal Welds	Continuous Monitoring[2]	External or adjacent to surface monitored (continuously)
B3.11	Structural and Seal Welds (In Austenitic S.S. only)	Continuous Monitoring[2]	Same
B3.12	Non-welded gas seals	Continuous Monitoring[2]	Same
	Control Rod Drives and Other Above-Cover Mechanisms Containing Cover Gas		
B3.20	Welds in Outer Housing	Continuous Monitoring[2]	Same
B3.21	Non-welded gas seals	Continuous Monitoring[2]	Same
	Primary Cover Gas Systems		
B3.30	Reactor Vessel Welds	Continuous Monitoring[2]	Same
B3.31	Piping Welds	Continuous Monitoring[2]	Same
B3.32	Valves (including stem seals)	Continuous Monitoring[2]	Same

(1) As described in IMA-2100(p).
(2) As described in IMA-2242.
(3) ASME Code Item Nos. B3.33 through B4.20 are not applicable to CRBRP.

TABLE 5-1

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY B-F		DISSIMILAR METAL WELDS	CODE CLASS 1
ITEM[1] NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Reactor Vessel		
B5.10	Circumferential Shell Weld	Visual, VTM-2[4]	100% of all welds
B5.11	Same as B5.10	Continuous Monitoring[2]	100% of all welds
	Cover Gas Retaining Piping (Primary Systems Only)		
B5.30	Dissimilar Metals Welds at Freeze Vents	Continuous Monitoring[3]	100% of all welds

(1) ASME Code Items B5.20 and B5.21 are not applicable to CRBRP.
(2) As described in IMA-2242 for gas leakage, as described in IMA-2241 for sodium leakage.
(3) As described in IMA-2242.
(4) See Exception A.1 in Addenda A.

TABLE 5-1

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY B-G		BOLTING	CODE CLASS 1
ITEM NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Cover Gas Space Closures		
B6.10	Bolting	Continuous Monitoring	All closures
	(Reactor Internal Bolting See Category B - N)		

TABLE 5-1

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY B-H		INTEGRAL ATTACHMENTS FOR VESSELS	CODE CLASS 1
ITEM NO.[1]	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Other Vessels		
B7.40	Integrally Welded Attachments	Surface or Volumetric	Attachment weld joining attachment to liquid metal retaining membrane[2]

(1) ASME Code Nos. B7.10, B7.20 and B7.30 are not applicable to CRBRP.
(2) Examination includes essentially 100% of the length of the weld to vessel and the integral attachment weld to a cast or forged integral attachment to the vessel, as applicable.

TABLE 5-1

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY B-J-1 LIQUID METAL RETAINING WELDS IN PIPING PROTECTED BY GUARD PIPE OR TANK CODE CLASS 1

ITEM[4] NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Heat Transport Loop Piping		
B8.10	Circumferential Welds including nozzle-to-pipe welds	Visual, VTM-2	33% of Circumferential Weld Joints in PHTS[2]
B8.11	Same as B8.10	Continuous Monitoring[1]	All welds
B8.12	Longitudinal Welds	Visual, VTM-2	Welds[3]
B8.13	Same as B8.12	Continuous Monitoring	All welds

(1) As described in IMA-2241.
(2) Succeeding examinations shall not repeat previous examinations until essentially 100% of joints have been examined.
(3) The examination shall include not more than 12 inches of each longitudinal weld intersecting the circumferential welds required to be examined.
(4) ASME Code Item B.14 is not applicable to CRBRP.

TABLE 5-1

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY B-J-2	LIQUID METAL PIPING NOT PROTECTED BY GUARD PIPE OR TANK		CODE CLASS 1
ITEM NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Heat Transport Loop Piping		
B9.10	Piping Boundary	Visual, VTM-2	33% of boundary(1)
B9.11	Piping Boundary	Continuous Monitoring	100% of boundary
	Other Piping		
B9.20	Piping Boundary	Continuous Monitoring	100% of boundary

(1) Different sample each interval to cover essentially 100% of boundary by end of 3rd interval.

TABLE 5-1

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY B-K		INTEGRAL ATTACHMENTS FOR PIPING AND VALVES	CODE CLASS 1
ITEM[1] NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Valves		
B10.20	Integrally Welded Attachments:		
	CLCV Snubber Attachment Welds	Visual, VTM-3	Welds on one valve, different valve each interval. (Ref. Addenda A, Sect. A.3)

(1) ASME Code Item Nos. B10.10 and B10.11 are not contained in CRBRP.

TABLE 5-1

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY B-M		LIQUID METAL RETAINING VALVES	CODE CLASS 1
ITEM NO.[1]	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Primary Heat Transport Loops		
B11.10	Valve Body (CLCV)	Visual, VTM-2	Essentially 100%
B11.11	Same as B11.10	Continuous Monitoring	Continuously
	Auxiliary Liquid Metal Systems		
B11.30	Valves	Continuous Monitoring	Continuously

(1) ASME Code Item No. B11.20 is not contained in CRBRP.

TABLE 5-1

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY B-N		REACTOR INTERNAL COMPONENTS	CODE CLASS 1
ITEM NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Integrally Welded Internal Components		
B12.10	Core Support Structure		
B12.11	Passive Core Restraint	CRBRP is in the course of evaluating the ASME Section XI Division 3 requirements and will develop CRBRP requirements based upon the results of this evaluation.	
B12.12	Coolant Flow Ducts and Plenums		
B12.13	Thermal Barriers		
	Removal Internal Components		
B12.20	Core Support Items		
B12.21	Other Internal Items		

TABLE 5-2

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY C-A		LIQUID METAL RETAINING WELDS IN VESSELS	CODE CLASS 2
ITEM[1] NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Intermediate Heat Transport System Pumps		
C1.10	Liquid Metal Retaining Boundary	Visual, VTM-2	33% of all pumps[2]
C1.11	Same as C1.10	Continuous Monitoring	All welds
	Steam Generator Shells		
C1.20	Liquid Metal Retaining Boundary	Visual, VTM-2	33% of all shells[2]
C1.21	Same as C1.20	Continuous Monitoring	All welds
	Other Heat Exchanger Shells		
C1.30	Liquid Metal Retaining Boundary	Visual, VTM-2	33% of all shells[2]
C1.31	Same as C1.30	Continuous Monitoring	All welds
	Other Vessels, Normally Filled		
C1.40	Liquid Metal Retaining Boundary	Visual, VTM-2	33% of all vessels[2]
C1.41	Same as C1.40	Continuous Monitoring	All welds

(1) ASME Code, Item No. C1.50 is not applicable to CRBRP.
(2) Different sample each interval to cover essentially 100% of boundary at end of 3rd interval.

TABLE 5-2

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY C-B		WELDS IN GUARD TANKS AND GUARD PIPES	CODE CLASS 2
ITEM NO.[2]	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
C2.11	Guard Tanks on Primary Liquid Metal Boundary	Visual, VTM-3[3]	25% of all welds[1]
C2.21	Guard Pipes on Primary Liquid Metal Boundary	Visual, VTM-3[3]	25% of all welds[1]

(1) Different sample each interval.
(2) ASME Code Nos. C2.10, C2.20, and C2.30 are not applicable to CRBRP.
(3) Performed in conjunction with examination of associated vessels and piping.

TABLE 5-2

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY C-C and C-E		INTEGRAL ATTACHMENTS	CODE CLASS 2
ITEM NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Integrally Welded Attachments to:		
C3.10	Liquid Metal Containing Vessels	Visual, VTM-3	Essentially 100% of each weld
C3.20	Guard Tanks	Visual, VTM-3	Essentially 100% of each weld
C3.30	Pump Tanks	Visual, VTM-3	Essentially 100% of each weld
C3.40	Steam Generator Shells	Visual, VTM-3	Essentially 100% of each weld
C3.50	Other Heat Exchanger Shells	Visual, VTM-3	Essentially 100% of each weld
C3.60	Piping	Visual, VTM-3	Essentially 100% of each weld
C3.70	Valves	Visual, VTM-3	Essentially 100% of each weld

TABLE 5-2

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY C-D		BOLTING	CODE CLASS 2
ITEM NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Cover Gas Space Closures		
C4.10 (2)	Primary Coolant Systems Bolting and Auxiliary Sodium System Bolting (Radioactive)	Continuous Monitoring	Adjacent spaces communicating with bolted closures
C4.11	Intermediate Coolant Systems Bolting	Visual, VTM-3	Essentially 100% of bolted closures
C4.12	Auxiliary Liquid Metal Systems (Non-radioactive)	Visual, VTM-3	33% of bolted closures (1)

(1) Different sample each interval to cover essentially 100% by end of 3rd interval.
(2) The requirement for examination of radioactive Auxiliary Sodium System Bolting is not defined in ASME Section XI Division 3.

TABLE 5-2

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY C-F		DISSIMILAR METAL WELDS	CODE CLASS 2
ITEM NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Liquid Metal Retaining Piping		
C5.10	Circumferential Welds	Volumetric	33% of all welds (1)
C5.11	Same as C5.10	Continuous Monitoring(2)	100% of weld
	Cover Gas Retaining Piping		
C5.20	Any Dissimilar Metal Welds	Visual, VTM-3	33% of all welds (1)
	Liquid Metal Retaining Vessels		
C5.30	Circumferential Shell Weld on EVST	Visual, VTM-2(3)	100% of weld (3)
C5.31	Same as C5.30	Continuous Monitoring(2)	100% of weld

(1) Different sample each interval to cover essentially 100% by end of 3rd interval.
(2) As described in IMA-2242 for gas leakage, as described in IMA-2241 for sodium leakage.
(3) See Exception A.2 in Addenda A.

TABLE 5-2

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY C-G		LIQUID METAL PIPING AND VALVES	CODE CLASS 2
ITEM[1] NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR EACH INSPECTION INTERVAL
	Heat Transport Loop Piping		
C6.10	Liquid Metal Boundary on Intermediate System Piping and Valves	Continuous Monitoring[1]	100%
	Primary Coolant Auxiliary Systems		
C6.20	Liquid Metal Boundary on Auxiliary System Piping and Valves	Continuous Monitoring[1]	100%

(1) As described in IMA-2241.

TABLE 5-2

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY C-H		STEAM GENERATOR TUBING		CODE CLASS 2
ITEM NO.	PARTS EXAMINED	EXAMINATION METHOD	EXTENT AND FREQUENCY OF EXAMINATION	
			1st INSPECTION INTERVAL	SUCCESSIVE INSPECTION INTERVALS, 2nd, 3rd, 4th
C7.10	Evaporators and Superheaters	Volumetric	3% heating surface in each evaporator/ superheater (1)	1-1/2% heat transfer surface in each of the three evaporator/ superheaters contained in one loop (2)
C7.11	Same as C7.10	Continuous monitoring for water/steam to sodium leakage	Continuous	Continuous

(1) The heat transfer surface is specified in terms of the number of tubes to be examined over their entire length.
(2) All tubes to be examined are selected from previously examined tubes, including tubes which exhibited wall thinning rate greater than the design allowable.

TABLE 5-3

CRBRP SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY D	CLASS 3 COMPONENTS	
ITEM NO.	COMPONENTS SUBJECT TO TEST AND EXAMINATION	TYPE AND FREQUENCY OF EXAMINATION
D1.10	Liquid metal-retaining components within the boundary of Class 3 Systems	A visual examination (VTM-2) shall be performed under the conditions of a system leakage test(1)
D1.20	Integral Attachments (3)	Visual examination (VTM-3) at or near the end of each inspection interval
D1.X (4)	Liquid metal retaining components not normally filled with liquid metal, but maintained with an inert gas atmosphere, within the boundary of Class 3 systems	Continuously monitor for gas leakage(2)
D1.XX (4)	Same as D1.X	Leakage test (1)

(1) In accordance with IMD-5220.
(2) May be pressure monitoring or flow monitoring.
(3) Components exempt per IMD-1220.
(4) The requirement for this examination is not defined in ASME Section XI Division 3.

6.0 Inservice Pump and Valve Tests

ASME Code Section XI, Division 1 subsections IWP and IWV provide the rules and requirements for inservice testing of Class 1, 2 and 3 water/steam systems pumps and valves. Since the requirements for sodium and cover gas systems have not been included in Section XI Division 3, the intent of the requirements for Division 1 will be used at the present time for CRBRP.

The purpose of these tests is to obtain information through measurement or observation of the operational readiness of a pump or valve. The scope of performance testing extends to those pumps and valves required to perform a safety function either to shut down the reactor or mitigate the consequences of an accident. Evaluation of the test data is performed on a pump or valve to indicate its change in hydraulic or mechanical characteristics. If the change is within limits, as set by Section XI Division 1 of the ASME Code, then that pump or valve is returned to service. If the change is not within limits, then either a more frequent test schedule is established to monitor the component or a repair or replacement is implemented.

6.1 Inservice Pump Testing

This section addresses inservice testing of Class 1, 2 and 3 centrifugal or displacement type pumps that are required to perform a specific function in shutting down a reactor or in mitigating the consequences of an accident and are provided with an emergency power source to the pump driver.

Sodium and cover gas system pump examinations shall meet the intent of ASME Code Section XI, Division 1, Subsection IWP.

These examinations, when defined, will obtain information through measurement or observation of the pump. These specific examination requirements are being evaluated and will be provided when this evaluation is completed.

6.2 Inservice Valve Testing

This subsection concerns itself with inservice testing of Class 1, 2, and 3 valves including their activating and position indicating systems which are required to perform a specific function in shutting down the reactor to the refueling condition or in mitigating the consequences of an accident.

Sodium and cover gas system valve examinations shall meet the intent of ASME Code Section XI, Division 1, Subsection IMV.

These examinations, when defined, will provide information through measurement or observation of the valve. These specific examination requirements are being evaluated and will be provided when this evaluation is completed.

6.3 Additional Testing Information

- A test program for pre-service and in-service testing of safety-related pumps will be supplied by the FSAR. The program will include test frequencies, acceptance criteria, pump categorization and test requirements.

- A test program for pre-service and in-service testing of safety-related valves will be supplied by the FSAR. This will include valve categorization, test frequencies, acceptance criteria, and test requirements.

- All safety-related systems will be reviewed to assure that equipment and systems will be constructed to accommodate in-service testing to the above requirements. Completion of this review will be reported by the FSAR.

7.0 Component Supports and Snubbers

Inservice inspection requirements for ASME Code Class 1, 2 and 3 supports and snubbers shall be in accordance with the intent of ASME Section XI Division 3 for sodium/cover gas components as shown in the following Tables 7-1, 7-2 and 7-3.

Justification for relief from specific ASME Code requirements are contained in Addenda A of this document.

Tables 7-1, 7-2 and 7-3 identify the examination requirements from ASME Section X, Division 3 in all instances. In those instances where CRBPR requirements are different from the code, these differences are identified and the justification for the variance is referenced. In those instances where CRBRP has specified requirements not contained in the code, they are identified as such in the Tables.

TABLE 7-1

CRBRP COMPONENT SUPPORTS FOR SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY F-A — PLATE, SHELL, AND LINEAR TYPE SUPPORTS

ITEM NO.	PARTS EXAMINED(4)	EXAMINATION METHOD(3)	EXTENT OF EXAMINATION FOR PARTS EXAMINED(2)
F1.10	Mechanical connections to liquid metal and cover gas retaining components and building structures	Visual, VTM-3	CODE CLASS 1 Essentially 100% of the supports on Class 1 components and piping shall be examined each inspection interval. CODE CLASS 2 Essentially 100% of the supports on Class 2 components and piping in a composite loop shall be examined during plant life with at least 33-1/3% to be examined during the first inspection interval.
F1.11	Weld Connection to building structure	Visual, VTM-3	CODE CLASS 3(1) Essentially 100% of the supports on Class 3 components and piping in a composite loop shall be examined during plant life, with at least 33-1/3% to be examined during the first inspection interval.
F1.12	Weld and mechanical connections at intermediate joints in a multi-connected integral and non-integral support	Visual, VTM-3	
F1.13	Component displacement settings of guides and stops; misalignment of supports; assembly of support items.	Visual, VTM-3	

(1) Supports on Class 3 components and piping which are four inches nominal pipe size and smaller are exempt from examination and testing (IMD-1220).
(2) Support examination boundaries shall be per IMF-1300.
(3) Acceptance Standards shall be per IMF-3410, 3420 and 3430 as applicable.
(4) Component supports selected for examination shall be in accordance with IMF-2510.

TABLE 7-2

CRBRP COMPONENT SUPPORTS FOR SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY F-B		LINEAR TYPE SUPPORTS	
ITEM NO.	PARTS EXAMINED(4)	EXAMINATION METHOD(3)	EXTENT OF EXAMINATION FOR PARTS EXAMINED(3)
F2.10	Mechanical connections to liquid metal and cover gas retaining components and building structures	Visual, VTM-3	CODE CLASS 1 Essentially 100% of the supports on Class 1 components and piping shall be examined each inspection interval. CODE CLASS 2 Essentially 100% of the supports on Class 2 components and piping in a composite loop shall be examined during plant life with at least 33-1/3% to be examined during the first inspection interval.
F2.11	Weld Connection to building structure	Visual, VTM-3	CODE CLASS 3(1) Essentially 100% of the supports on Class 3 components and piping in a composite loop shall be examined during plant life, with at least 33-1/3% to be examined during the first inspection interval.
F2.12	Weld and mechanical connections at intermediate joints in a multi-connected integral and non-integral support	Visual, VTM-3	
F2.13	Component displacement settings of guides and stops; misalignment of supports; assembly of support items.	Visual, VTM-3	

(1) Supports on Class 3 components and piping which are four inches nominal pipe size and smaller are exempt from examination and testing (IMD-1220).
(2) Support examination boundaries shall be per IMF-1300.
(3) Acceptance Standards shall be per IMF-3410, 3420 and 3430 as applicable.
(4) Component supports selected for examination shall be in accordance with IMF-2510.

TABLE 7-3

CRBRP COMPONENT SUPPORTS FOR SODIUM AND COVER GAS SYSTEMS

CODE CATEGORY F-C		COMPONENT STANDARD SUPPORTS	
ITEM NO.	PARTS EXAMINED(4)	EXAMINATION METHOD	EXTENT OF EXAMINATION FOR PARTS EXAMINED(3)
F3.10	Mechanical connections to liquid metal and cover gas retaining components and building structures	Visual, VTM-3	CODE CLASS 1 Essentially 100% of the supports on Class 1 components and piping shall be examined each inspection interval.
F3.11	Weld connection to building structure	Visual, VTM-3	
F3.12	Weld and mechanical connections at intermediate joints in a multi-connected integral and non-integral support	Visual, VTM-3	CODE CLASS 2 Essentially 100% of the supports on Class 2 components and piping in a composite loop shall be examined during plant life, with at least 33-1/3% to be examined during the first inspection interval.
F3.13	Component displacement settings of guides and stops; misalignment of supports; assembly of support items	Visual, VTM-3	
F3.14	Spring-type supports, constant load-type supports, snubbers, and shock absorbers	Visual, VTM-4(2)	CODE CLASS 3 (1) Essentially 100% of the supports on components and piping in a composite loop shall be examined during plant life, with at least 33-1/3% to be examined during the first inspection interval.

(1) Supports or snubbers on Class 3 components and piping which are four inches nominal pipe size and smaller are exempt from examination and testing (IMD-1220).
(2) This examination shall consist of a direct or remote visual examination at different system temperatures to determine functional adequacy, correct settings, and required freedom of motion of component supports. This examination may require disassembly and/or operability testing (IMF-5000).
(3) Support examination boundaries shall be per IMF-1300.
(4) Component supports selected for examination shall be in accordance with IMF-2510.

8.0 Preservice Inspection Requirements

Preservice examinations shall be performed on CRBRP components and systems upon completion of the applicable ASME construction code requirements. Detailed requirements for preservice examinations to be per-formed prior to fuel load shall be specified in the appropriate component purchase specifications, system construction test requirements, preoperational test specifications or startup test specifications. Principal features of the CRBRP program are as follows:

All visual, surface, volumetric weld examinations, and testing identified in this document shall be performed completely once as a preservice examination.

All welds requiring continuous monitoring shall be visually examined during the preservice system leakage tests (IMA-5200).

All liquid metal coolant boundary welds and supports intended for inservice inspection at 400°F shall receive a preservice inspection at 400° F sodium temperature to verify the equipment, techniques, and access that are to be utilized in the inservice inspection program.

Dimensional verification shall be performed on ASME Section III component and piping positions (except Guard Vessels) when the plant is:

(1) cold empty (ambient)
(2) preheated empty (400°F)
(3) preheated and filled with liquid metal (400°F)
(4) during hot functional testing (700°F-800°F).

9.0 Inspection Intervals

The CRBRP Inservice Inspection Program is applicable for the 30 year life of the plant. Each inspection interval is 10 years for a total of 3 inspection intervals. Each inspection interval consists of 3 nominal 40 month periods. The inservice examinations and system pressure tests required by IWC and IWD for water/steam systems, and IMB, IMC and IMD for sodium/cover gas systems shall be completed during each of the inspection intervals for the service lifetime of the plant. The inspections shall be performed in accordance with the schedule of Inspection Program B of IMA2420.

Inspection Interval	Inspection Period Calendar Years of Plant Service	Minimum Examinations % Completed	Maximum Examinations % Completed
1st	3	16	34
1st	7	50	67
1st	10	100	100
2nd	13	16	34
2nd	17	50	67
2nd	20	100	100
3rd	23	16	34
3rd	27	50	67
3rd	30	100	---

The inspection interval shall be determined by calendar years following placement of the plant into commercial service.(1) Each inspection interval may be decreased or extended (but not cumulatively) by as much as 1 year. If the plant is out of service continuously for 6 months or more, the inspection interval during which the outage occurred may be extended for a period equivalent to the outage.

The inspection intervals for inservice testing of pumps and valves shall be in accordance with the requirements of IWP and IWV, which will be expanded to include the sodium and cover gas systems.

The inspection intervals for examination and testing of component supports shall be in accordance with the requirements of IWF and IMF.

The inspection intervals for component replacements, additions, and alterations that may be required during the service lifetime of the plant shall coincide with remaining intervals, as determined by the calendar years of plant service at the time of replacements, additions, or alterations.

- - - - - -

(1) The date of placement of the plant into commercial service is defined by the regulation of the Federal Power Commission, Chapter 1-Title 18 Code of Federal Regulations 101 Paragraph 9.D.

10.0 Examination Methods

The four types of examination methods used for CRBRP inservice inspections are defined in ASME Section XI, Division 1 and 3 as visual, surface, volumetric, and continuous monitoring. This section of the document identifies these methods.

Specific Acceptance Standards are in the course of preparation, per ASME Section XI, Division 3, and are applicable to the plant operator for normal plant operation.

(1) Visual Examination, VTM-2

Shall be conducted on exterior surfaces in such a way that accumulations of liquids, liquid streams, liquid drops and smoke are discernable - may be direct or remote - using aids. The removal of exterior covering, such as insulation, is not required for VTM-2 visual examination.

(2) Visual Examination, VTM-3

Shall be conducted in a manner and with equipment that will determine the general mechanical and structural conditions of items. The presence of loose parts, debris, and loss of integrity at bolted or welded connections to components shall be detectable. Such examination may be conducted by remote devices such as remote optical and lighting equipment, undersodium scanning, or remote dimensional gaging.

(3) Visual Examination, VTM-4

Shall be conducted in a manner and with equipment that will determine the condition with respect to operability of components or devices which have mechanical functions during plant operation. These may include snubbers, component supports, and pipe hangers. Shall consist of a direct or remote visual examination to determine functional adequacy, correct settings, and required freedom of motion of component supports.

(4) Surface Examination

Shall be used to delineate or verify the presence of surface or near-surface cracks or discontinuities. It may be conducted by either a magnetic particle or a liquid penetrant method where the surface condition, material, and accessibility permit such an examination.

(5) Volumetric Examination

Used to indicate the presence of discontinuities throughout the volume of material and may be conducted from either the inside or outside surface of a component.

(6) Continuous Monitoring

Continuous monitoring of the region exterior to components or systems containing liquid metal and/or radioactive cover gas to provide for detection of leakage of liquid metal or radioactive gas.

10.1 Continuous Monitoring-Class 1 Components Requirements For Sodium Leak Detection (IMB-5110-1)

Class 1 liquid metal systems shall have diverse, redundant leak detection devices installed to monitor each component and liquid metal appurtenance of the system to monitor for liquid metal to gas leakage. During periods of extended reactor shutdown, only one redundant continuous monitoring technique shall be required.

NOTE: Diverse redundant shall be interpreted to mean more than one device with at least one of the multiple devices utilizing a different detection technique than is used in the primary device.

10.2 Continuous Monitoring-Class 2 and 3 Component Requirements for Sodium Leak Detection (IMC-5110-1, IMD-5110-1)

Class 2 and 3 liquid metal systems shall have detection devices installed on, under, or near each component and liquid metal appurtenance of the system to monitor for sodium to gas leakage.

10.3 Radioactive Cover Gas Leakage Monitoring - Class 1, 2 and 3

ASME Code Class 1, 2 and 3 cover gas systems shall have radiation monitors installed to monitor for leakage from the radioactive gas spaces. Such radiation monitors need not be dedicated to leak detection but may serve additional functions.

10.4 Sodium to Sodium Leak Detection

To detect leakage of primary sodium into intermediate sodium systems, radiation detectors shall be located to monitor continuously the radioactive content of the IHTS.

ADDENDA A

JUSTIFICATION FOR ASME CODE EXCEPTIONS

INTRODUCTION

This Addenda addresses on a case-by-case basis the CRBRP exceptions to the ASME Code Requirements and provides the justification for each exception. Some exceptions to the Code are for a specific component and some are for generic exceptions.

ADDENDA A

TABLE OF CONTENTS

SECTION		PAGE
	Introduction	A-I
	Table of Contents	A-II
A.1	Justification for Visual Inspection of Reactor Vessel Dissimilar Metal Welds	A1-1
A.2	Justification for Visual Inspection of EVST Dissimilar Metal Welds	A2-1
A.3	Justification for Visual Inspection of CLCV Snubber Attachments	A3-1

A.1 JUSTIFICATION FOR VISUAL INSPECTION OF REACTOR VESSEL DISSIMILAR METAL WELDS

ASME Section XI, Division 3 specifies volumetric examination of Class 1 dissimilar metal welds, whereas this report specifies visual examination of the dissimilar metal weld in the reactor vessel. This weld is in the upper section of the vessel above the coolant level. It is the only dissimilar metal weld in the primary system. The basis for this exception follows:

The reactor vessel consists principally of an austenitic stainless steel shell which is top end supported from a carbon steel flange. A transition shell of Inconel 600 is utilized between the carbon steel flange and stainless steel shell. Hence, there are two dissimilar metal welds in the reactor vessel primary boundary, one for joining the stainless steel to Inconel and one joining the Inconel to the carbon steel flange. However, since the metallurgical structure of Inconel 600 and Type 304 stainless steel are similar, the Inconel to stainless steel weld is not classed as a dissimilar metal weld in the classical sense, but the SA 508 carbon steel to Inconel 600 weld is recognized as a dissimilar metal weld.

Visual inspection of stainless steel welds has been established as a necessary and sufficient inservice inspection for adequate protection of public health and safety. The Inconel to stainless steel weld is also assigned to this class. Volumetric inspection has been considered in the ASME Code Section XI, Division 3, as being necessary for inservice inspection for dissimilar metal welds. This requirement is not necessary for inspection of the reactor vessel carbon steel flange to Inconel shell weld on the basis of the moderate operating environment and the consequences of a weld crack.

The service environment and temperature will not result in material degradation effects in the dissimilar metal weld. The transition region of the vessel is located in a position which has a total fluence of less than 10^{15} n/cm^2 over the life of the plant. This low fluence level is considered to be below the threshold level for mechanical property degradation of the materials involved. This service temperature for the carbon steel to Inconel weld is approximately 465°F. At this operating temperature, the base metals involved together with the Inconel 82 weld filler metal are metallurgically stable.

Both the internal and external environment are considered benign with regard to degradation of the material in the transition region. The internal environment will be argon gas and sodium vapor and essentially no sodium mass transfer or interstitial transfer effects occur at temperatures below 800°F, especially when the sodium is present as a vapor or thin condensed layer. The external environment is reactor cavity gas, nitrogen with an oxygen content of less than 2% and the interaction of the materials involved with this reactor cavity gas are negligible at the low service temperature.

Selection of the materials for this transition joint was made based on the above considerations coupled with the requirement to minimize thermal expansion differences which could cause high stresses to be built up during thermal cycling. In addition, the use of the nickel base alloy filler metal, Inconel 82, minimizes the depletion of carbon from the fusion zone of the carbon steel during welding and subsequent high temperature stress relief.

The maximum primary membrane stress intensity in this region due to dead weight and design pressure is relatively low, about 1400 psi, in comparison to the allowable stress level of 17,400 psi. Similarly, the maximum primary membrane plus bending stress intensity due to dead weight and design pressure at the dissimilar metal weld is only about 1700 psi, in comparison to the allowable value of 26,100 psi. For a crack in this weld to be significant relative to public health and safety, it would have to extend around a large portion of the vessel's 65 foot circumference. The leakage of sodium from any crack through the weld thickness would be detected by the aerosol type leak detectors in the reactor vessel/guard vessel annulus. Periodic visual inspection would confirm and locate any significant cracks allowing such leakage.

Effects on the reactor vessel bimetallic weld from seismic events can be considered to be two types:

1) A catastrophic failure of the weld resulting in complete separation of the vessel cylinder from the support assembly, and

2) fatigue type growth of a crack to such an extent that a large distortion of the vessel would occur during a seismic event.

The first effect, catastrophic failure, has been addressed by Babcock & Wilcox, the vessel vendor, through performance of an ASME Code, Section III, subsection NA, Appendix G brittle fracture analysis. The analysis was based on considering a postulated 1 inch defect experiencing SSE loading conditions at 70°F, which is conservative since fracture resistance increases with temperature. Although Appendix G is based on using minimum fracture resistance properties for SA 533 and SA 508 steels, and therefore does not consider any degradation of critical stress intensity values due to the welding, the analysis is considered conservative because of the temperaure chosen for performance of the analysis. The value for K_{IR} at a temperature of 210°F is almost twice the value used in the analysis. Fracture resistance at 400 F to 600°F will be higher than the value at 210°F which represents the maximum temperature for which values are given.

Consideration of the possibility of fatigue type growth is significantly more difficult since fatigue crack growth properties for SA 508 in a bimetallic Heat Affected Zone (HAZ) are not available. However, for stress levels of about 12 ksi, a stress intensity of 21.3 ksi would result for a 1 inch long through thickness crack.

Considering this as the alternating stress intensity value during seismic loading would result in a crack growth of about 10^{-6} inch/cycle for HAZ material from a SA 533-A-2 submerged arc weldment. For a 4 inch long crack the growth would be less than 10^{-5} inch/cycle. Growth rates in the SA 508 HAZ would be expected to be of the same order of magnitude (testing programs have confirmed that Fatigue Crack Growth Rates (CGR) in a 2-1/4 CR-1 Mo bimetallic weld HAZ are lower than in the base metal); however, even a growth rate two orders of magnitude larger would be inconsequential.

Finally, it should also be noted that the RV bimetallic weld joint is considered less subsceptible to crack initiation than those typically found in fossil fired plants. This is because an Inconel transition piece is used to join the stainless shell to the low alloy frange, and a nickel based weld filler material was used. Both of these contribute to reducing the thermal stresses present in the ferritic HAZ. In addition, cracking of the ferritic material is generally associated with the formation of an "oxide notch", a phenomenon which will be retarded by the low oxygen operating environment.

Based on the foregoing discussion, circumferential cracking of any significant extent is not expected and is considered extremely unlikely. Additional supporting information is contained in W-ARD Engineering Study Report ES-RV-81-001 "Bi-Metallic Weld Study" dated February 12, 1981, which is provided as a supplement to this section.

Hence, on the basis of the benign operating environment experienced by the dissimilar metal weld, stresses of less than 10% of the allowable, and the detectable nature of any significant cracks in that weld, visual inservice inspection in lieu of volumetric inspection is considered adequate to assure protection of public health and safety.

WESTINGHOUSE ADVANCED REACTORS DIVISION

CRBRP ENGINEERING STUDY REPORT

Number: ES-RV-81-001
Date: February 12, 1981

Title: Bi-Metallic Weld Study

Prepared by: F. J. Baloh 2/13/81

Approved by: W. E. Pennell 2/13/81

SUMMARY

A literature review was performed to evaluate the performance of bi-metallic welds and to classify the causes of observed failures. The factors which contribute to dissimilar metal weld failure were found to fall into four basic categories: 1) cyclic thermal stress, 2) oxidation at the weld interface, 3) carbon migration away from the interface, and 4) metallurgical deterioration at elevated temperatures. Of these four factors, only one, high temperature operation, was a common factor in all observed failures. Each of these factors was then reviewed relative to the CRBRP reactor vessel bi-metallic weld and shown to have effectively no potential to initiate failure. For this reason it is believed that volumetric inservice inspection of this weld is unnecessary and would do little to improve assurance of vessel integrity.

TABLE OF CONTENTS

PAGE

Introduction----------A1-7

Cyclic Thermal Stress----------A1-8

Low Oxidation Resistance of the Ferritic Steel----------A1-8

Carbon Migration----------A1-8

Metallurgical Deterioration Caused by
Elevated Temperature Exposure----------A1-8

Initial Weld Integrity----------A1-9

PWR Experience----------A1-9

Conclusion----------A1-9

Figure 1 - CRBRP Reactor Vessel----------A1-9a

References----------A1-10

Attachment A - Bi-Metallic Weld Failure, Reference Review-----A1-12

INTRODUCTION

The bulk of the CRBRP reactor vessel is fabricated from two materials; SA 508 ferritic steel and 304 austenitic stainless steel. The A 508 is used in the low temperature upper portion of the vessel (See Figure 1) making its thermal expansion compatible with that of the closure head which is fabricated of the same material. The joining of the two vessel materials is accomplished by incorporating an Inconel 600 cylindical transition piece of 14" height by 3-1/8" thickness. The weld joint at the top and bottom of the transition piece is a double "J" prep with the weld metal being Inconel 82. The primary function of the transition piece is to minimize stresses due to differential thermal expansion btween the SA 508 and the stainless steel portions of the vessel. The transition cyclinder together with its circumferential welds is located above the maximum vessel sodium level. In this position the welds cannot experience the indeterminate thermal striping which has been associated with inner surface cracking of pressure boundary welds. Transient stresses in the bi-metallic weld are limited to those produced by the well characterized differential expansion stresses associated with the reactor duty cycle events. These stresses are within the allowable code limits. "Block loading" fatigue tests have shown that the safety factors incorporated into the fatigue design curves are such as to preclude crack initiation.

The purpose of this memorandum is to present the results of a review of industrial experience with bi-metallic welds and to relate that experience to the bi-metallic welds of the reactor vessel.

The vast majority of failure experience on bi-metallic welds have been derived from two sources; 1) commercial fossil-fired power plants, and 2) tests performed specifically to provide a better understanding of the failure mechanisms. Experience on bi-metallic joints in PWR reactors has shown no failures. The factors known to contribute to dissimilar-metal weld failure have been thoroughly documented[1-5]. Tucker and Eberle summarized them as: 1) cyclic thermal stresses, 2) low oxidation resistance of the low-alloy ferritic steel, 3) carbon migration, and 4) metallurgical deterioration caused by elevated-temperature exposure.

Attachment A summarizes the results of a review of several references on bi-metallic weld failures and their causes. The essence of the review results is to substantiate that the factors which are always found to contribute to failure are those identified above and that no failures have been identified wherein at least one of those factors has not contributed to the failure. Further, the one common element associated with all identified failures is extended operation at high temperature (>800°F). There was a general consensus among all references concerning the benefits of using a nickel base weld metal to minimize the potential for carbon migration.

To evaluate the potential for failure of the CRBRP reactor vessel bi-metallic weld, each of the factors known to contribute to failure will be reviewed for its relevance to this weld.

CYCLIC THERMAL STRESS

The vessel bi-metallic weld is located in a relatively benign thermal environment. It is well above the sodium level and hence is not exposed to rapid fluctuations in temperature, as an example, the RV-9N transient requires 28 minutes to return to normal operating temperatures. The normal operating temperature at the ferritic steel joint is 450°F. Temperature gradients thru the wall during the RV-9N transient do not exceed 15°F and longitudinal gradients during the same transient in the 14" above the weld are a maximum of 65°F. The maximum stress range due to thermal gradients is 32 ksi.

OXIDATION RESISTANCE OF THE FERRITIC STEEL

Oxidation of the carbon steel region of the dissimilar metal weld is virtually undetectable at temperatures below 800°F (References 11 and 12). Reaction of the weld metals exposed to nitrogen in the reactor cavity gas will not occur at temperatures below 800°F (Reference 13).

CARBON MIGRATION

The use of nickel base alloy filler metal, Inconel 82, for the dissimilar metal joint minimizes the depletion of carbon from the fusion zone of the carbon steel both during welding and subsequent stress relief. This material is the recommended filler metal for ferritic to austenitic dissimilar welds the rationale for which is described in Attachment A. Carbon migration from the ferritic steel during service is essentially unmeasurable at the temperatures in question. Values of diffusion coefficients are so low that references providing experimetally determined magnitudes are not available. Extrapolations of the diffusion coefficient from higher temperature data would put it in the range of 10^{-16} - 10^{-17} cm^2/sec. (Reference 14). Carbon migration is therefore not believed to be a potential failure causing factor in the vessel bi-metallic weld.

METALLURGICAL DETERIORATION CAUSED BY ELEVATED TEMPERATURE EXPOSURE

As noted previously, the vessel bi-metallic weld will not be exposed to temperatures at which metallurgical instability could exist. Reference material examined does not even address questions of stability until temperaturs at least in the 700-800°F range have been reached. Discussions with materials experts support the contention that in the thermal environment of the bi-metallic weld material stability is non-controversial.

INITIAL WELD INTEGRITY

Another factor to consider in the reliability of the vessel bi--metallic weld in the quality of the initial weld. As noted in one of the references discussed in Attachment A, slag inclusion when using nickel base weld metals is a more likely occurrence than for other weld filler materials. This concern was recognized in established the NDT procedures to confirm the quality of the weld. Rather than using narrow spectrum isotopic sources which under some circumstances have failed to identify inclusions of this type wide spectrum x-ray equipment was employed. Additionally exposures were made from multiple angles to give high confidence that all inclusions were identified and repaired thus effectively eliminating initial flaws as a potential failure mechanism.

PWR EXPERIENCE

To gain additional support for the conclusions reached concerning the high reliability of the vessel bi-metallic weld, discussions were held with personnel from the W PWR division to evaluate their experience with similar welds. The weld joint most nearly like the vessel weld is the primary piping "safe end weld" on as four loop PWR. This joint connects the SA 508 vessel to the Type 304SS outlet piping utilizing Inconel 82 filler metal. Operating temperatures are 560°F with pressure, pipe expansion, seismic and thermal transient loading. In service U.T. has been performed on approximately 50 of these welds. None of these welds, some of which have been in service for more than 15 years, have generated detectable cracks which have required corrective action.

CONCLUSION

Volumetric inservice inspection of this weld, because of the well-characterized low-temperature service conditions, is unnecessary and would do little to improve assurance of the vessel integrity.

REFERENCES

1. R. V. Blaser, F. Eberle, and J. T. Tucker, "Welds Between Dissimilar Alloys and Full Size Steam Piping," Am. Soc. Test. Mater. Proc. 50: 789-807 (1950).

2. G.E. Lien, F. Eberle, and R. D. Wylie, "Results of Service Test Program on Transition Welds Betwen Austenitic and Ferritic Steels at the Phillip Sporn and Twin Branch Plants," Trans. ASME 76: 1075-83 (1954).

3. R. W. Emerson, R. W. Jackson, Ca. A. Dauber, "Transition Joints Between Austenitic and Ferritic Steel Piping for High Temperature Steam Service," Weld. J. (N.Y.) 41(9): 385-s-93-s (1962).

4. J. T. Tucker and F. Eberle, "Development of a Ferritic-Austenitic Weld Joint for Steam Plant Applications," Weld J. (N.Y.) 35(11): 529-s-40 (1956).

5. J. Barford and K. S. Probert, "Interfacial Effects in Dissimilar Steel Joints," paper presented at the International Conference on Welding Research Related to Power Plants, Marchwood Engineering Laboratories, Sept. 17-21, 1972.

6. "Dissimilar-Metal-Weld Failure Problems in Large Steam Generators," Schaefer, Adolph; Power Magazine, December 1979.

7. "Behavior and Properties of Welded Transition Joints Between Austenitic Stainless Steels and Ferritic Steels - A Literature Review," by J. F. King; ORNL-TM-5163, November 1975.

8. "Interfacial Structures in Nickle-Based Transition Joints After Long-Term Service," R. D. Nicholson; Central Electricity Generating Board RD/M/N1131, September 1980.

9. "Austenitic/Ferritic Steampipe Transition Joints in Midlands Region Plant; Service Experience to December 1977," T. R. Rowberry and B. I. Bagnall; Central Electricity Generating Board SSD/MID/R66/78, November 1978.

10. "Ferritic-Austenitic Transition Joint Failures," R. L. Kluch and J. F. King; ORNL-5637, April 1980.

11. Metal Handbook, Volume 1, Properties and Selection of Metals, 8th Edition, 1961, page 265.

12. R. C. King, Piping Handbook, 5th Edition, McGraw-Hill Book Co. 1967, pages 2-92 to 2-93.

13. Metals Handbook, Vol. 1, Properties and Selection of Metals, 8th Edition,, 1961, page 580.

14. "Na Tech Program Component Materials Compatibility," Quarterly Prog. Report for Period Ending 4/30/74, Shiels, S.; WARD NA3045-11.

ATTACHMENT A

BI-METALLIC WELD FAILURE REFERENCE REVIEW

Reference 6 catalogs the results of experience gained at 54 utilities on superheater/reheater bi-metallic weld joints. Of the 54 utilities responding, 20 had experienced dissimilar-metal weld failures. None of the failures were attributable to poor welding techniques. They involved 60 units out of a total of 320 for which data was reported. Of the sixty units experiencing failure, seven had used nickel base filler metal, the remainder used ferritic or austenitic stainless steel filler metal. It was determined that carbon migration (diffusion of carbon from ferritic steel at the weld interface to the dissimilar weld metal) was a major cause of failure. Ths was much more pronounced in stainless steel weld metal and found to become significant in nickel base weld metals only after many years of service at elevated temperature (>800°F). Many of the failures did not occur until after extended high temperature operation (>15 years). The reduction of carbon caused by migration lowers the ferritic materials creep strength in a narrow zone immediately adjacent to the weld interface. Because most cracking was found to occur in this carbon depleted zone, carbon migration was found to be an important contributing factor. However, failures were also found to occur where carbon migration was prevented by metallurgical techniques and some failures were observed away from the carbon depleted zone. Externally applied loads were also found to be important, since several failures were found in tubes which showed evidence of large bending strain. It is impoetant to note that all of these failures occurred at temperatures (1000 F) well above those to be imposed on the vessel bi-metallic weld.

Reference 7 provides the results of an extensive review of dissimilar-metal weld failures from the 1950's through to 1975. It concluded that the use of nickel based fillers was very effective in reducing failures because of its ability to reduce carbon migration. They did however express concern that even when nickel based weld filler is used in the presence of an oxidizing environment, a mechanical notch will form at the interface between weld metal and base metal due to differences in oxidation resistance. This notch can lead to premature failure. An additional disadvantage identified for nickel base weld metal is its high sensitivity to notches such as from small slag inclusions along the fusion line. Test pieces with small slag inclusions at the interface failed at elongations of 10% of those imposed on inclusion free samples.

References 8 and 9 provide data on bi-metallic weld cracking in thick section transition joints in Power Plants in England. The materials joined and weld filler material used varied over a wide range. However, in all cases one or more of the previously identified contributing conditions were present for all failed welds. It was concluded that it was not possible to quantify the effects of thermal cycling upon joint life and that the data suggest that time at temperature is the only reliable method of assessing expected joint life. This latter point on high temperature operation is extremely important. In all of the

investigations relating to bi-metallic weld failures the one element common to all failures is extended operation at high temperature. All other conditions such as carbon migration, oxidation, etc., are considered to be contributing factors and in all cases "active" factors only if the operating temperature is high (>800°F).

Reference 10 identified a new potential failure mechanism. Observations were combined with previous results to develop a mechanism by which the interface microstructure forms during welding, how it evolves during elevated temperature service, and how this microstructure leads to failure. The new feature of the proposed mechanism is the formation of a chromium depleted region that was demonstrated experimentally. The chromium depleted region is prone to preferential oxidation and eventually leads to notch formation, crack nucleation, and propagation. While the proposed mechanism may be a variation of those previously proposed, two already clearly identified factors were essential for failure; oxidation and high temperature.

A.2 JUSTIFICATION FOR VISUAL EXAMINATION OF EVST DISSIMILAR METAL WELD JOINT

ASME Section XI, Division 3 specifies volumetric examination of Class 2 dissimilar metal welds. Herein is provided justification for visual inspection (VTM-3) of the single dissimilar metal weld joint in the EVST storage vessel. This weld is in the upper section of the vessel above the coolant level. (See Figure A-2). The justification for this exception follows:

The EVST storage vessel consists principally of an austenitic stainless steel sheet, Type 304, which is top supported from a carbon steel flange forging (ASME-SA-508) similar to the reactor vessel. The forging bottom end weld preparation is overlayed with a 9/16-inch-thick butter overlay of Inconel 82 and then final heat-treated before welding to the Type 304 stainless shell. Hence, there are two dissimilar metal welds in the storage vessel pressure boundary, the butter overlay of Inconel 82 to the SA-508 forging and one joining the Inconel 82 to the stainless steel Type 304. However, since the metallurgical structure of Inconel 82 and Type 304 stainless are similar, this joint is not classed as a dis-similar metal weld in the classical sense, but the SA-508 steel to Inconel 600 weld overlay is recognized as a dissimilar metal weld joint.

Visual inspection of stainless steel welds has been established as a necessary and sufficient inservice inspection for adequate protection of public health and safety. The Inconel to stainless steel weld is assigned to this class. Volumetric inspection has been considered in the ASME Code Section XI as being necessary for inservice inspection for dissimilar metal welds (Class 2 vessels). This requirement is not necessary for inspection of the EVST storage vessel carbon steel flange to Inconel shell weld on the basis of the moderate operating environment and the unlikely consequence of a weld crack.

The service environment and temperature will not result in material degradation efects in the dissimilar metal weld. The transition region of the vessel is located in a position which has a total fluence of less than 2.5×10^{13} n/cm^2. This low fluence level is considered to be below the threshold level for mechanical property degradation of the materials involved. This service temperature for the carbon steel to Inconel weld is approximately 475°F. At this operating temperature, the base metals involved, together with the Inconel 82 weld filler metal, are metallurgically stable.

Both the internal and external environment are considered benign with regard to degradation of the material in the transition region. The internal environment will be argon gas and sodium vapor, and essentially no sodium mass transfer or interstitial transfer effects occur at temperatures below 700°F, especially when the sodium is present as a vapor or thin condensed layer. The external environment is EVST cavity gas, nitrogen with an oxygen content of less than 2%, and the interaction of the materials

involved with this EVST cavity gas are negligible at the low service temperature.

Selection of the materials for this transition joint was made based on the above considerations coupled with the requirement to minimize thermal expansion differences which could cause high stresses to be built up during thermal cycling. In addition, the use of the nickel-base alloy filler metal, Inconel 82, provides a ductile fusion joint.

The maximum primary membrane stress intensity in this region due to dead weight and design pressure is relatively low, about 1,000 psi, in comparison to the allowable stess level of 15,900 psi at 650°F. Similarly, the maximum primary membrane plus bending stress intensity due to dead weight and design pressure at the dissimilar metal weld is only 4,500 psi, in comparison to the allowable value of 23,850 psi. A third evaluation for the dissimilar metal weld joint was the combination primary membrane bending and thermal stresses with a safety factor of 150% (using the latest response spectra curves). For a crack in this weld to be significant relative to public health and safety, it would have to extend around a large portion of the storage vessel's 60-foot circum-ference. The leakage of sodium from any crack through the weld thickness would be detected by the aerosol-type leak detectors in the storage vessel/guard tank annulus. Periodic visual inspection would confirm and locate any significant cracks allowing such leakage.

Effects on the EVST bimetallic weld from seismic events can be considered to be of two types:

1) A catastrophic failure of the weld resulting in complete separation of the vessel cylinder from the support assembly, and

2) fatigue type growth of a crack to such an extent that a large distortion of the vessel would occur during a seismic event.

The first effect, catastrophic failure, has been addressed by Babcock & Wilcox, the vessel vendor, through performance of an ASME Code, Section III, Subsection NA, Appendix G brittle fracture analysis. The analysis was based on considering a postulated 1 inch defect experiencing SSE loading conditions at 70°F, which is conservative since fracture resistance increases with temperature. Although Appendix G is based on using minimum fracture resistance properties for SA 533 and SA 508 steels, and therefore does not consider any degradation of critical stress intensity values due to the welding, the analysis is considered conservative because of the temperaure chosen for performance of the analysis. The value for K_{IR} at a temperature of 210°F is almost twice the value used in the analysis. Fracture resistance at 400 F to 600°F will be higher than the value at 210°F which represents the maximum temperature for which values are given.

Consideration of the possibility of fatigue type growth is significantly more difficult since fatigue crack growth properties for SA 508 in a bimetallic Heat Affected Zone (HAZ) are not available. However, for stress levels of about 12 ksi, a stress intensity of 21.3 ksi would result for a 1 inch long through thickness crack.

Considering this as the alternating stress intensity value during seismic loading would result in a crack growth of about 10^{-6} inch/ cycle for HAZ material from a SA 533-A-2 submerged arc weldment. For a 4 inch long crack the growth would be less than 10^{-5} inch/ cycle. Growth rates in the SA 508 HAZ would be expected to be of the same order of magnitude (testing programs have confirmed that Fatigue Crack Growth Rates (CGR) in a 2-1/4 CR-1 Mo bimetallic weld HAZ are lower than in the base metal); however, even a growth rate two orders of magnitude larger would be inconsequential.

Finally, it should also be noted that the EVST bimetallic weld joint is considered less susceptible to crack initiation than those typically found in fossil fired plants. This is because an Inconel transition piece is used to join the stainless shell to the low alloy frange, and a nickel-based weld filler material was used. Both of these contribute to reducing the thermal stresses present in the ferritic HAZ. In addition, cracking of the ferritic material is generally associated with the formation of an "oxide notch", a phenomenon which will be retarded by the low oxygen operating environment.

Based on the foregoing discussion, circumferential cracking of any significant extent is not expected and is considered extremely unlikely.

Hence, on the basis of the benign operating environment experienced by the dissimilar metal weld, moderate stress combinations, and the detectable nature of any significant cracks in that weld, visual inservice inspection in lieu of volumetric inspection is considered adequate to assure protection of public health and safety.

A.3 JUSTIFICATION FOR VISUAL INSPECTION OF CLCV ATTACHMENTS

ASME Section XI, Division 3 requires either volumetric or surface inspection of integrally welded attachments to valves. In the case of the CLCV integrally welded snubber attachment, a visual inspection is proposed versus a volumetric or surface examination.

The rational for elimination of the volumetric or surface exami-nation requirement is that the attachment weld is subject only to low loads in normal service, the thermal expansion and contraction of the piping. The attachment welds are challenged only during seismic events. The mechanical joints of the attached snubbers are subject to a program of essentially 100% visual inspection. The snubbers themselves are subject to periodic performance testing to verify functional adequacy. The snubber attachment welds will be periodically visually inspected by removal of insulation to allow direct viewing of the welds. Such visual inspection can be per-formed at the same time the snubbers are tested.

Presently the technology does not exist for doing volumetric (ultrasonic) inspections for this type weld. Also, high temperature dye penetrants have not been developed to perform acceptably at CRBRP inspection temperatures of 400°F. The Sherwin Corporation has developed a dye penetrant which demonstrates acceptable performance at 250°F, but is marginal at 400°F + temperatures. Code acceptability of such high temperature methods has not been clearly defined. Because the volumetric and surface inspection techniques are not fully developed, a visual inspection is proposed for the CLCV integrally welded snubber attachments. Development of volumetric and/or surface inspection techniques will be appropriately monitored for application to CRBRP and the visual inspection requirement will be reassessed when the technology has been successfully demonstrated.

If the plant should sustain a seismic event which would impose loads of unknown magnitude, it is reasonable to assume a non-routine inspection and test of all snubbers and a corresponding inspection of the snubber attachment welds will be performed.

PART II

CRBRP INSERVICE AND PRESERVICE INSPECTION REQUIREMENTS FOR WATER/STEAM SYSTEMS

Inservice and Preservice Inspection Requirements for water/steam systems in CRBRP will be in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Division 1.

The Inservice Inspection requirements for water/steam systems are in the course of preparation.

PART III

CRBRP INSERVICE AND PRESERVICE INSPECTION REQUIREMENTS FOR REACTOR CONTAINMENT VESSEL CELL LINERS, AND SODIUM FIRE SURPRESSION FEATURES

A. Inservice and Preservice Requirements for the CRBRP Reactor Containment Vessel will be in accordance with Code of Federal Regulations 10 CFR 50, Appendix J. Reference PSAR Section 6.2.1.

B. Inservice and Preservice Requirements for Cell Liners are contained in PSAR Section 3.8-B Attachment E, and Sodium Fire Surpression Features are contained in PSAR Section 3.8-C.

CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

APPENDIX H

NUREG-0718 INFORMATION REQUIREMENTS FOR TMI-2 ACTION PLAN ITEMS-APPLICATION TO CRBRP

PROJECT MANAGEMENT CORPORATION

APPENDIX H

NUREG-0718
INFORMATION REQUIREMENTS
FOR
TMI-2 ACTION PLAN ITEMS

TABLE OF CONTENTS

NUREG-0718 ITEM NO.	SUBJECT	PAGE
	Introduction	H-3
	Index to Application to CRBRP	H-4
I.A.4.2	Long-Term Training Simulator Upgrade	H-6
I.C.5	Procedures for Feedback of Operating, Design and Construction Experience	H-7
I.C.9	Long-Term Program Plan for Upgrading of Procedures	H-10
I.D.1	Control Room Design Reviews	H-11
I.D.2	Plant Safety Parameter Display Console	H-13
I.D.3	Safety System Status Monitoring	H-15
I.F.1	Expand QA List	H-17
I.F.2	Develop More Detailed QA Criteria	H-18
II.B.1	Reactor Coolant System Vents	H-19
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	H-20
II.B.3	Post-Accident Sampling	H-21
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	H-22
II.D.1	Testing Requirements	H-26
II.D.3	Relief and Safety Valve Position Indication	H-27
II.E.1.1	Auxiliary Feedwater System Evaluation	H-28
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	H-30
II.E.3.1	Reliability of Power Supplies for Natural Circulation	H-32
II.E.4.1	Dedicated Penetration	H-33
II.E.4.2	Isolation Dependability	H-34
II.E.4.4	Purging	H-36
II.E.5.1	Design Evaluation	H-37
II.F.1	Additional Accident Monitoring Instrumentation	H-38

ITEM NO.	SUBJECT	PAGE
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	H-39
II.F.3	Instrumentation for Monitoring Accident Conditions (Reg. Guide 1.97)	H-40
II.G.1	Power Supplies for Pressurizer Relief Valves Block Valves, and Level Indication	H-42
II.J.3.1	Organization and Staffing to Oversee Design and Construction	H-43
II.K	Measures to Mitigate Small-Break LOCAs and Loss of Feedwater Accidents	H-46
III.A.1.2	Upgrade Licensee Emergency Support Facilities	H-47
III.D.1.1	Primary Coolant Sources Outside the Containment Structure	H-49
III.D.3.3	In-Plant Radiation Monitoring	H-50
III.D.3.4	Control Room Habitability	H-51

INTRODUCTION

The CRBRP Project has closely followed the developments that have resulted from the 1979 incident at Three Mile Island including the findings of NRC Lessons Learned Task Force, the Presidents Commission on the Accident at Three Mile Island, and the Report to the Commissioners and to the Public (Rogovin). New regulatory requirements emanating from these findings which potentially affect CRBRP include:

- o NUREG-0660 - "NRC Action Plan Developed as a Result of the TMI-2 Accident" (May 1980)
- o NUREG-0737 - "Clarification of TMI Action Plan Requirements" (October 1980)
- o NUREG-0718 - "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License" (August 1981)

As a pending applicant for a Construction Permit CRBRP has concentrated its review on NUREG-0718 and is fully committed to implement those requirements that are applicable or applicable in principle.

Recognizing that differences in technology between LWRs and LMFBRs affect whether the concerns which are addressed by the requirements of NUREG-0718 are appropriate for CRBRP, categorization of the requirements has been made to establish the basis for application to the project. Many requirements address specific technology or features which are not incorporated into CRBRP. Thus some requirements are applicable only to LWR's. However, some others of these requirements, while specifically written for LWR's, address a principle which is applicable to CRBRP. For each NUREG-0718 item, CRBRP has searched for the underlying engineering concern or principle, and in categorizing each item has determined how such engineering concern or principle must be applied to LMFBR technology.

This Appendix H provides CRBRP's (the applicant's) evaluation of and resolution to the requirements delineated in NUREG-0718.

APPENDIX H

NUREG-0718
INFORMATION REQUIREMENTS
FOR
TMI-2 ACTION PLAN ITEMS

APPLICATION TO CRBRP
INDEX

NUREG-0718 ITEM NO.	SUBJECT	NUREG-0718 CATEGORY	CRBRP** CATEGORY
I.A.4.2	Long-Term Training Simulator Upgrade	4	1
I.C.5	Procedures for Feedback of Operating, Design and Construction Experience	5	2
I.C.9	Long-Term Program Plan for Upgrading of Procedures	4	1
I.D.1	Control Room Design Reviews	4	1
I.D.2	Plant Safety Parameter Display Console	4	2
I.D.3	Safety System Status Monitoring	4	1
I.F.1	Expand QA List	5	1
I.F.2	Develop More Detailed QA Criteria	5	1
II.B.1	Reactor Coolant System Vents	4	1
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	4	1
II.B.3	Post-Accident Sampling	4	2
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	5	2
II.D.1	Testing Requirements	4	3
II.D.3	Relief and Safety Valve Position Indication	4	3
II.E.1.1	Auxiliary Feedwater System Evaluation	3	1
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	4	1
II.E.3.1	Reliability of Power Supplies for Natural Circulation	4	2
II.E.4.1	Dedicated Penetration	5	3
II.E.4.2	Isolation Dependability	4	2
II.E.4.4	Purging	4	2
II.E.5.1	Design Evaluation	4	3
II.F.1	Additional Accident Monitoring Instrumentation	4	2

NUREG-0718 ITEM NO.	SUBJECT	NUREG-0718 CATEGORY	CRBRP** CATEGORY
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	4	1
II.F.3	Instrumentation for Monitoring Accident Conditions (Reg. Guide 1.97)	4	2
II.G.1	Power Supplies for Pressurizer Relief Valves Block Valves, and Level Indication	4	3
II.J.3.1	Organization and Staffing to Oversee Design and Construction	5	1
II.K	Measures to Mitigate Small-Break LOCAs and Loss of Feedwater Accidents	-	3
III.A.1.2	Upgrade Licensee Emergency Support Facilities	4	1
III.D.1.1	Primary Coolant Sources Outside the Containment Structure	4	2
III.D.3.3	In-Plant Radiation Monitoring	4	2
III.D.3.4	Control Room Habitability	4	1

**CRBRP Categories of Applicability

1 - Applicable to CRBRP

2 - Applicable in Principle to CRBRP but Modified to reflect the different Technology of an LMFBR.

3 - Would be applicable to CRBRP, but the (specific) intent and specifics of the requirements address technology, equipment, and features of reactors not used or found in CRBRP. Thus there is no appropriate action required.

I.A.4.2 LONG-TERM TRAINING SIMULATOR UPGRADE

REQUIREMENT:

Applicants shall describe their program for providing simulator capability for their plants. In addition, they shall describe how they will assure that their proposed simulator will correctly model their control room. Applicants shall provide sufficient information to permit the NRC staff to verify that they will have the necessary simulator capability to carry out the actions described in this Action Plan item as well as Action Plan Item II.K.3.54. Applicants shall submit, prior to the issuance of construction permits, a general discussion of how the requirements will be met. Sufficient details shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP. CRBRP will require training and retraining of operators and a high level of realism will be needed in the modeling of the control room and the plant operating characteristics as is required for other power reactors. Plant systems and technology are different from other power reactors and accordingly the operating characteristics will be unique. Thus the CRBRP simulator will need to provide capability to simulate a spectrum of operating conditions and transients, including failures and errors, different from those for LWR's.

IMPLEMENTATION:

A CRBRP plant simulator will be provided which correctly models the Control Room and which has the capability for presenting displays and alarms, and for responding to operator actions, in a manner reflecting actual plant operating conditions. A spectrum of normal operating and transient conditions will be modeled which will include appropriate pipe breaks, equipment failures and operator errors. This spectrum of operating conditions will be sufficient to provide the necessary training for operators in plant diagnostics and corresponding corrective actions under all expected plant conditions.

The design of the simulator will conform appropriately to those industry standards and regulatory requirements in effect at procurement. Design specifications are currently being developed.

I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING, DESIGN AND CONSTRUCTION EXPERIENCE

REQUIREMENT:

Applicants shall submit a description of their administration procedures for evaluating operating, design, and construction experience and describe how they will assure that applicable important industry experiences originating from both within and outside the applicant's construction organization will be provided in a timely manner to those designing and constructing the plant. Applicants shall submit a general discussion of how the requirements will be met. These procedures shall: (1) Clearly identify organization responsibilities for review and identification of these important experiences and the feedback of pertinent information to those responsible for designing and constructing the plant; (2) Identify the administrative and technical review steps necessary in implementing applicable important experiences; (3) Identify the recipients of various categories of information from these experiences or otherwise provide means through which such information can be readily related to the job functions of the recipients; (4) Assure that applicant and contractor personnel do not routinely receive extraneous and unimportant experience-related information in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency; (5) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to applicant and contractor personnel for implementation until resolution is reached; and (6) Provide practical interim audits to assure that the feedback program functions effectively at all levels. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of construction permits or manufacturing license.

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP.

IMPLEMENTATION:

The CRBRP Project Office and principal contractors have established procedures to evaluate major sources of pertinent industry experience, including:

1. Light water reactor experience described in NRC's office of Inspection and Enforcement bulletins, circulars, and notices.

2. Liquid Metal Fast Breeder Reactor Experience from the Fast Flux Test Facility (FFTF).

3. Each unusual or unplanned event (unusual occurrence) in the CRBRP Project having programmatic significance.

4. The Project commits to active interactions with INPO. The details will be developed further in meetings with INPO personnel planned for early January, 1983.

NRC's I&E Bulletins, Circulars, and Notices

When an NRC I&E Bulletin, Circular, or Notice is received, a responsible Project Office organization evaluates the NRC bulletin, circular or notice, confirms its applicability to CRBRP, informs and involves contractors to the degree required, determines the actions required for resolution, and subsequently develops any formal response that may be required or desired. Status logs and files are maintained.

The procedure fulfills the specific requirements of item I.C.5 as follows:

1. Organization responsibilities are clearly identified.

2. The administrative steps necessary to implement applicable important experiences are clearly established.

3. The procedure assigns a project organization responsibility to determine the necessary recipients of the information.

4. Assurance that extraneous material does not overwhelm personnel is accomplished by: (1) initial screening to determine appropriate applicability, (2) further assessment by the responsible line organizations that the item is applicable before assigning it for action.

5. Assurance that conflicting or contradictory information is not conveyed to personnel assigned action is accomplished by multiple levels of screening.

6. The procedure is audited regularly for compliance and effectiveness in accordance with established Project requirements.

The Assistant Director for Public Safety ensures that NRC I&E Bullentins, Circulars and Notices are properly evaluated by the responsible CRBRP Project Divisions, who implement and follow-up on the necessary activities.

FFTF Experience

The "FFTF Experience Integration" procedure's objective is to evaluate and utilize experiences from the construction, test, start-up and operation of the Fast Flux Test Facility (FFTF) operated by the Westinghouse Hanford Company for the Department of Energy in Richland, Washington.

A CRBRP representative was onsite at FFTF during Construction, and start-up and provided valuable feedback of experience and recommendations. Following start-up the FFTF Project Office is assigned the responsibility to provide CRBRP with descriptions of FFTF operating experience. Weekly reports are provided to the Lead Reactor Manufacturer for initial evaluation; important experiences are identified and recorded. A CRBRP Project organization is assigned responsibility for implementation of the experience.

The organization assigned the action has the responsibility to evaluate and resolve the experience, document the implementation and provide a description of the resolution action to the Lead Reactor Manufacturer and the Project Office for approval. The FFTF experience is maintained in the "FFTF Experience Items Semi-annual Report" and the responsible organization is

required to present the results of the resolution action at the appropriate system final design review.

The Assistant Director for Engineering has responsibility for ensuring FFTF experience is implemented into the CRBRP Project. He appoints an Experience Coordinator to coordinate Project Office activities relative to industry experience identification and evaluation.

INTERNAL REPORTABLE OCCURANCES

It is the policy of the CRBRP Project to review and disposition via approved procedures, all Unusual Occurance Reports resulting from unusual occurances during the design, development, procurement, testings, construction, start-up, operation, maintenance, or modification of the CRBRP. An unusual occurrence is an unusual or unplanned event having programmatic significance such that it adversely affects or potentially affects the performance, reliability , security, or safety, of a reactor or test facility, or causes injury or potential hazard to personnel which may require evaluation, correction, or preventive actions to be taken.

The Unusual Occurrence Report System is structured to meet DOE Standard requirements. PSAR Chapter 17 contains information on unusual occurrence reporting and its implementation in the CRBRP Quality Assurance Program.

I.C.9 LONG-TERM PROGRAM PLAN FOR UPGRADING OF PROCEDURES

REQUIREMENT:

Applicants shall describe their program plan, which is to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analysis, human factors engineering, crisis management and operator training. Applicants shall also insure that their program will be coordinated, to the extent possible, with INPO and other industry group efforts. Applicants will submit, prior to the issuance of construction permits, a general discussion of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP.

IMPLEMENTATION:

CRBRP's program plan to develop plant procedures requires the operator, TVA, to write detailed operating procedures using as a guide procedure outlines developed by designers. The CRBRP design organizations have the responsibility to assure that the design properly considers all aspects affecting safe operability of the plant. In preparing these procedure outlines, the most recent recommendations on upgrading procedures are being considered.

When TVA prepares detailed plant procedures from these outlines the most recent industry recommendations on procedure development will be considered. TVA, as a member of INPO, will use this relationship to maintain an awareness of the 'state-of-the-art' in procedure upgrading and development.

As a follow-up of "TMI Lessons Learned" an extensive review and update of the CRBRP control room procedure outlines were made. This was accomplished by having a multidisciplinary review team 'walk through' the operating procedure outlines and then upgrade the procedure outlines to reflect the experience gained. The scope of this activity included both normal operating procedures and emergency procedures. Reliability analysis and human factors engineering was a major contribution to this review.

The CRBRP Probabilistic Risk Assessment (PRA) Program will also be utilized to assess procedures and operator interaction. Appendix J, Program Plan for the CRBRP PRA, paragraphs 2.6.1 and 2.6.3, provide additional information on the utilization of Operator Action Event Trees to investigate the role of the plant operation staff in important accident sequences and additional PRA applications to improve understanding of the plant. These PRA applications utilize best estimate analyses. Further, the Project will also evaluate plant safety functions to ensure adequate procedures for events with failure combinations not specifically addressed through the PRA effort.

I.D.1 CONTROL ROOM DESIGN REVIEWS

REQUIREMENT:

Applicants shall provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Applicants shall provide a general discussion of their approach to control room designs that reflect human factor principles by specifying the design concept selected and the supporting design bases and criteria. Cosmetic revisions to conventional (1960 technology) designs are unacceptable. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses. Applicants shall commit to control room designs reflecting human factors principles prior to issuance of a CP or ML and shall supply design information for review prior to committing to fabrication or revision of fabricated Control Room panels and layouts.

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP. The Control Room in the plant will serve the same function as in any other power reactor and accordingly the same kinds of operational and human factors design considerations must be applied.

IMPLEMENTATION:

Following the incident at Three Mile Island, CRBRP conducted an in depth review of the Control Room design. The objective of the review was to ensure that the design of the Control Room and the operations procedure outlines will facilitate safe operability of the plant, through consideration of human-factors engineering principles, systems design and integration and Control Room configuration. A review team was selected with expertise in design, analysis, operations, testing and human factors engineering.

The review process included manual simulation of a spectrum of actual plant events and supporting analyses of the required operator actions, detailed analyses of transients, corresponding Control Room indications, the controls available to the operator for responding to those indications, and the role and function to be fulfilled by the operator for each situation. This simulation was conducted using a full-scale mock-up of the Main Control Room panel. The methodology required analysis of each simulated event to derive expected control panel indications, operator walk-throughs of procedure outlines for that event, and evaluations of the procedures. Recommendations for changes were prepared for the appropriate assessment and implementation by affected line organizations within the Project.

The CRBRP control room design reflects state-of-the-art human factors principals. Documents will be submitted to demonstrate that our design methodology is equivalent to that described in NUREG-0700 "Guidelines for Control Room Design Review," including Appendix B which describes methods of systems analysis, functional analysis, task analysis, and sequence analysis. The control room incorporates an advanced design information system, including computer based CRT displays. The control room design and the documents that

demonstrate that the design methods are equivalent to those described in NUREG-0700 will be provided to NRC prior to committing to fabrication of the main control panel or simulator control panels.

NUREG-0700 presents human factors engineering guidelines and sample checklists for analyzing operator-control room interfaces. Appendix B to NUREG-0700 describes the analyses which the NRC believes is an acceptable approach to control room design. The methodology defines the equipment, personnel, and procedures required to meet all functional objectives of the control room. The methodology is summarized in the following.

The first step is to perform a systems/operator analysis to define functions and interaction.

The next step in the development of Control Room design is the allocation of functions to operator or machine. The basis of the allocation is actual/anticipated performance of the operator or the machine in the implementation of the function. The actual/anticipated performance is based on actions in which humans excel, such as the ability to exercise judgment where events cannot be completely defined, and on actions in which machines excel, such as carrying out many tasks simultaneously. Function allocation based on actual/anticipated performance of the operator or the machine is preferred over random allocation.

The tasks required to perform each function are then defined and analyzed to establish design details. The design details for functions and tasks allocated to machines become performance requirements, data needs, and decision points in operational sequences. These design details will impact work station design, operator training requirements, and elements of normal and emergency operating procedures. Preliminary design is then conducted by specifying work station configurations and by establishing guidelines for procedures.

The next step in the design process is to verify the integration of humans and machines at the work station level. This process is achieved by analyzing work station design, operation sequence, operator workload, human error rate, and work station links. The goal is to verify that the human can indeed perform all assigned functions and tasks with the specified work station design. If the goal is not achieved, then a redesign beginning with a reallocation of function is necessary. In reallocating the function, the designer should consider all options, such as work station redesign, new training, new procedures, new job aids, the addition of more operators, and automation of the function.

The Control Room configuration design is achieved by integrating work stations based on their functions and interactions. The Control Room configuration design is then validated against the functional requirements established from operations analysis. This process is achieved by simulating operations with a Control Room mock-up or a Control Room simulator. Alternatively, computer program synthesis of Control Room design and operations may prove to be cost effective in evaluating the integrated design. The goal is to validate that humans and work stations can interact to achieve all required functions. When a failure to verify/validate a task/function occurs, it will be necessary to

reallocate/redesign and then reverify and revalidate. The final step in the design process is to document the Control Room design specifications.

The Project has utilized and is continuing to use a design and review process which is equivalent to the process outlined above and detailed in NUREG-0700 Appendix B. Further demonstration of this equivalence will be provided in documentary form in advance of initiation of fabrication of the main control panel.

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

REQUIREMENT:

Applicants shall describe how they intend to meet the staff criteria contained in NUREG-0696 for a plant safety parameter display console. The console shall display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable in principle to CRBRP. Concise displays of plant operating status will enhance control room operators' abilities to prevent accidents or to cope with accidents if they occur. CRBRP operating characteristics and design are different from those of LWR's and so accordingly are the man-machine interfaces and operator responses. The published criteria are more specific to the needs of LWRs', and accordingly a separate criteria pertaining to safety functions unique to LMFBRs must be applied.

IMPLEMENTATION:

CRBRP recognizes the need to provide plant operators with sufficient information in convenient formats to assist in determining the safety status of the plant. CRBRP commits to providing a human factors engineered Safety Parameter Display Function (SPDF) that will display to the operators a set of parameters for assisting the operators in assessing the safety status of CRBRP. The SPDF is in the conceptual design phase for CRBRP. The concept being considered is consistent with the intent of guidance provided in NUREG-0718 and NUREG-0696 as follows:

System Description

The primary function of the Safety Parameter Display Function would be to aid the operator in the rapid detection of abnormal operating conditions by providing a continuous indication of plant parameters or derived variables representative of the safety status of the plant. The system will have the capability of displaying the full range of important plant parameters and data trends on demand. The system will also indicate when plant parameters are approaching or exceeding process limits. The safety parameter displays would be provided as part of and integrated with displays provided by the state-of-the-art Plant Computer System. Backup secondary indication would be provided on the Main Control Panel in human engineered functional arrangements.

a. Location

The safety parameter displays would be available in the control room as well as in the Technical Support Center (TSC). The displays would be readily accessible and visible to control room operators in the normal operating area but would not interfere with normal movement or with full visual access to other control room systems or displays. The safety parameter displays would be readable from the control room senior reactor operator's emergency operating station.

b. Parameters

The set of safety parameters to be displayed will be determined by the ongoing project design efforts. The displayed parameters will be selected consistent with Regulatory Guide 1.97, Rev. 2, as applied to CRBRP, to enable the operator to determine the safety status of the plant process.

The display formats will be designed in accordance with human factors principles. Data trends would be available on demand and the system would be capable of indicating when process limits are being approached or exceeded, for the selected set of safety parameters.

c. Reliability

The safety parameter displays used in the control room would be designed to an operational unavailability goal of .01. The term unavailability is used to express a complete loss of the primary safety parameter displays.

The primary safety parameter display information would be provided by the PDH&DS, a high quality and highly reliable computer system. The SPDF display, and the computer central process and data file equipment would be located in the control room environment which is controlled during transient or accident conditions. Backup secondary display information will be provided on the control panel utilizing the redundant, 1E, seismically qualified displays provided in compliance with Regulatory Guide 1.97 as applied to CRBRP. These displays are identified as Accident Monitoring Category 1 instruments and are arranged to clearly and quickly indicate to the operator whether safety systems are performing their design function.

SPDS concepts being utilized for light water reactor plants have been reviewed as being typical of the state-of-the art. These reviews have indicated that a system for CRBRP can be designed and implemented effectively.

I.D.3 SAFETY SYSTEMS STATUS MONITORING

REQUIREMENT:

Applicants shall describe how their design conforms to Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is, technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP. Control room operators must know at all times the operability status of plant safety systems.

IMPLEMENTATION:

CRBRP will provide automatic indication in the control room of each bypass or deliberately induced inoperable status of any portion of an automatic protection system, the safety systems actuated or controlled by the protection system, and the auxiliary or supporting systems that must be operable for the protection system and the systems it actuates or controls to perform their safety-related functions. This indication will be performed by the Inoperable State Monitoring System (ISMS) which will be designed to comply with Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety System." To the extent practical, inputs to the ISMS will be direct measurements from the desired variables.

ISMS instrumentation and control room displays will be provided to the extent necessary for automatic indication at the system level of the intentionally rendered inoperable or bypassed status of any system that is important to the health and safety of the public, except for the Reactor Shutdown System (RSS) which provided separate indication, and the Containment Isolation System which has no identified bypass modes. The ISMS will monitor the following systems:

- Decay Heat Removal System
- Fuel Storage Heat Removal System
- Control Room Habitability
- Annulus Filtration
- RSB Filtration

Automatic indication will be provided in the control room for each bypass or deliberately induced inoperable status that meets all of the following conditions:

- Renders inoperable any redundant portion of the protection system, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection system and the systems it actuates to perform their safety-related functions; and

- It is expected to occur more frequently than once per year, or if expected to occur less frequently than once per year, the consequence of the condition would severely limit the ability of the safety system to perform its safety function; and

- Is expected to occur when the affected system is normally required to be operable.

Manual capability shall exist in the control room to activate each system-level indicator provided.

The ISMS equipment will be isolated, where necessary, from the associated safety-related equipment so as to preclude any abnormal or normal action of the ISMS from preventing the performance of a safety function.

The Plant Data Handling and Display System (PDH&DS) will provide the ISMS functions for the CRBRP. When an ISMS input signal is received, the signal is analyzed to determine if a system has been rendered inoperable and the source of the signal is printed out on the PDH&DS logger printer. If it is determined that a system has been rendered inoperable, the ISMS panel, which is located on the supervisors panel, will indicate which system is inoperable and the ISMS annunciator on the Main Control Panel will provide an audible and visual alarm.

I.F.1 EXPAND QA LIST

REQUIREMENT:

Prior to issuance of the construction permits or manufacturing license, applicants shall revise their QA programs by expanding their QA lists to include all items and activities affecting safety as defined by Regulatory Guide 1.29 and Appendix A to 10 CFR Part 50, and shall provide a commitment to apply the revised QA program to all such items and activities.

APPLICATION TO CRBRP:

This requirement is directly applicable to CRBRP.

IMPLEMENTATION:

The Owner's Quality Assurance Program described in Chapter 17 of the PSAR is applicable to the planning, design, procurement, manufacturing, construction including testing, and operation of those safety-related structures, systems, and components identified in Section 3.2, 7.1 and 9.13 of this PSAR. For the Owner's purposes, the program described is not limited to this application alone, however, and is or will be appropriately applied by the Owner to the CRBRP in its entirety including all structures, systems and components where satisfactory performance is required for the plant to operate reliably, safely and with minimum environmental effects.

In accordance with the basic intent of 10 CFR 50, Section 50.55a and Regulatory Guide 1.26 and 1.29 safety related CRBRP system components and piping have been classified as Safety Class 1 (SC-1), Safety Class 2 (SC-2), or Safety Class 3 (SC-3). The importance, as established by safety class assignment, is considered in the design, material selection, fabrication, erection, construction and operational aspects of this equipment.

I.F.2 DEVELOP MORE DETAILED QA CRITERIA

REQUIREMENT:

Applicants shall describe the changes to their QA programs that have resulted from their review of the accident at TMI-2. In addition, applicants shall address the appropriate matters discussed in this Action Plan Item, including the establishment of a quality assurance (QA) program based on consideration of: (a) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions: (b) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (c) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (d) establishing criteria for determining QA programmatic requirements; (e) estabishing qualification requirements for QA and QC personnel; (f) sizing the QA staff commensurate with its duties and responsibilities; (g) establishing procedures for maintenance of "as-built" documentation; and (h) providing a QA role in design and analysis activities. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a revised description of their QA program that includes consideration of these matters.

APPLICATION TO CRBRP:

This requirement is directly applicable to CRBRP.

IMPLEMENTATION:

The CRBRP Quality Assurance Program as described in Chapter 17 of the PSAR was already developed, giving full consideration to the eight (8) criteria in Item I.F.2. The Program description has been re-reviewed and clarified to assure that these criteria are fully addressed.

II.B.1 REACTOR COOLANT SYSTEM VENTS

REQUIREMENT:

Applicants shall modify their plant designs as necessary to provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. Applicants shall, to the extent possible provide preliminary design information at a level consistent with that normally required at the construction permit state of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting these requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP. Unplanned accumulations of non-condensible gases within the reactor coolant system could potentially compromise required core cooling.

IMPLEMENTATION:

An inert cover gas is provided over free liquid sodium surfaces including within the reactor vessel. This cover gas is continuously pumped to circulate through the top of the reactor vessel at controlled pressures of a few inches W.G. above atmospheric pressure. Any non-condensible gases entrained within reactor coolant within the reactor vessel will mainly be released to this cover gas and circulated away for processing and removal of unwanted gas before re-use.

The other high point in the primary heat transport system is at the intermediate heat transport system heat exchanger, (IHX) which does not have a free liquid sodium surface. To preclude potential non-condensible gas accumulation in the IHX, there is continuous venting of sodium from the top of the IHX to the primary pump tank via the IHX vent return line. This sodium flow will carry any gas evolved in the IHX to the pump tank where the gas will migrate to the cover gas space.

Further discussion is provided in PSAR Sections 5.1.2 and 5.3. This concept has been successfully demonstrated as feasible and within the state-of-the-art at the Fast Flux Test Facility.

II.B.2 PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION

REQUIREMENT:

Applicants shall (1) perform radiation and shielding design reviews of spaces around systems that may contain highly radioactive fluids and (2) implement plant design modifications necessary to permit adequate access to vital areas and protect safety equipment. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is directly applicable to CRBRP.

IMPLEMENTATION:

As part of the normal design process, the Project has performed radiation and shielding design reviews of spaces around systems that may contain highly radioactive fluids. Source terms for equipment have been developed for the components of these systems. In addition the Project has conducted as part of the key system review process an assessment of the accessability of equipment following accidents such as sodium spills, radioactive gas leaks, and an hypothesized core melt accident. Following these key system reviews, the Project has modified the design to assure accessability of components and control panels and acceptability of the accident consequences. The PSAR has been appropriately modified to reflect the changes resulting from the above.

II.B.3 POST-ACCIDENT SAMPLING

REQUIREMENT:

Applicants shall (1) review the reactor coolant and containment atmosphere sampling system designs and the radiological spectrum and chemical analysis facility designs, and (2) modify their plant designs as necessary to provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable in principle to CRBRP. While sampling of reactor coolant and containment atmosphere can provide means for assessing core damage in the event of an accident, the specifics of this requirement are addressed to the needs of LWR's and a correlation to LMFBR technology is required.

IMPLEMENTATION:

Means to assess the degree of core damage in the event of an accident will be provided by CRBRP as part of the plant accident monitoring capability.

The approach to be utilized has not yet been finalized. It is anticipated that cover gas monitoring may be utilized.

II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

REQUIREMENT:

Applicants shall:

(1) commit to performing a site/plant-specific probabilistic risk assessment and incorporating the results of the assessment into the design of the facility. The commitment must include a program plan, acceptable to the staff, that demonstrates how the risk assessment program will be scheduled so as to influence system designs as they are being developed. The assessment shall be completed and submitted to NRC within two years of issuance of the construction permit. The outcome of this study and the NRC review of it will be a determination of specific preventive and mitigative actions to be implemented to reduce these risks. A prevention feature that must be considered is an additional decay heat removal system whose functional requirements and criteria would be derived from the PRA study.

It is the aim of the Commission through these assessments to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and doe not impact excessively on the plant. Applicants are encouraged to take steps that are in harmony with this aim.

(2) include provisions in the containment design for one or more dedicated penetrations, equivalent in size to a single 3-foot diameter opening. This shall be done in order not to preclude the installation of systems to prevent containment failure, such as filtered vented containment systems.

(3) provide a system for hydrogen control capable of handling hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction.

(4) provide preliminary design information at a level consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(a) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering the requirements of the ASME Boiler and Pressure Vessel Code, Division 2, Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent, depending upon which option is chosen for control of hydrogen. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown

by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(b) The containment and associated systems will provide reasonable assurance that uniformly-distributed hydrogen concentrations do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(c) The facility design will provide reasonable assurance that, based on a 100% fuel clad metal-water reaction, combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(d) If the option chosen for hydrogen control is post-accident inerting: (a) Containment structure loadings produced by an inadvertent full inerting (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Division 1, Subsubarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Division 2, Subsubarticle CC-3720, Service Load Category), (b) demonstrate that a pressure test, which is required, of the containments at 1.10 and 1.15 times for steel and concrete containments, (respectively) the pressure calculated to result from carbon dioxide inerting can be safely conducted, (c) demonstrate that inadvertent full inerting of the containment can be safely accomodated during plant operation.

(e) If the option chosen for hydrogen control is a distributed ignition system, equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity shall be designed to perform its function during and after being exposed to the environmental conditions created by activation of the distributed ignition system.

CRBRP APPROACH:

The overall objective of these requirements is to enhance public safety and reduce individual and societal risk (through) consideration of core degradation and melting beyond the design basis. (NUREG 0660, Task II.B - Objective)

NRC and the Project were meeting this objective for CRBRP prior to the incident at TMI, and continue to do so through consideration of structural

and thermal margins beyond the design base (SMBDB and TMBDB). These aspects of the plant design are discussed in CRBRP-3 Volumes 1 and 2, (PSAR Section 1.6, Item 10a & b).

REQUIREMENT 1

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP.

IMPLEMENTATION:

PRA studies have been completed. Additional PRA studies will be completed.

REQUIREMENT 2

APPLICATION TO CRBRP:

This requirement is applicable in principle to CRBRP

The CRBRP design includes a filtered containment venting system with redundant dedicated pentrations.

REQUIREMENT 3

APPLICATION TO CRBRP:

This requirement is applicable in principle to CRBRP

The containment vent and purge systems are provided for control of hydrogen not otherwise consumed-for hypothesized core melt including reaction of 100% of water released from the reactor cavity floor into the sodium pool.

REQUIREMENT 4a

APPLICATION TO CRBRP:

This requirement is applicable in principle to CRBRP

Containment integrity will be maintained - considering pressure and dead loads - considering hydrogen burning, and purging. Systems necessary to ensure containment integrity - for hypothetical core melt will be designed to perform under TMBDB conditions.

REQUIREMENT 4b

APPLICATION TO CRBRP:

This requirement is applicable in principle to CRBRP

The containment vent and purge systems assure that uniform hydrogen concentrations do not exceed 10% for hypothesized core melt including reaction of 100% of water released from the reactor cavity floor into the sodium pool.

REQUIREMENT 4c

APPLICATION TO CRBRP:

This requirement applicable in principle

The capabilities and configurations of reactor cavity vent, containment vent and containment purge systems provide assurance that combustible concentrations of hydrogen will not collect for hypothesized core melt including reaction of 100% of the water released from the reactor cavity floor into the sodium pool.

REQUIREMENT 4d

APPLICATION TO CRBRP

This requirement is not applicable to CRBRP

Post-accident inerting is not the option chosen for hydrogen control.

REQUIREMENT 4e

APPLICATION TO CRBRP

This requirement is not applicable to CRBRP

A distributed ignition system is not the option chosen for hydrogen control.

II.D.1 TESTING REQUIREMENTS

REQUIREMENTS:

Applicants and their agents shall provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transient without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed and not before issuance of an ATWS rule. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a general explanation of how the testing requirements will be met. Sufficient detail should be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Applicants shall (1) demonstrate the applicability of the generic tests conducted under II.D.1 to their particular plants and (2) modify their plant designs as necessary. Applicants shall commit, prior to the issuance of the construction permits or manufacturing license, to comply with these requirements and shall submit within six months following the completion of the generic tests or the issuance of construction permits, whichever is later, a detailed explanation of how the test results will be incorporated in the plant design. Sufficient detail should be presented to provide reasonable assurance that the requirements resulting from the test will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement would be applicable to CRBRP if the project design incorporated a pressurized reactor coolant system which requires relief and safety valves or PORV block valves. However CRBRP has an unpressurized reactor coolant system and no safety and relief valves or PORV block valves are used. Thus neither the requirement nor the principle of the requirement have any application on CRBRP. CRBRP design does include an inert cover gas to be maintained at a few inches W.G. above atmospheric pressure over the free liquid sodium surface of the reactor coolant in the reactor vessel and the primary pump tanks. Overpressure protection for the relief of this cover gas is provided in accordance with Section III of the ASME Boiler and Pressure Vessel Code. However, the relief valving which is provided is required only to pass single phase gas. Thus no special considerations are involved which would require special and unique testing.

II.D.3 RELIEF AND SAFETY VALVE POSITION INDICATION

REQUIREMENT:

Applicants shall modify their plant designs as necessary to provide direct indication of relief and safety valve position in the control room. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement would be applicable to CRBRP if the project design incorporated a pressurized reactor coolant system which requires relief and safety valves. However CRBRP has an unpressurized reactor coolant system and no relief and safety valves are used. Thus neither the requirement nor the principle of the requirement have any application on CRBRP.

CRBRP design does include an inert gas to be maintained over the free liquid sodium surface of the reactor coolant in the reactor vessel. Overpressure protection for the relief of this gas is provided. However, a cover gas pressure relief valve which has failed open has no safety consequence and thus no special consideration for indicating the position status of this valve is required.

Also, should a relief or safety valve fail to close in one loop of the steam generator system, since the other loops are separate and independent their ability to perform their required safety function will not be affected.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

Applicants shall perform a reevaluation of their proposed auxiliary feedwater (AFW) system. This reevaluation shall include the following:

(1) Performance of simplified auxiliary feedwater system reliability analyses using event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss of main feedwater transient conditions, with particular emphasis being given to determining potential failures that could result from human errors, common causes, single point vulnerabilities, and test and maintenance outages. The results of this evaluation shall be compared with the results of the NRC staff's generic AFW system evaluation published in Appendix III to NUREG-0611 and Appendix III to NUREG-0635. Applicants with plants with AFW systems with relatively low reliabilities shall submit proposed design changes and/or procedural actions which will improve the relative reliability of the AFW system to above average. Applicants whose plant designs do not include high head high pressure injection system pumps for use in the feed and bleed mode of decay heat removal in case of AFW system failure shall assure that their AFW system has a very high reliability relative to those AFW systems evaluated by the NRC and staff and reported in NUREG-0611 and NUREG-0635 respectively.

(2) Completion of deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 as principal guidance. This requirement applies to those plants where the Standard Review Plan was not used as criteria during the NRC staff's CP review.

(3) Reevaluation of the AFW system flow design bases and criteria. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP. The auxiliary feedwater system (AFWS) serves the same function in CRBRP as in a PWR plant.

IMPLEMENTATION:

The reliability of the AFWS has been studied in three separate programs. The decay heat removal key system review included event-tree and fault-tree study of the AFWS and qualitative investigation of vulnerability to human factors, single point failures, common causes, test and maintenance failures. Similar qualitative review of the AFWS was performed in the project's qualitative reliability program. The qualitative reliability program centers around the use of detailed failure modes and effect analyses and common cause failure analyses. A quantitative analysis of the AFWS as part of the overall shutdown heat removal system (SHRS) was performed using a failure state model of the SHRS. The quantitative and qualitative reliability program activities are discussed in PSAR Appendix C.

The CRBRP AFWS has been designed in compliance with the criteria of Standard Review Plan Section 10.4.9. Specifically with regard to Branch Technical Position ASB 10-1, an evaluation of the CRBRP AFWS versus the Standard Review Plan is provided in PSAR Section 5.6.1.3.12.

PSAR Section 5.6.2 also provides the basis for determining the flow requirements for the AFWS. These bases will be reviewed against the re-analysis guidelines provided in enclosure 2 of the NRC (D. F. Ross, Jr.) letter to pending operating license applicants of NSSS plants designed by Westinghouse and Combustion Engineering, dated March 10, 1980.

II.E.1.2. AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION

REQUIREMENT:

Applicants with PWR plants shall provide automatic and manual auxiliary feedwater (AFW) system initiation and auxiliary feedwater system flow indication in the control room. These systems shall be safety grade and meet the requirements specified in NUREG-0737. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP. The auxiliary feedwater system (AFWS) serves the same function in CRBRP as in a PWR plant.

IMPLEMENTATION:

(a) The CRBRP design incorporates the automatic initiation of the AFWS in accordance with NRC requirements. The design of the AFWS as a part of the Steam Generator Auxiliary Heat Removal System (SGAHRS) is presented in PSAR Sections 5.6.1, 7.1.1, 7.1.2 and 7.4.1. These sections discuss:

1. the provision for automatic initiation of the AFWS;

2. the requirement that a single failure of the initiation signals and circuits, and manual circuits will not result in loss of AFWS or SGAHRS function;

3. the testability of the initiating signals and circuits;

4. that AFWS can be initiated by manual means from the control room;

5. the supply of the initiating signals and circuits from the emergency buses;

6. the inclusion of the ac motor-driven pumps and valves into the automatic actuation sequence of the loads onto the emergency buses;

7. the capability to manually initiate the AFWS from the control room with failure of the automatic initiating signals and circuits;

8. and the automatic initiating signals and circuits design in accordance with safety-grade requirements.

(b) The CRBRP design includes a venturi flowmeter in each AFWS line to a steam drum and the provision of an indication in the control room of AFW flow through each venturi. The design of the AFWS as a part of the SGAHRS is presented in PSAR Sections 5.6.1, 7.1.1, 7.1.2, and 7.4.1. These sections discuss the inclusion of a safety-grade indication of AFW flow to each steam drum, and the emergency power supplies to the flow indication instrument channels. Thus the requirements of NUREG-0718 and NUREG-0737 have been fully met.

II.E.3.1 RELIABILITY OF POWER SUPPLIES FOR NATURAL CIRCULATION

REQUIREMENT:

Applicants shall (1) upgrade the power supplies for the pressurizer heaters and associated motive and control power interfaces to meet the applicable requirements specified in NUREG-0737 and (2) establish procedures and training for maintaining the reactor coolant system at hot standby conditions with only onsite power available.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP in principle. Equipment or systems which are needed to establish and maintain natural circulation, including instrumentation, must be provided with reliable power supplies as they are required. However, the requirements are specifically stated for the needs of LWR's and are not directly appropriate for LMFBR's, such as for pressurizer heaters which are not included in CRBRP design.

IMPLEMENTATION:

CRBRP design includes provisions for reliable power supplies to equipment which is required for establishing and maintaining natural circulation. Design reviews have been conducted to assure the adequacy of power supplies. Instrumentation will be provided in the control room which will meet the design criteria of Regulatory Guide 1.97, Rev. 2 as applied to CRBRP. These design criteria will define the requirements for reliable power supplies for this instrumentation.

II.E.4.1 DEDICATED PENETRATION

REQUIREMENT:

Applicants for plant designs with external hydrogen recombiners shall modify their applications as necessary to include redundant dedicated containment penetrations so that assuming a single failure, the recombiner systems can be connected to the containment atmosphere. Applicants shall submit, prior to the issuance of construction permits or the manufacturing license, a detailed explanation of how the requirements will be met in order to provide reasonable assurance that the requirements will be implemented properly.

APPLICATION TO CRBRP:

This requirement would be applicable to CRBRP if the project design incorporated external recombiners. However, hydrogen is not generated as a result of any design basis event for CRBRP. No hydrogen recombiners are provided outside of containment for such design basis events thus there is no need to provide any dedicated penetrations to the containment in conformance to this requirement.

II.E.4.2 ISOLATION DEPENDABILITY

REQUIREMENT

Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4.

All plants shall give careful consideration to the definition of essential and non-essential systems, identify each system determined to be essential, identify each system determined to be non-essential, and describe the basis for selection of each essential system. All non-essential systems shall be automatically isolated by the containment isolation signal. Revision 2 to Regulatory Guide 1.141 will contain guidance on the classification of essential versus non-essential systems and is due to be issued by June 1981.

For post-accident situations, each non-essential penetration (except instrument lines) is required to have two isolation barriers in series that meet the requirements of General Design Criteria 54, 55, 56, and 57, as clarigied by Standard Review Plan, Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Manual valves must be sealed closed, as defined by Standard Review Plan, Section 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a non-essential penetration must receive diverse isolation signals.

The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action. Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting this requirement.

Ganged reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be performed on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

The containment setpoint pressure that initiates containment isolation for non-essential penetrations must be reduced to the minimum compatible with normal operating conditions. The containment pressure history during normal operation for similar operating plants should be used as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. The pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 1 psi will require detailed justification.

All systems that provide a path from the containment to the environs (e.g., containment purge and vent systems) must close on a safety-grade high radiation signal.

II.E.4.4 PURGING

REQUIREMENT:

Applicants shall (1) provide a capability for containment purging/venting designed to maximize purging time, consistent with ALARA principles for occupational exposure, (2) evaluate the performance of purging and venting isolation valves against accident pressure, (3) address the interim NRC guidance on valve operability, (4) adopt procedures and restrictions consistent with the revised requirements, and (5) provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable in principle. This requirement is appropriate for an LWR but less so for an LMFBR. Accident pressures under DBA and TMBDB conditions in CRBRP are much lower than those in LWRs and are well within the capacity of the CRBRP containment boundary. Also, the rate of pressure buildup under accident conditions is lower. Therefore, strict application of this requirement (which is more appropriate for LWRs) is not warranted for CRBRP.

IMPLEMENTATION:

This requirement has been implemented on CRBRP to the extent consistent with LMFBR technology. PSAR Section 9.6 identifies how the design meets the regulatory requirements of this item.

(1) The plant is designed for continuous purge of 14000 CFM. This is to allow personnel presence within containment during reactor operation.

(2) Continuous containment purge of 14000 cfm, provided in CRBRP, is adequate and meets the ALARA principles for occupational exposures.

The purge line isolation valves are normally open valves located one inside and one outside the containment. Low pressures and low rate of rise of pressures in the case of LMFBRs provide minimal challenge to the isolation function. During accidents, the containment isolation system will reliably isolate the purge (see PSAR Section 6.2.4). The vent line design includes a 10 second time delay between radiation detector and the containment isolation valve. The valves will be periodically tested for leakage during plant shutdowns in accordance with the requirement of Appendix J of 10 CFR 50. valve operability tests will also be performed periodically.

Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979, must be sealed closed as defined in SRP 6.2.4, Item II.3f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable in principle to CRBRP. Containment isolation is required under certain plant accident conditions in order for the containment to perform its safety function. However, the requirements are more specific to the needs of LWRs and do not address the differences required by LMFBR technology. Exceptions to certain detailed requirements of Standard Review Plan (SRP) 6.2.4 are required for CRBRP. The advertised intent of Regulatory Guide 1.141 Rev. 2 is applicable to CRBRP; however, the application of detailed guidelines will be assessed when the Regulatory Guide is issued.

IMPLEMENTATION:

CRBRP design incorporates features that assure the reliability and compatibility of the containment to reduce the radiological consequences and risks to the public from design basis events. Accidents that may occur within containment would develop slowly; unlike LWR's containment pressure and temperature would not rise rapidly. There is no fluid within containment that will experience a phase change thus providing large energy releases in short time periods. Containment pressure is not a consideration in assuring adequate N PSH for safety related pumps. Accordingly, restrictions on normal containment venting (BTP CSB 6-4) are not necessary; containment isolation on high containment pressure would be ineffective in assuring timely execution. Separate and diverse radiation detectors have been determined to provide the best initiation detectors for containment isolation.

Requirements for resetting the Containment Isolation System (CIS), reopening CIS valves, and for containment isolation on radiation signals will be met. However, the requirement of SRP 6.2.4 that pressure be a CIS initiator will not be met because it will not be effective. Containment Isolation Valve operability will be demonstrated for appropriate CRBRP accident loadings.

II.F.1 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

REQUIREMENT:

Applicants shall provide instrumentation to measure, record and readout in the control room: (a) containment pressure, (b) containment water level, (c) containment hydrogen concentration, (d) containment radiation intensity (high level), and (e) noble gas effluents at all potential, accident release points. The requirements for the specific monitors are listed in NUREG-0737. Applicants requirements for the specific monitors are listed in NUREG-0737. Applicants shall also provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential, accident release points, and for onsite capability to analyze and measure these samples. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable in principle to CRBRP. Accident monitoring instrumentation is needed which will provide sufficient information for plant operators to assess conditions during and following an accident and conditions preceding a potential accident. However this requirement is specific to the technology and the needs of LWR's and does not reflect the unique requirements of LMFBR technology; for instance there is no free or uncontained water in the containment and thus there will be no water level in containment to measure.

IMPLEMENTATION:

CRBRP will provide necessary instrumentation to monitor plant variables and systems during and following an accident and for the assessment of conditions which could potentially lead to an accident.

As described in CRBRP Implementation of Item II.F.3 CRBRP has implemented a program for the generation of design bases and criteria for appropriate accident monitoring instrumentation. The set of accident monitoring instrumentation which will subsequently be required will include instruments that meet the principle concern of this requirement; parameters will be chosen giving proper consideration of LMFBR characteristics. Instrumentation will be provided to assess the potential for and actual release of radioactivity from the containment; the containment vessel temperature will be monitored.

CRBRP Implementation of Item II.F.3 will accomplish satisfactory implementation of this requirement.

II.E.5.1 DESIGN EVALUATION

REQUIREMENT:

Applicants with B&W-designed reactors shall (1) identify the most severe overcooling events (considering both anticipated transients and accidents) that could occur at the facilities, (2) show, in view of the arrival rate for these events, that the design criterion for the number of actuation cycles of the emergency core cooling system and reactor protection system is adequate, (3) recommend changes to systems or procedures that would reduce primary system sensitivity. Applicants with B&W-designed reactors shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement would be applicable to CRBRP if the plant used a B&W-designed reactor. However, CRBRP is an LMFBR and not an LWR, and the specific concerns about the consequences of over-cooling events addressed by this requirement are not appropriate to LMFBR designs.

II.F.2 IDENTIFICATION OF AND RECOVERY FROM CONDITIONS LEADING TO INADEQUATE CORE COOLING

REQUIREMENT:

Applicants shall describe their program for developing and implementing procedures to be used by the reactor operators to detect and recover from conditions leading to inadequate core cooling.

Applicants shall provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and incore thermocouples in PWR's and BWR's.

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP. Adequate core cooling is essential to maintaining the plant in a safe condition, and accordingly plant operators must be able to detect and recover from conditions leading to inadequate core cooling.

IMPLEMENTATION:

CRBRP design includes instrumentation which will provide in the control room unambiguous indication of inadequate core cooling. This instrumentation includes reactor coolant level in the reactor, reactor core outlet temperatures and the intermediate heat exchanger outlet temperatures. Discussion of this instrumentation is provided in PSAR Sections 7.4 and 7.5.

II.F.3 INSTRUMENTATION FOR MONITORING ACCIDENT CONDITIONS (REG. GUIDE 1.97)

REQUIREMENT:

Applicants shall provide in their facility design instrumentation to monitor plant variables and systems during and following an accident in accordance with defined design bases and Regulatory Guide 1.97, Rev. 2, December 1980. Designs are already established for much of the instrumentation that will be required; some of the requirements, however, may involve state-of-the-art designs or designs which have yet to be developed.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the contruction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable in principle to CRBRP. Proper provisions for instrumentation to monitor plant variables and systems during and following an accident are necessary for CRBRP as they are for an LWR plant. However, the requirements delineated in Regulatory Guide 1.97, Rev. 2, are specifically for LWR plants and a different set is required for LMFBR's because of fundamental technological differences affecting their design and operation.

IMPLEMENTATION:

CRBRP commits to provide Accident Monitoring Instrumentation in accordance with RG 1.97, Rev. 2, as applied to CRBRP. The design basis for the accident monitoring function will be as follows:

FUNCTION

The functions (consistent with RG 1.97) to be performed by accident monitoring are:

- Provide Primary Information To Permit Manual Actuation of Safety Systems (Type A Variable)

- Indicate That Safety Functions Are Being Accomplished (Type B Variable)

  Safety Functions Are Reactor Shutdown, Reactor Core Cooling, Containment/ Confinement Integrity

- Indicate Potential For Or Breach Of Barriers To Fission Product Release (Type C Variable)

  Barriers Are Fuel Cladding, Primary Coolant Boundary, And Containment

- Indicate Operation Of Safety Systems (Type D Variable)

- Indicate Magnitude Of Release Of Radioactive Materials (Type E Variable)

DESIGN AND QUALIFICATION

The design and qualification requirements for the instruments performing these functions are based on a graded approach. The instruments are categorized (consistent with RG 1.97) and the category depends on the importance to safety of a measured variable. Instrument categories are summarized below:

- CATEGORY 1 - IE Instrumentation, Redundant Instrument Channels, Seismic and Environmental Qualification, Single Failure Criteria, IE Power Source.

- CATEGORY 2 - Environmental Qualification, Highly Reliable Power Source

- CATEGORY 3 - High Quality Commercial Grade Instruments

The qualification environments for Category 1 and Category 2 instruments are based on CRBRP design Basis Events. The Accident Monitoring DBE's, envelope a wide spectrum of environments possible in the CRBRP Design Basis.

Most of the instrumentation required to meet these Accident Monitoring Criteria were part of the CRBRP design prior to issuance of RG 1.97 Rev. 2. In some cases, however, the instrumentation will be upgraded to extend its range and include seismic and single failure criteria appropriate for an accident monitoring key variable.

II.G.1 POWER SUPPLIES FOR PRESSURIZER RELIEF VALVES, BLOCK VALVES, AND LEVEL INDICATION

REQUIREMENT:

Applicants with PWR plants shall provide power supplies for the pressurizer relief valves, block valves, and pressurizer level indicators to meet the applicable requirements specified in NUREG-0737. Level indicators shall be powered from vital buses, motive and control power connections to emergency power sources shall be through devices qualified in accordance with requirements applicable to systems important to safety, and electric power shall be provided from emergency sources. Applicants with PWR plants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the support design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP

This requirement would be applicable to CRBRP if the plant used a PWR which would use a pressurizer. However, CRBRP is an LMFBR which has an unpressurized reactor coolant system and does not feature a pressurizer. Thus CRBRP design does not include pressurizer relief valves, block valves, or pressurizer level indicators and the requirements for power supplies to these equipments are not appropriate to CRBRP.

11.J.3.1 ORGANIZATION AND STAFFING TO OVERSEE DESIGN AND CONSTRUCTION

Applicants shall describe their program for the management oversight of design and construction activities. Specific items to be addressed include: (1) the organizational and management structure which is singularly responsible for the direction of the design and construction of the proposed plant, (2) technical resources which are directed by the utility organization, (3) details of the interaction of design and construction within the utility organization and the manner by which the utility will assure close integration of the architect engineer and nuclear steam supply vendor, (4) proposed procedures of handling the transition to operation, and (5) the degree of top level management oversight and technical control to be exercised by the utility during design and construction, including the preparation and implementation of procedures necessary to guide the effort.

Draft NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources", is the keystone for similar development of guidelines for this task. Therefore, the principal applicable elements of NUREG-0731 shall be used by CP and ML applicants in addressing this task.

Applicants shall submit detailed information in order to provide reasonable assurance that the requirements will be implemented properly to issuance of the construction permits or manufacturing license.

APPLICATION TO CRBRP:

Directly applicable to CRBRP.

IMPLEMENTATION:

Requirement 1

The organizations involved in the design, construction and operation of the CRBRP Project, the organizational structure and arrangement, and the scope of responsibility and authority for each of the participating organizations are described in Section 1.4 of this PSAR. The U. S. Department of Energy (DOE) has overall management responsibility and authority for the Project. The Project has been assigned to the Oak Ridge Operations Office (ORO) for implementation. The organizational unit of the ORO that has been established to execute the overall Project management responsibility and authority is the CRBRP Project Office located in Oak Ridge, Tennessee. That office is under the management responsibility of the CRBRP Project Director. His duties, as established by DOE and in accordance with the Project arrangements, are to direct and coordinate all activities of the CRBRP Project, including overall Project direction, administration, technical guidance, evaluation of

performance of Project participants, and continued assessment of Project quality, cost, and schedule. It is his responsibility to staff and manage the CRBRP Project Office in accomplishing the technical direction and contract administration necessary to carry out the design, construction, testing and operation of the CRBRP and the research and development activities to support this effort.

To assist in the management of the plant design, the Director has appointed an Assistant Director for Engineering and provided him with sufficient personnel, the Engineering Division, to oversee and direct the performance of the contractors responsible for the plant design. In support of the Assistant Director for Engineering's lead management of the design contractors, other divisions of the Project Office are assigned various functional responsibilities. These divisions support the Engineering Division in its lead role in the overall management and coordination of the work of the design contractors. The supporting divisions may also obtain the input of contractors under their direction to assist in the activity.

To assist in the management of the construction of the plant, the Director has also appointed an Assistant Director for Construction and provided him with a staff, the Construction Division, to oversee and direct the performance of the contractor responsible for the construction of the plant. In support of the Assistant Director for Construction's lead management of the construction contractor, other divisions of the Project Office are assigned various functional responsibilities. These divisions support the Construction Division in its lead role in the overall management and coordination of the work of the construction contractor. The supporting divisions may also obtain the input of contractors under their direction to assist in the activity.

Requirement 2

The CRBRP Project Office, serving in the conventional role as the utility, has established prime contracts with the following organizations:

Westinghouse - for the NSSS
Burns and Roe - as the Architect-Engineer
Stone and Webster - as the Constructor
Tennessee Valley Authority - as the Operator

These contracts provide for these organizations to apply their resources and management to the execution of designated functions in support of the design, construction, testing, and operation of the CRBRP. These organizations, as necessary, have established subcontracts to assist them in the execution of their assigned scope of work. The CRBRP Project Director is the DOE contracting officer for these contracts and provides technical direction to these organizations in accordance with the established contracts.

In addition to the resources available under the prime contracts, the CRBRP Project Office has an extensive staff of technical and managerial personnel to carry out the objectives of the Project. At its peak, the CRBRP Project Office is projected to include about 34 DOE employees and 150 employees provided by the utility industry. This organization is described in Section 1.4.2 of this PSAR.

The DOE national laboratories support the CRBRP Project Office in the research and development efforts necessary to support the design and licensing of the plant and are available to provide support in the event that unusual or unique circumstances are encountered during the execution of the Project. These organizations include Oak Ridge National Laboratory, Hanford Engineering Development Laboratory, Argonne National Laboratory, and Energy Technology Engineering Center. If the need should emerge, the Project partners -- Project Management Corporation, Commonwealth Edison, DOE, and TVA -- are available to the Project Office to provide specific expertise of a technical or managerial nature.

Requirement 3

The lead responsibility in the CRBRP Project Office for the management and direction of the Project contractors in the development of the plant design and the manufacture of the plant components has been assigned to the Assistant Director for Engineering. The Project Office Engineering Division functions according to defined procedures in coordinating the Project Office action on all design decisions. This is accomplished by the Engineering Division leading the efforts of the Project Office in defining the design criteria, the design planning requirements, the development requirements, the design control requirements, the interface control activities that are to be accomplished and the design data that are required, for example, the construction specifications, and the technical requirements for operations, maintenance and testing. The Project Office reviews the design submitted by the contractors to determine if the requirements have been met and exercises control through its approval authority. Even though the Engineering Division has lead management responsibility for the design of the plant and manufacture of its equipment, the Construction Division and other divisions of the Project Office concur in the requirements that are developed and participate in the design review and approval process.

The lead responsibility in the CRBRP Project Office for the management and direction of the construction contractor has been assigned to the Assistant Director for Construction. The Project Office Construction Division functions according to defined procedures in coordinating the Project Office actions on all construction decisions. This is accomplished by the Construction Division leading the effort of the Project Office in defining for the Constructor the construction planning requirements, the interface control requirements that are to be exercised during construction, the construction material acquisition requirements, and the technical data that are to be prepared by the

contractors performing the construction work and the installation of plant equipment. The data received from the Contractor in response to these requirements are reviewed and approved by the Construction Division. The Engineering Division and other divisions of the Project Office concur in the requirements that are developed and participate in the approval process to verify that all designer specified requirements and construction specifications are complied with and to properly control the field engineering change process.

To ensure close integration of all design work, the Project Office has required that the plant design be developed in sequential steps of conceptual, preliminary, and final design. During the completion of each step, design organizations are required to review and concur in the interfacing work of other design organizations. Multi-disciplined design reviews are conducted by the designers and observed or participated in by the Project Office. Proposed changes to the design are described in an Engineering Change Proposal which is submitted for review and approval by the involved Project participants. Major changes are reviewed and concurred in by the AE, the Reactor Manufacturers, the Lead Reactor Manufacturer, and if constructibility is involved, by the Constructor, and approved by the Project Office. The CRBRP designers have thoroughly defined the system-to-system interfaces which may consist of physical constraints, process requirements, service demands, etc. These are documented in interface control documents, and any subsequent changes are reviewed and approved by all impacted participants. A computerized system is used to track and assure definition of all system interface requirements.

The CRBRP Project has conducted Key Systems Design Reviews, which considered the plant from an integrated system design standpoint with specific emphasis placed on operations. Tools such as event trees and fault trees were used in conjunction with actual walk-throughs of procedure outlines, maintenance outlines, and casualty events to verify that individual systems have been properly integrated into the total plant that adequate and proper information and control functions are provided to the plant Operator, and to ensure the man-machine interfaces have been appropriately factored into the design. In addition, systems integration groups have been established at the Lead Reactor Manufacturer and the Project Office to maintain a constant review of the design from a plant-wide point of view.

An engineering design model of the plant, constructed and maintained by the AE, is being built in parallel with the development of the plant design. A change to the model accompanies a change to the design. The model plays a major role in identifying and resolving interface problems.

The coordinating activities of the Engineering and Construction Divisions and support functions of the other CRBRP Project Office divisions are carried out in accordance with defined procedures. This results in a disciplined approach to insure that the efforts by the lead divisions in the management of plant design and construction activities include the

involvement of divisions with different functional responsibilities. It also insures close integration of the work performed by the contractors.

Requirement 4

The Project Director has appointed an Assistant Director for Operations to assist him in managing plant operations. The Assistant Director has been provided a staff, the Operations Division, with responsibility for the overall management coordination and direction of the operator in the testing, operation, and maintenance of the plant. In this role, the Operations Division has the lead responsibility for coordinating the Project Office action in the overall management and direction of plant operations.

The viewpoint and experience of the operator is factored into plant design and construction by the participation of the Assistant Director for Operations and his staff in the design management and construction management activities. This has included the review and concurrence in the requirements for design, manufacturing, and construction that have been defined and issued by the Project Office. It has also included the review and concurrence in the design output documentation requiring review and approval of the Project Office. The Project Office Operations Division is responsible for obtaining input from the Operator on plant design matters that the Operations Division considers important to plant operation. This input is obtainted by requesting the Operator to review and comment on selected plant design features and participate in design reviews.

A contract between DOE and TVA establishes TVA's responsibilities for testing, startup, operation, maintenance, and demonstration activities of the Project. To insure continuity from design/construction to operation and to insure proficiency of operator personnel, the Operator will begin assigning operation personnel to the site about five years before criticality. The first assignment will consist of about ten people, including the Plant Manager and his top supervisory staff. From this initial assignment, the plant staff will be increased to a peak of some 380 people at criticality and will then level off at about 340 people two years after criticality.

The approximate plant staff size and activities of the Operator are summarized as follows: From the time of initial assignment onsite until start of preoperational testing, three years later, the staff size will increase from about ten to 170. Principal activities during this period consist of observation training at an operating LMFBR; CRBRP specific plant and systems training; observation of the installation of equipment and systems; preparation of detailed plant procedures, including administrative, operating, maintenance, surveillance test, and preoperational test procedures.

From the time preoperational tests are started about two years before criticality until fuel loading, the staff size will increase to about 380. Principal activities during this period include finalizing the detailed

procedure writing activity; acceptance of completed systems from the Constructor; preoperational testing; operation, maintenance, and surveillance testing of turned-over systems; and continuation of training, including simulator training; responsibility for plant security; and preparation for, and obtaining NRC licenses for the operators.

From fuel loading until about two years later, the staff size will decrease from 380 to about 340. Principal duties during this period include startup testing, plant opeartion, maintenance, surveillance testing, and modification work.

Physical control of plant structures, systems, and components is established during the manufacturing of plant items and is maintained throughout delivery, storage, and installation. The Operator will be on site, staffed, trained, and qualified to take custody of the plant components at turnover from the Constructor. The physical operation of all components and systems for testing purposes, and subsequent startup, operation, and maintenance will be accomplished by personnel of the Operator. As plant equipment and systems progress through design and manufacturing, the Operator will, as requested, provide advice and support to the organization responsible for design, manufacturing, and construction. As the equipment progresses through the construction phase and approaches the turnover point, the Operator will play a more central role in the planning and managing of plant facilities. After the transfer of custody is accomplished, the Operator will be in full custody of the equipment and will perform the normal operation and maintenance required of those components. At this point, the engineering and construction organizations will become supporting of the Operator.

The CRBRP Project Office will continue to perform its overall Project management responsibilities throughout this phase-to-phase transition of the Project. As the plant is completed and transfer of custody to the Operator is accomplished, the resources of the Project Office will be available to support the Project Office Operations Division in the management and direction of the operating program. The resources of the Project contractors and other participating organizations in the design and construction of the plant will be available for any necessary support of the Operator with regard to plant testing, operation, maintenance and modification.

Requirement 5

An underlying principle of the government/industry partnership for the design, construction, and operation of the CRBRP is that the senior management of the partners will be brought to the Project to assist in the effective execution of the demonstration Project. To this end, the partner organizations are committed to supply to the CRBRP Project Office experienced and proficient technical and managerial personnel. Even though the partner senior management personnel are involved in the support of the Project and its execution, the organization with direct responsibility for the Project is the CRBRP Project Office. The Project Director is

directly involved, full-time, in the day-to-day management and direction of the Project activities. His staff, as identified in Section 1.4.2.1 of this PSAR, is committed fulltime to support the Director in the management and direction of the Project and will be involved throughout the design, construction, and operation phases of the demonstration plant project.

To integrate and coordinate the management and direction of the Project, the Project Office has developed and issued a set of policies and requirements which defines the management systems to be implemented and the roles of the principal contractors and support organizations. These management policies and requirements provide the basic documentation for the management systems necessary to assure that the plant is designed, constructed, and operated in accordance with Project requirements. Each participating organization has been required to develop a set of internal procedures by which it will execute assigned responsibilities, identify where its internal activities interface with other Project participants, and define how its activities will be accomplished.

The CRBRP plant and systems technical requirements were developed under the CRBRP Management Control Systems and were approved by the Project Office. Changes to these requirements are conducted under change control procedures and require approval by the Project Office.

To ensure that the Project Office has the visibility required to control the technical design, a computerized Project Status and Control System has been developed to provide data indicating the technical progress of the design.

To provide additional technical oversight, the Project Office conducts formal design reviews. These serve as a customer check on the design before the release of major system milestones, for example, preliminary design, final design, fabrication, or construction.

To ensure Project Office design and construction control, construction drawings and documents are reviewed and approved by designated Project Office personnel.

The CRBRP Project Quality Assurance Program described in Section 17 of the PSAR provides the management control systems to assure that facilities, systems, components, and equipment are designed and constructed to conform to specified requirements.

The CRBRP Project Office has a set of internal procedures that define and guide its own actions in the management and direction of the Project. These procedures have been prepared and issued at the direction of the CRBRP Project Director and are implemented by the various divisions of the Project Office. These procedures carefully define the roles of the divisions in their lead or support functions as they integrate and contribute to the overall management and direction of the Project. It is through this family of procedures that the Project Office coordinates its own activities and ensures that the overall management of the Project will be accomplished and transition from one phase to another will be effected in a satisfactory manner.

II.K MEASURES TO MITIGATE SMALL-BREAK LOCAs AND LOSS OF FEEDWATER ACCIDENTS

REQUIREMENTS:

A set of items resulting from requirements generated in IE Bulletins, Commission Orders on Babcock and Wilcox Plants, and the Final Recommendations of B&O Task Force, delineate requirements specific to particular designs of Light Water Reactors.

APPLICATION TO CRBRP:

This set of requirements addresses concerns specific and unique to particular LWR designs. Features of designs and engineering principles addressed by these items are not incorporated into CRBRP. Thus no action is required.

III.A.1.2 UPGRADE LICENSEE EMERGENCY SUPPORT FACILITIES

REQUIREMENT:

Applicants shall address the requirements for a Technical Support Center, Operational Support Center and the Emergency Operations Facility. Applicants shall provide preliminary design information in accordance with the functional criteria in NUREG-0696 at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable to CRBRP. Emergency planning and appropriate emergency support facilities are necessary for CRBRP like any other power reactor.

IMPLEMENTATION:

CRBRP will provide on-site Emergency Response Facilities in accordance with guidance provided in NUREG-0696 and will utilize the same off-site emergency planning, organization and facilities that have already been established by the plant operator, TVA.

The on-site emergency response facilities include the Technical Support Center and the Operational Support Center

Technical Support Center (TSC)

The TSC will be designed and equipped to perform the following functions:

- Monitor plant operations and provide technical advice and overall plant management from a non-control room location.
- Supply sufficient plant information for analysis of plant status and extent of any plant damage.
- Record sufficient information to allow for a thorough review of plant incidents and to aid in plant recovery.
- Provide communications with off-site personnel and provide for other peripheral functions not directly related to reactor operations.
- Relieve control room congestion.

Onsite Operational Support Center (OSC)

An Onsite Operational Support Center will be provided to perform the following functions:

- Provide a location where support personnel can assemble and support can be coordinated.
- Provide communication with the Control Room and the TSC.

III.D.1.1 PRIMARY COOLANT SOURCES OUTSIDE THE CONTAINMENT STRUCTURE

REQUIREMENT:

Applicants shall review the designs of such systems outside containment, and their provisions for leakage control and detection, overpressurization design, discharge points for waste gas venting systems, etc., with the goal of minimizing potential exposures to workers and public following an accident, and providing reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. Applicants shall provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident, and submit a leakage control program, including an initial test program and a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems.

In this regard, applicants shall submit, prior to the issuance of construction permits, a general discussion of their approach to minimizing leakage from such systems outside containment, in sufficient detail to provide reasonable assurance that this objective will be met satisfactorily prior to issuance of operating licenses.

APPLICATION TO CRBRP:

This requirement is applicable in principle to CRBRP. A primary design and operational goal is to minimize potential exposures to workers and public following an accident. However, the detailed requirements are specific to LWR's and do not reflect the unique technology of LMFBR's.

IMPLEMENTATION:

CRBRP is committed to accomplish, through design features and operational measures, the goal of minimizing potential exposures to workers and public following an accident. CRBRP reactor coolant is contained by the Primary Heat Transport System (PHTS) and in the overflow tank. The PHTS and the overflow tank, and all associated piping containing reactor coolant is located entirely within the containment. CRBRP has no operating fluid systems connected to the PHTS which are located outside of the containment and thus there is no plan for reactor coolant to be outside containment, with the exception of transfers of reactor coolant samples for processing and analysis. One line penetrates containment and is connected directly to the PHTS, for use during the initial sodium fill of the system. However, this line is normally valved closed within containment and does not provide a path for reactor coolant outside of containment except under specifically planned and closely controlled shutdown maintenance operations. Thus CRBRP meets this requirement through maintaining the reactor coolant entirely within containment.

III.D.3.3 IN-PLANT RADIATION MONITORING

REQUIREMENT:

Applicants shall review their designs to assure that provisions for monitoring inplant radiation and airborne radioactivity are appropriate for a broad range of routine and emergency conditions. Applicants shall, to the extent possible provide preliminary design information at a level consistent with that normally required at the cosntruction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

The requirement is applicable in principle.

Of concern in LWRs, is the release of radioidine and adequacy of sampling instrumentation following an accident. In LMFBRs, the potential for release of radioiodine is minimized due to its affinity to sodium. However, the general requirements for the capability to monitor inplant radiation for a broad range of routine and emergency conditions is equally applicable to LMFBRs. This is to provide an appropriate level of protection to the plant workers.

IMPLEMENTATION:

The requirements have been reviewed and identified in conjunction with implementation of R.G. 1.97, Rev. 2 in the CRBRP design. The inplant monitoring system has been designed to the latest state of the art and incorporates provisions for monitoring radiation and airborne radioactivity over a broad range of routine and emergency condition.

The complement of portable radiation monitors and samplers and the on-site analysis and calibration capability will be the same as found acceptable by NRC for other nuclear power plants operated by TVA.

III.D.3.4 CONTROL ROOM HABITABILITY

REQUIREMENT:

Applicants shall review the design of their facilities for conformance to requirements stated in the Action Plan. NRC will consider possible new criteria to preclude control room contamination via potential internal pathways indicated by the TMI-2 experience.

Applicants shall address prior to the issuance of the construction permits or manufacturing license, how they will implement the existing requirements set forth in this Action Plan item. Applicants shall also address the extent to which improvements have been made to prevent control room contamination via pathways not previously considered. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exiss reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

APPLICATION TO CRBRP:

The requirements is directly applicable to CRBRP. This requirement is fully applicable as the control room habitability requirements are the same as those for LWRs. An additional habitability concern in a LMFBR is the release of sodium aerosols. The control room design addresses these concerns through appropriate design of auxiliary systems (H&V system) serving the control room.

IMPLEMENTATION:

Compliance of the CRBRP control room habitability design with regulatory criteria is described in PSAR Section 6.3. Radiological analysis assuming release of the Site Suitability Source Term has been completed and included in PSAR Section 6.3. Furthermore, radiological analyses for Hypothetical Core Melt are provided in a Topical Report CRBRP-3 Volume 2, "Hypothetical Core Disruptive Accident Considerations in CRBRP."

CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

APPENDIX J

PRA PROGRAM PLAN

PROJECT MANAGEMENT CORPORATION

TABLE OF CONTENTS

	PAGE
1.0 INTRODUCTION	1
2.0 OVERVIEW OF THE RISK ASSESSMENT PROGRAM	1
2.1 Accident Initiator Development	1
2.2 Plant Logic Model Development and Quantification	3
2.2.1 System Functional Event Tree Development	3
2.2.2 System Functional Fault Tree Development	4
2.2.3 Analyses of Plant Response	5
2.2.4 Accident Sequence Quantification	6
2.2.5 Uncertainty Analyses	7
2.2.6 Common Cause Failure Analyses (CCFA)	8
2.2.6.1 Explicit Modeling of Dependencies	8
2.2.6.2 Qualitative CCFA	8
2.2.6.3 Detailed CCFA	9
2.2.6.4 Special CCFA Investigations	10
2.3 Core and Containment Accident Modeling	11
2.3.1 Phenomenological Event Trees	11
2.3.2 Source Term Evaluation	12
2.4 Health Consequence Analysis	13

Amend. 73
Nov. 1982

TABLE OF CONTENTS (Continued)

	PAGE
2.5 Risk Analysis	13
2.6 PRA APPLICATIONS TASKS	13
2.6.1 Operator Action Event Trees	14
2.6.2 Assessment Of The Effectiveness Of Postulated Design Variations Including Consequence Mitigation Features	14
2.6.3 Improve Understanding of the Plant	14
2.6.4 Characterization Of Risk From Early Life Failures	15
2.6.5 Implementation Of A Continuing Risk Management Program	15
2.6.6 Input To The Site Emergency Procedures	16
2.7 Interaction With the NRC	16
2.8 Accident Delineation	17
2.9 Study Limitations	18
3.0 PRA PERFORMANCE AND REVIEW	19
4.0 SCHEDULE, MILESTONES, AND RESOURCE ALLOCATION	20
TABLE 1 CRBRP PRA PRODUCTS	21
TABLE 2 RESOURCE ALLOCATION	22
TABLE 3 CURRENT SCHEDULE OF UNCOMPLETED TASKS	23
FIGURE 1 PRA FLOWCHART	24
FIGURE 2 CRBRP PRA ORGANIZATION	25
FIGURE 3 ANTICIPATED PRA INTERNAL REVIEW PLAN	26
FIGURE 4 EXPECTED SCHEDULE OF PRA PRODUCTS	27

PROGRAM PLAN FOR THE CLINCH RIVER BREEDER REACTOR PLANT

PROBABILISTIC RISK ASSESSMENT (PRA)

1.0 INTRODUCTION

This Program Plan describes the plan for the CRBRP PRA and applications of the study. The PRA was initiated principally related to the desire of the project to perform an integrated safety assessment as one ingredient in the decision process leading to safe design and operation. The PRA will also satisfy the requirements of NUREG-0718, Section II.B.8 and it is consistent with the current direction of the NRC in development of safety goals and PRA applications. As details are developed, they will be incorporated into this Program Plan. This review process is discussed in Sections 2.6 and 3.0. The PRA will be used as an aid in evaluation of the current design and alternative designs and as a tool to provide further assurance of safe plant operation.

The PRA will be, in the terminology of the PRA Procedures Guide (NUREG/CR-2300, Rev. 1), a Level III PRA. In addition to the tasks which comprise a Level III PRA, several tasks will support application of the study. These tasks include use of the PRA models to support operational programs such as emergency procedure preparation and operator training. The purpose of using the PRA in this way is that it can serve as a basis for defining potential operational incidents, thereby helping to reduce uncertainties caused by the limited availability of LMFBR operating experience.

2.0 OVERVIEW OF THE RISK ASSESSMENT PROGRAM

The PRA can be divided into the following major elements: accident initiator development, plant logic models for system functional event trees, and system functional fault trees, phenomenological event trees, and release and consequences analysis. The phenomenological event trees are divided into two groups; those that describe the phenomena from core melt to breach of the reactor vessel (i.e., core damage phenomenological event trees) and those that describe the phenomena from breach of the reactor vessel to containment integrity failure (i.e., containment phenomenological event trees). Refer to Figure 1 for a flowchart depicting how major elements will be tied together. These elements will be discussed in the following subsections.

2.1 ACCIDENT INITIATOR DEVELOPMENT

The approach being taken to logic model construction emphasizes the investigative nature of the task and results in an iterative model building process which ensures the accuracy of the final logic models. The following describes the iterative investigative method to be used.

A preliminary list of initiating events will be developed by extracting information from a variety of relevant sources. These sources include:

- Compilations of Generic Experience: Examples are NUREG-0460 and EPRI NP-2230. In addition, NSAC has produced a screened list of LERs which identifies a number of risk significant PWR initiating events.

- Previous PRAs: A number of other PRAs have been completed or are on-going. Each of these PRAs has compiled a list of initiating events (often from the generic sources listed above).

- CRBRP Project Documentation: A number of design-specific documents are being screened to identify potential initiating events. These include the PSAR, plant design descriptions, Key Systems Review, Availability Analyses, and the existing Reactor Shutdown System (RSS) and the Shutdown Heat Removal System (SHRS) reliability assessments, (see PSAR Appendix C).

- Breeder Reactor Experience: Including foreign and domestic sodium and/or breeder experience.

The resultant list of initiating events will allow the event tree and fault tree analyses to commence, but is not considered the final list. It is important that information gained during the event tree/fault tree analyses be continuously fed back into the task of identifying initiating events. By definition, an important initiating event is one that can evolve into an important sequence. It is impossible, therefore, to confidently list all the important initators before the event tree and fault tree analyses have been performed. This process systematically utilizes knowledge gained in the event tree/fault tree analyses to ensure that all important initiating events are identified. This approach is based upon the recognition that: (1) important initiators are either relatively high frequency events or are events which adversely impact the ability of the safety systems to respond, and (2) an initiating event must, by definition, require an active plant response to avoid core damage. The cut-sets of the fault tree models will be systematically examined for their relationship to event tree headings (developed for the preliminary list of initiators). It is then possible to identify any failure events which both call for an active plant response and adversely impact the performance of the safety systems. These types of events will be considered as potential initiating events.

The initiating event development requires performance of a fault tree analyses of the initiating event where appropriate. By comparing the cut-sets of the initiator with those of safety systems required to respond to the initiator it is possible to ascertain whether the specific cause of the initiator could also impact the ability of safety systems to respond. This additional step in initiating event identification ensures the accurate quantification of conditional probabilities and allows an initiating event to be broken down into subevents to highlight potential dependencies between the initiator and subsequent events. As an example of this step, the "loss of offsite power" event would be identified in the cut-sets of a "loss of feedwater" initiator and also show up as an element in the cut-sets of the Shutdown Heat Removal Systems; accordingly, "loss of offsite power" is always identified as a separate initiating event.

Thus, the proposed approach is an iterative process of initiating event identification which starts with the application of available compilations of operating experience and design information and feeds back crucial information from the ensuing event tree/fault tree analysis. In this way, completeness is assured not only by searching available compilations of data but also by explicitly and systematically investigating the CRBRP plant design.

2.2 PLANT LOGIC MODEL DEVELOPMENT AND QUANTIFICATION

Considerable attention will be devoted to the task of constructing accurate logic models. The design of the CRBRP which is being analyzed was that in place as of February 1, 1982. The analysis will be updated to reflect the current design at several stages during the PRA; specifically, any changes derived from the Construction Permit licensing review will be included in the final models.

The specific activities included under the general heading of Plant Model Development and Quantification are the development of system functional event tree and fault tree logic models, the analyses of plant response, accident sequence quantification, uncertainty analyses, and common cause failure analyses. The result of the plant model development and quantification tasks is a compilation of those probabilistically quantified accident sequences which lead to core damage and their particular plant state. These plant states are then the entry point to the analysis of the severity of damage resulting from an uncoolable core within the primary coolant boundary (i.e., core damage phenomenological event trees) and to the severity of challenge to containment integrity associated with the accident sequences which breach the primary coolant boundary (i.e., containment phenomenological event trees). Damage analyses will be discussed in subsequent subsections.

2.2.1 System Functional Event Tree Development

System functional event trees (ET) will be constructed for the initiating events discussed in Section 2.1. Before actual event tree construction begins, the numerous individual initiating events will be grouped into preliminary categories based upon their impact on the plant and the subsequent demands upon the plant safety systems. The approach to be used to perform the event tree analyses following preliminary categorization of initiating events is as follows:

1. Determine the functional requirements which must be met in response to the initiating event. Examples of such functions are reactor shutdown and decay heat removal.

2. Define the plant systems available to perform each of the necessary functions.

3. List all of the supporting systems which are common to the plant systems identified in Step 3. Examples of supporting systems are service water, instrument air, and electrical power. This information will assist in the identification of potential dependencies between systems and will affect the identification and ordering of event tree headings.

4. Identify operator actions associated with the systems identified in Steps 2 and 3.

5. Define potential functional dependencies between the plant systems identified in Step 2. This is the first step in an iterative process of identifying important system interrelationships which are not apparent by merely listing common hardware.

6. Perform the necessary analyses to determine timing of events, systems requirements, and the corresponding failure state definitions.

7. Construct the system level event trees. The two elements of constructing system event trees are determining: (1) the definition of the individual event headings and (2) the ordering of the events to produce the logic model for the event tree.

8. Document the event trees. The assumptions and reasoning which produced the event tree form are carefully documented.

Although the above outline describes a step-by-step process, the event tree construction process will be an iterative one in which the failure state definitions, timing, and system headings are continuously influenced by information fed back from the fault tree development and the plant response analyses.

2.2.2 System Functional Fault Tree Development

Fault trees will be drawn for most of the event tree headings. Decisions concerning the necessity to develop individual fault trees will be based upon the recognition that the purpose of a fault tree is to: (1) quantify the probability of an event for which no statistically acceptable data exist by logically breaking down the event into its constituent parts for which acceptable data do exist, and/or (2) identify potential dependencies among multiple systems. Fault trees will not be drawn for systems for which: (1) acceptable data exist for the event heading and no significant dependencies could exist between this event and subsequent headings, or (2) the event heading could not be involved in any risk-important accident sequences even if its conditional failure probability were extremely high.

The fault tree (FT) analyses will be performed using procedures and symbols presented in NRC's Fault Tree Handbook (NUREG-0492) or in the PRA Procedures Guide (NUREG/CR-2300, Rev. 1).

A package of FTs will be provided for:

- Shutdown Heat Removal System Top Logic
- Primary and Intermediate Heat Transport Systems
- Steam Generator System
- Main Feedwater and Condensate
- Turbine Bypass Valves and Condenser
- Steam Generator Auxiliary Heat Removal System
- Auxiliary Feedwater
- Direct Heat Removal Service
- Normal & Emergency Chilled Water

- Plant Service Water
- Class 1E Electrical
- Containment Cleanup
- Annulus Filtration
- Annulus Air Cooling
- Compressed Gas
- Containment Isolation

Each FT will include all known support systems such as electric power, instrument and control, instrument air, and service water.

In addition, a fault tree data base will be produced which will allow quantification of all fault trees. The data base will be derived from the same sources as those identified for accident initiator development.

2.2.3 Analyses of Plant Response

Analyses of realistic plant responses to postulated accidents will be performed throughout the course of the PRA to ensure that the plant logic models for ETs and FTs represent an accurate picture of the plant response and that all dominant risk contributors have been identified. Analyses will also be performed to assure that the system success criteria (i.e., successful system function) is realistically based on the physical capabilities of the plant.

A key element of the approach to this task is the efficient, systematic identification of specific analytical needs. Due to the costs and time delays in obtaining best-estimate plant response data, it is crucial that the analyst be able to determine: (1) what analyses are necessary for the PRA, and (2) what are the specific inputs and desired outputs of the analyses. Event trees will be systematically used to make these determinations. At each branch point in the event tree, the analyst will ask:

- What is required of the plant systems to maintain the necessary functions?
- What are the existing plant conditions important to maintenance of these functions?
- What are the realistic capabilities of the systems under these conditions?

These questions will determine the basic analytical requirements. When answers to these questions are not readily available to the PRA analyst, one of three avenues will be pursued to supply the needed information:

(1) Locate applicable analyses in available documentation.

(2) Perform hand calculations or extrapolation of existing analyses. Analytical needs not satisfied by available documentation can be satisfied by hand calculations where appropriate.

(3) Perform additional computer analyses. This avenue will be used only when documented analyses are not available and hand calculations would be insufficient. The role of the risk analyst at this point is to ensure that the results of the PRA are truly sensitive to the results of the desired analyses and to define the analytical requirements as carefully as possible.

The plant logic models will be modified where incorporation of best-estimate calculations will more accurately model actual plant behavior under accident conditions.

The heat transport and heat sink systems will be characterized in terms of their heat removal capabilities. Best-estimate calculations will provide the heat removal rate for one-, two-, and three-loop operation of the heat transport system under forced and natural circulation conditions. The characterization will also assess the amount of heat transported to the steam drum in excess of the capability of the closed-cycle sinks (i.e., Protected Air Cooling Condensers). In regard to the heat sinks, the capabilities of the PACCs will also be characterized for forced and natural draft and capabilities. The Steam Generator Auxiliary Heat Removal System vents will be characterized for operation with different Auxiliary Feedwater pumps and vents. Characteristic values of the heat transport and heat sink systems will allow determination whether various Loss of Heat Sink sequences will satisfy the success criterion.

The success criterion (prevention of core damage) for the loss-of-heat-sink event will be formulated and the analysis supporting the rationale leading to the criterion will be provided. The analyses will be best-estimate and the criterion will be simply-stated but will be broadly applicable to accident sequences leading to a loss-of-heat-sink.

The core damage event tree will trace through the phenomenological steps from the initial conditions in the core to the release of core material from the reactor vessel, if any. Event trees will be developed for all accidents identified by the event trees as leading to core damage. The plant states identified in the event trees will be carried through the core damage event tree to the containment event trees. Quantification at each node will be derived from an understanding of the basic phenomenological processes involved in the event.

2.2.4 Accident Sequence Quantification

Using the initiating events, ETs, FTs, and associated data bases, the dominant accident sequences will be quantified. The products of this task will include:

- A description of the process used to "link" the fault trees together and to ensure that all dependencies among system fault trees are identified and incorporated into sequence quantification.

- o A listing of the dominant accident sequences and a description of each including the individual events comprising each sequence.

- o A systematic justification for omitting any ET sequence from the dominant list (e.g., sequence Z, while producing the same effects, has an occurrence frequency four orders of magnitude less than sequence A).

The computer code which will be used in the generation of cut-sets and accident sequence quantification is COMCAN III (COMmon Cause ANalysis) developed by Idaho National Engineering Laboratory (INEL). Quantification will be an iterative process in which early analyses are used to help focus the more detailed common cause failure analyses, (see Section 2.2.6).

2.2.5 Uncertainty Analyses

Early in the PRA sensitivity studies will be utilized to provide information on the relative importance of equipment and human failures. Detailed uncertainty analyses will be delayed until later in the PRA program.

After the best-estimate quantification, a detailed uncertainty analysis will be performed in order to establish uncertainty bounds in the overall results of the PRA study. Estimates will be made of the probability distributions or confidence limits for those component failure rates and event frequencies which are potentially important to risk. The limitations of the uncertainty analyses will be addressed. The assessment methodology will include evaluation of the uncertainties on the importance of accident sequences which are known to have large uncertainties. The work will also establish confidence limits for total core damage frequency and a cumulative probability distribution curve. Probability distributions or confidence limits will be estimated only for those component failure rates and event frequencies which are potentially important to risk. For sequences, such as seismic, which are known to have large uncertainties, the assessment methodology will include evaluation of the effects of these uncertainties on the importance of the accident sequences. Any uncertainties which cannot be quantified will be qualitatively discussed.

Sensitivity analyses will be performed using the plant logic models to identify those input data or assumptions which significantly affect the dominant accident sequence list. These sensitivity analyses will focus on (1) uncertain failure rates, or (2) uncertain assumptions concerning success criteria or system dependencies. The final product of this task will include:

- o A description of each sensitivity analysis, including motivation,

- o The results of each analysis,

- o Interpretation of each analysis including the impact on detailed common cause failure analyses.

2.2.6 Common Cause Failure Analyses (CCFA)

The common cause failure analyses are broken down into four subtasks. These include: explicit modeling of dependencies, qualitative CCFA, detailed CCFA, and special CCFA investigations. These subtasks are delineated in the following four sections.

The CCFA is an assessment of common failure susceptibility and opportunity, including quantification of common cause events for:

a. internal events (such as temperature and pressure extremes and transients, common locations, proximity to degrading influences, fires); and

b. external events (such as seismic events, tornados, floods, lightning, chemical, radiation, explosions and aircraft or missile impacts).

2.2.6.1 Explicit Modeling of Dependencies

This portion of the CCFA entails those efforts required to ensure that all common support systems and functional dependencies between and among plant systems are explicitly and accurately included in the plant logic models. This task will be carried out in the process of constructing the event and fault trees.

All known inter- and intra-system dependencies will be modelled (such as common support systems identified in the fault trees), and performance of preliminary analyses of functional dependencies in which the physical response of the plant to the failure of one system has an adverse affect on the ability of another system to operate, and potential operator errors which can disable multiple systems or provide the link between failure of one system and another. The specific products of this task will include:

- A description of all known inter-system dependencies resulting from common support or interfacing systems.

- A "system level" failure modes and effects analysis (FMEA) in which the physical plant response to all event tree failure modes is described in terms of the behavior of major parameters (e.g., failure of system A to start will result in a rapidly rising pressure in system B).

- A listing for each event tree of the potential impacts of major parametric changes (e.g., a rapidly rising pressure will trip off pump C and cause failure of system B).

- A description of the more detailed dependency analyses to be performed and a preliminary approach for performing these analyses. An approach to treating external events such as seismic events will be included.

2.2.6.2 Qualitative CCFA

There may be failure causes common to multiple components which fall below the

practical level of resolution in the event trees and fault trees. Examples of such potential common failure causes are:

- o manufacturing, installation, or maintenance errors
- o adverse environmental influences such as high temperature, humidity, or radiation
- o corrosion, carburization, rust, or other chemical degradation processes.

In this subtask, these types of common cause failures which could potentially have a significant impact on plant risk will be identified. This will be achieved by a conservative filtering process. This filtering process will allow the subsequent, more detailed analyses decribed below to focus on those dependencies which could actually be important to risk. This filtering process is based upon the recognition that for a common cause failure to occur, two criteria must be met:

(1) Both components must be susceptible to failure by the common cause (e.g., two different valve operators might both be susceptible to failure due to flooding, but a pipe and valve do not share a common susceptibility to failure by flooding).

(2) The common cause must have the opportunity to affect both components.

Based on this recognition, a two-step qualitative CCFA will be performed. The first step involves determination of which redundant components are susceptible to failure by the common cause. The second step involves determination of which susceptible redundant components (identified in the first step) are located such that an event could subject them to the common cause. Any susceptible redundant components so located are considered common cause candidates.

2.2.6.3 Detailed CCFA

The input to this subtask will be the relatively small number of common cause candidates which survive the screening process discussed above. In this subtask a more detailed assessment of common failure susceptibility and opportunity will be performed and probabilities estimated for these common cause events.

The detailed CCFA will address an extensive list of potential common failure causes (e.g., vibration, high temperature, etc.) for each component and will determine the potential for these mechanisms coincidentally affecting the components. For redundant components in different locations, this will entail evaluating the likelihood that causes can be coincidentally present in both locations. For components in the same location, this will entail a determination of whether the components are both (or all) susceptible to the same causes and the likelihood of those causes existing in that particular location.

2.2.6.4 Special CCFA Investigations

The above three subtasks will allow a practical, effective CCFA to be performed for most potential failure mechanisms. However, there are additional potential causes for multiple failures which will be addressed separately. These are fires, seismic events, and other significant external events.

Fires

The common location analysis performed (see Section 2.2.6.2) will form the basis for the fire analysis. The location analysis will provide:

- o List of key locations with potential for exposure to combustibles, an oxidizer, and an ignition source.
- o Key components in these locations
- o Fire related failure modes of these key components.

Based on this information, a preliminary scoping fire analysis will be performed to determine if fire-related accident sequences could contribute significantly to risk at the CRBRP.

Should the above scoping analysis identify any single or double location cut-sets which could realistically support a fire of sufficient size and duration to fail the components associated with these cut-sets, a more detailed fire analysis will be performed for these specific locations. Thus, the scoping fire analysis will be used to focus any detailed fire analyses which are required on those particular fire-related sequences which could potentially contribute to risk.

Seismic and Other External Events

External events such as seismic events, tornados, floods, explosions, aircraft or missile impacts, etc. will be evaluated to ascertain their significance to risk. Also, a detailed seismic methodology will be developed.

The preliminary analysis will be comprised of six basic steps:

(1) Estimate the frequency of occurrence of each external event.

(2) Identify the specific components or systems which could be adversely impacted by the event.

(3) Calculate the failure probability of such equipment and recalculate the probability of core damage with these components or systems unavailable.

(4) Multiply the frequency of occurrence from (1) by the conditional probability of core damage from (3).

(5) Compare the results of (4) to the baseline core damage frequency.

(6) If the frequency comparison in (4) indicates the sequence is risk significant, consider phenomenological aspects of the sequences.

If the results of the preliminary investigation indicate that there are potentially risk significant sequences initiated by one of the external events a more detailed analysis will be performed. The approach to this more detailed analysis will be very similar to that outlined above for the preliminary analysis. However, conservatisms in the preliminary investigation will be replaced by realistic evaluations of the impacts of the initiating event on plant systems. The overall risk shall include the contribution from external events.

2.3 CORE AND CONTAINMENT ACCIDENT MODELING

The result of the tasks on Plant Logic Model Development and Quantification will be a set of probabilistically quantified dominant accident sequences each of which is expected to produce damage to the core. Sequences which do not lead to core damage will be identified. Associated with each accident sequence leading to core damage will be a plant state which will include:

1. An indication of the successful operation of the Plant Protection System.

2. An indication of the availability of mitigating systems, (i.e., Containment Isolation, Annulus Air Cooling, etc.).

3. An indication of the capability of structures and surfaces to act as static and convective heat sinks during the accident sequences.

The definition of plant state for accident sequences terminated by core damage will allow two evaluations to be performed. First, the potential for various degrees of mechanical damage to the primary system resulting from energetic disassembly of the core can be evaluated. Second, the potential for failure of the containment system to maintain its integrity following a variety of severe accident sequences can be evaluated.

2.3.1 Phenomenological Event Trees

Phenomenological event trees will be prepared for both core damage and containment behavior resulting from accident sequences that lead to core damage (i.e. core damage phenomenological event trees and containment phenomenological event trees). The accident sequences will include those which could potentially lead to core energetics as well as those which have no significant energetics associated with the core disruption. The combined core damage and containment event trees will sequentially start with a definition of the plant state and sequentially terminate with a description of either a stable coolable state for the core debris or the time and size of the containment failure. As part of this evaluation, the radioactive source term above the operating floor at the time of a stable end point or containment failure will be defined. The event trees will describe, in detail, the major physical processes occurring within the primary system and containment which precede, cause, and follow, hydrodynamic core disassembly and/or loss of core coolability. This will include consideration of the thermal margins provided by the CRBRP design to mitigate the consequences of core damage as well as the structural margins to mitigate energetic effects and minimize a direct release of sodium and radionuclides from the primary system through the reactor vessel head. Both the core damage and containment event trees will be quantified.

The bases for selecting probabilities for each node will be documented. Development of the phenomenological event trees will include analyses, as follows:

- o Thermal-hydraulics evaluation of the loss of decay heat removal following reactor shutdown including thermodynamic and heat transfer evaluation of the primary heat transport system.

- o Extrapolation of currently available CACECO analyses to apply to the loss of decay heat removal following reactor shutdown.

- o Structural calculations to assess the structural integrity of systems and components where necessary to support the phenomenological event trees.

The products of these analyses will include definition and probabilistic quantification of the range of potential sequences by which large quantities of radionuclides might be released from containment following a variety of accident sequences which produce core damage. These containment phenomenological sequences will define the conditions under which detailed analyses of the radionuclide source term from containment will be completed.

The core damage event tree will trace through the phenomenological steps from the initial conditions in the core to the release of core material from the reactor vessel, if any. Event trees will be developed for all accidents identified by the event trees as leading to core damage. The plant states identified in the event trees will be carried through the core damage event tree to the containment event trees. Quantification at each node will be derived from an understanding of the basic phenomenological processes involved in the event.

These phenomenological processes and the considerations leading to the selection of the quantification will be documented. The resulting core damage states will be grouped on the basis of energetics (ability of structurally loading the primary system) and on the basis of the degree of core melting (ability of thermally loading the primary system).

The containment event tree will combine the phenomenology of the molten core and sodium reacting with the environment outside the reactor vessel and the response of the containment/confinement system, beginning with the plant states previously identified and a small number of core damage states. The quantification of the phenomenological branch will be based on the understanding of the processes, the results from existing analysis and new supporting calculations where necessary. Quantification of the availability of containment system responses will be provided in the plant states.

2.3.2 Source Term Evaluation

An analysis will be performed to define the environmental source term for each of the unique paths through the containment phenomenological event trees for which significant releases of radionuclides are expected. Existing computer codes which will be utilized in this analyses include CACECO, HAA-3, and COMRADEX.

The potential for release and related health effects from ex-core sources of radionuclides will be defined. These sources will be evaluated using fault tree analysis techniques and the appropriate source terms given various plant system responses. Ex-core sources include radioactive cover gas, ex-vessel spent fuel storage, and other auxiliary systems.

2.4 HEALTH CONSEQUENCE ANALYSIS

The ex-plant consequence analysis will characterize the distribution of public health effects which can result from accidents involving core damage and significant radionuclide releases to the environment. Results from this analysis will assess the uncertainties in public health effect distributions which result from uncertainties in predicted accident sequence probabilities and radionuclide releases from containment.

The characterization of health consequences will be accomplished using the CRAC II computer code together with the meteorological and demographic data for the CRBRP site.

The health consequences associated with the release will be defined based on the source terms derived for each release category. The study will use state-of-the-art modeling codes which accurately project doses for the LMFBR source terms. The health consequences will examine both acute fatalities and latent cancer fatalities.

2.5 RISK ANALYSIS

Based on health consequences and sequence probabilities derived in the above tasks, an overall assessment of the risk shall be provided, including a breakdown of the major contributors to the risk. This assessment of the dominant contributors to risk shall include design and operational aspects, and sensitivity to key assumptions, and shall be kept current with knowledge of the plant and the PRA model.

2.6 PRA APPLICATIONS TASKS

The purpose of this section of the plan is to summarize the tasks which will be used for application of the PRA.

A number of PRA applications will be implemented. These applications rely on two characteristics of the PRA results:

1. The PRA is a complete description of the accident sequences which have the potential to cause damage to the core;

2. The PRA incorporates sufficient information to provide a quantitative ranking of the importance of equipment failures and human errors to both the frequency of core damage and the public health risk.

The use of these characteristics in a variety of application tasks is discussed below.

2.6.1 Operator Action Event Trees

Operator Action Event Trees (OAETs) will be developed. OAETs are a method to investigate the role of the plant operation staff in important accident sequences (Ref. NUREG/CR-1440). The analysis addresses three fundamental questions:

1. What actions can (or must) the operator take in response to a specific accident condition?

2. What information is required by the operator to take this action?

3. What instrumentation is necessary and sufficient to provide this information?

By developing logic models and supporting information which allow these questions to be addressed systematically, a very detailed description of the operator's role in managing an accident sequence can be developed. This description will also provide information about the specific role of plant instrumentation in informing the operator of the status of the plant. The complete set of OAETs WILL consist of one tree applicable for each dominant accident sequence. Common characteristics of a number of dominant sequences will allow the total number of OAETs required to be reduced to fewer than the number of dominant sequences.

2.6.2 Assessment of the Effectiveness of Postulated Design Variations Including Consequence Mitigation Features

Models developed during the PRA will be utilized to assess the potential benefits or lack thereof associated with postulated changes in plant design. These changes may be oriented toward reducing the frequency of events which produce core damage or mitigating the consequences of these events.

The present design of the CRBRP containment includes a number of systems designed to mitigate accident consequences. A quantitative display of the effects of these features on the risk from the CRBRP will be developed. Such a comparative evaluation is called for in NUREG-0718, item II.B.8. This evaluation will include sensitivity studies in which the effectiveness both of currently designed and of postulated consequence mitigation systems can be assessed.

In addition, a search of dominant accident sequences will be conducted to assess whether cost effective modifications to the existing design can be postulated. Where such potentially useful modifications are identified, a more detailed evaluation of alternative approaches to reducing the risk contribution from one or more dominant accident sequences can be performed. This evaluation can include assessment of feasibility, effectiveness, and cost of a variety of postulated changes.

2.6.3 Improve Understanding of the Plant

Additional PRA applications will be undertaken to factor insights gained in the conduct of the PRA into the design and operation of the plant. These applications will:

1. Supplement the existing programs designed to address operator aids including Reg. Guides 1.47, 1.97, and NUREG-0497. The PRA will be used to define and rank the risk significance of alarms and instrumentation which are designed to improve the operator's ability to prevent and mitigate the consequences of severe accident sequences.

2. Assist in the development and validation of emergency procedure guidelines.

3. Provide information on the integrated performance of plant systems and instrumentation for use in evaluating the design and utilization of the plant simulator, as well as to train operators and other plant personnel.

4. Assess of the sensitivity of the CRBRP risk to uncertainties in the reliability of equipment required to perform its function in a degraded environment. If appropriate, alternative design features intended to reduce the sensitivity of the overall plant risk to these uncertainties will be defined and evaluated.

5. Evaluate the risk contribution and sensitivities to the testing interval of equipment and to the allowable on-line maintenance interval. This evaluation will allow Technical Specifications to be implemented in a manner which assures the minimum plant risk without unnecessarily restricting plant operation during the maintenance of safety-related equipment.

2.6.4 Characterization of Risk From Early Life Failures

The approach used to assess with the failures anticipated to occur during early years of operation will include two important elements. Both of these elements involve the careful screening of available operational data and results from the PRA. The first element is to analyze the data to focus on the potential for systematic recurring failure causes and to identify measures which have been successfully used in the past to contend with these causes. The second element is to focus on the equipment which has or is expected to produce the most significant operational problems and to define operational, maintenance, or training programs which might reduce the severity of these specific equipment failures.

2.6.5 Implementation of a Continuing Risk Management Program

The PRA will have application as a tool to evaluate operational experience and to address licensing issues which will arise during the operation of the plant. Implementation of such a continuing risk management program will include:

1. Formalization of the models and documentation developed during the PRA to facilitate ease of long-term utilization;

2. Transfer of the PRA technology and associated tools to the TVA operatons staff;

3. Definition of a TVA program by which the PRA and its associated documentation cna be updated to reflect the current state of the plant design and operation as well as current operational experience.

The risk management program will allow applications to be carried out by the TVA plant staff throughout the life of the plant. The program will include evaluating operational experience and addressing licensing issues which might arise during operation of the plant.

The program will also provide assurance that operational and back-fit decisions will be based on a realistic and complete understanding of the important safety characteristics of the plant. This understanding will be influenced by experience gained in the operation of the plant.

2.6.6 Input to the Site Emergency Procedures

The PRA will be used in the development and implementation of the site emergency procedures. The use of the PRA in this role is supported by the fact that it embodies a description of important accident sequences which includes estimates of the timing of significant radionuclide releases relative to the occurrence of the initiating event and the subsequent system failures which lead to significant core damage.

By using the PRA estimates for the timing of the accident together with a description in the CRAC-II code of the effect of meteorology and demography on population exposure, various strategies will be developed and assessed to determine the combination of evacuation and shielding (i.e., non-evacuation) which minimize population exposure given a set of meteorological conditions.

2.7 INTERACTION WITH THE NRC

The Project plans to support an interactive, phased review process on a schedule acceptable to both NRC and the Project. The review will promote an improved understanding of the PRA complexities, uncertainties, and validity. NRC is expected to provide comments on schedule, scope, and detailed implementation for consideration as the work progresses.

This review process will be carried out at appropriate intervals during the PRA program in a two-stage format. The first stage will be an overview to provide information to NRC management on the overall status of the effort and the significant results. The second stage will involve informal detailed discussions of methodology and interim results. This latter stage is aimed at providing technical detail to the NRC staff and its consultants.

As noted earlier, as the PRA progresses a more detailed definition of methodology to be used in such analyses as seismic risk characterization will be developed. It is expected that this more detailed methodology will be presented at appropriate NRC review meetings and that comments will be considered on the selected methodology as it relates to issues which the NRC considers to be candidates for resolution or prioritization using the PRA.

2.8 ACCIDENT DELINEATION

The CRBRP Project is currently pursuing a program to assure that all appropriate sequences are included within the plant design bases. A complete set of initiators together with a well formulated and quantified set of event trees and fault trees will provide a set of accident sequences in a probabilistic context. An assessment will be performed to determine if all appropriate accident initiators and accident sequences are included in CRBRP's Design Basis Accidents. This assessment will provide the decision criteria for the conclusions presented. This review will include the following considerations:

1. Development of a set of criteria on which definition of sequences comprising the design envelope should be based will be presented in the referenced project documentation.

2. Characterization of accident sequences identified in the PRA (including sequences which do not result in core damage) by:

   a. Occurrence frequency;

   b. Number of active failures following the initiating event (minimum);

   c. Number of passive failures following the initiating event (minimum);

   d. Severity of sequence impact on the environment surrounding safety-related equipment;

   e. Severity of sequence challenge to systems designed to remove d heat (e.g., how many different systems are capable or availabl remove decay heat at the end of the sequence);

   f. Severity of sequence challenge to reactor structures, including the containment building;

   g. The availability of support systems in important sequences at point in the sequence at which a particular system is required to perform its function.

3. Consideration of both plant induced and external initiating event sequences in this analysis;

4. Select sequences based on the criteria in (1) and the characteristics in (2);

5. Compare the selected sequences with those which currently comprise the design envelope and group the sequences by the various measures of severity defined in (2). The result of this comparison and grouping should be a reduced set of sequences. Any significant new events which are identified will be added to the design basis.

The product of this effort will be presented and discussed at one of the NRC program review meetings.

2.9 STUDY LIMITATIONS

A brief listing of the study limitations is presented below:

1. COMPLETENESS AND LEVEL OF DETAIL OF THE MODELS- The completeness and level of detail of the models will be limited as a result of the state of the design and the unavailability of details of construction. These limitations can, however, also be viewed as strengths since there is the opportunity to utilize the PRA in reviewing these design details for their risk implications as they are established.

2. HUMAN FACTORS ANALYSIS- The role of the plant operations staff in the initiation, aggravation, and mitigation of an accident will be modeled in the fault trees and event trees developed to describe the sequence of events. The ability of the models and quantification methods is limited. However, a supplementary approach will be utilized. In this approach, operator action event trees will be used to investigate the role of the operations staff in severe accident sequences.

3. EXTERNAL EVENT QUANTIFICATION- Experience with the analysis of risk from external events (e. g., seismic events) has shown that the associated uncertainties are significantly larger than for accident sequences initiated by in-plant causes (which typically have a less pervasive effect on equipment reliability). Nevertheless, external events have in some cases been assessed to be significant contributors to plant risk. This analysis with its inherent uncertainties has, therefore, been included within the scope of the PRA.

4. FAILURE DATA- A significant quantity of failure data is available for equipment in many systems (i.e., steam, fire protection, electrical, control, and communication) of the CRBRP. These data have uncertainties no greater than those associated with LWRs. Components in the liquid metals systems and at the interface between the sodium and other systems are less well characterized. Although uncertainties in the reliability of these components exist, the implications of these uncertainties to the risk profile can be characterized using sensitivity studies carried out within the PRA. Other areas in which significant uncertainties exist which may be important to the overall description of plant risk include:

   a. Initiator frequency;

   b. Equipment reliability in a degraded environment;

   c. Equipment repair time distributions and allowable on-line maintenance intervals.

5. ACCIDENT CHARACTERISTICS- The response of the core to conditions which will produce core degradation and the response of the

containment to severe accident sequences are somewhat uncertain. These uncertainties are being handled by the use of phenomenological event trees developed to describe physical processes which can lead to accident energetics and to containment failure following core damage.

6. SITE SPECIFIC CHARACTERISTICS- The effects of uncertainties in site meteorology and demography as well as in emergency response procedures can produce significant uncertainties in overall risk. Again, the effect of these uncertainties will be investigated using sensitivity analysis.

3.0 PRA PERFORMANCE AND REVIEW

A program of plans and actions will be implemented to control and verify quality of the results from the PRA.

This program will include measures and documentation to assure that:

1. data is reviewed and evaluated systematically to verify completeness and correctness with respect to the PRA requirements. This includes assurance that analyses are verified by the PRA analysts and appropriate design organizations that the work has been performed satisfactorily; and

2. the methods used for verification are identified and the verification results are documented.

Organizations which are presently involved in implementation of the study are shown in Figure 2. Also shown on the figure in a box separated from the PRA performers are the design organizations. These organizations will serve to provide information about the plant features to the PRA performers and to review the technical results of the study for accuracy and completeness.

The overall review program is pictured in Figure 3. As shown, four levels of review by the CRBRP Project Office. The first level is a working review by the performing organizations designed to assure the technical accuracy, clarity, and consistency of elements of the product. The second level is a review by CRBRP Project interfacing organizations to assure consistency of the elements of the analysis with plant design and operational characteristics. The third level of review will be conducted by a project management review committee to assess the validity of the approach taken to project integration and to assure proper implementation of the approach. Finally, the fourth step is an overall review by a peer review-group made up of participants external to the Project. The purpose of this final review is to assess the adequacy of the program integration and to evaluate the consistency of the methods used with the state-of-the-art.

The CRBRP Project Office will ultimately be responsible for utilizing the results and insights from the risk assessment to help ensure that the systems designed to shutdown the plant, to cool the core, and to mitigate the effects of severe accident sequences are designed and operated to be consistent with the results of the PRA. Decisions on possible changes to the plant design will be made through established Project procedures. These procedures ensure

both consideration and review of proposals by all affected personnel throughout the project.

4.0 SCHEDULE, MILESTONES, AND RESOURCE ALLOCATION

The PRA products are listed in Table 1 and the schedule for key milestones for the entire PRA is depicted on Figure 4. As shown, the program is expected to produce a final report in late 1984.

Table 2 provides estimates of the resource allocation for each task. The values in the table will fluctuate over the course of the PRA, but the table communicates relative levels of effort for each task. Table 3 provides the planned completion dates for the tasks. The results of these tasks will be available for NRC review after the completion dates.

TABLE 1
CRBRP PRA PRODUCTS

PRODUCTS

- o INITIATING EVENT TOP LOGIC AND INITIATOR COMPLETENESS ANALYSIS
- o PROBABILISTICALLY QUANTIFIED ACCIDENT SEQUENCES AND THEIR BASIS
  - A. SYSTEM FUNCTIONAL EVENT TREES
  - B. FAULT TREES
  - C. DETAILED COMMON CAUSE FAILURE ANALYSIS (SYSTEMS INTERACTION EVALUATION)
  - D. EXTERNAL EVENT EVALUATION (SEISMIC, ETC.)
- o CORE DAMAGE PHENOMENOLOGICAL EVENT TREES AND QUANTIFICATION
- o CONTAINMENT PHENOMENOLOGICAL EVENT TREES AND QUANTIFICATION
- o UNCERTAINTY ANALYSIS
- o RADIONUCLIDE RELEASE ANALYSIS
- o HEALTH CONSEQUENCE ANALYSIS
- o ANALYSIS OF EX-CORE SOURCES OF RADIONUCLIDES
- o DEFINITION OF PROGRAM TO SUPPORT CONTINUING OPERATIONAL APPLICATIONS
- o OPERATOR ACTION EVENT TREES AND APPLICATIONS TO OPERATIONS SUPPPORT AND TRAINING PROGRAMS
- o DEFINITION OF AN ON-GOING RISK MANAGEMENT PROGRAM
- o EVALUATION OF POTENTIAL RISK REDUCTION ASSOCIATED WITH SUGGESTED DESIGN CHANGES
- o EVALUATION OF RISK CONTRIBUTION AND SENSITIVITIES TO EQUIPMENT TESTING INTERNALS (TECH. SPEC. IMPACT)
- o DETAILED DOCUMENTATION OF STUDY AND FINAL REPORT

TABLE 2
RESOURCE ALLOCATION

Listed below is each major task associated with the risk assessment and the manpower estimated for that task.

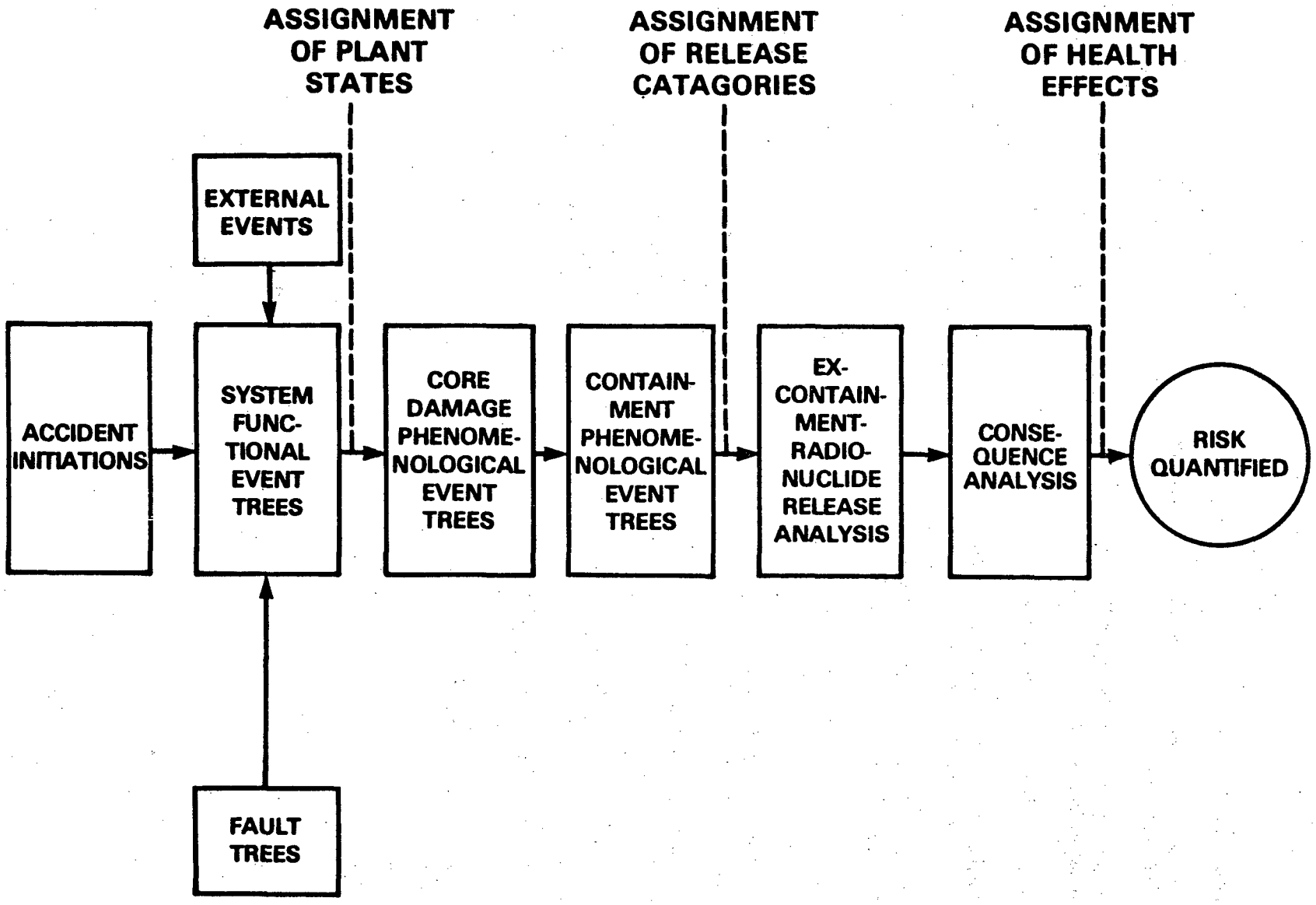
Task	Man-Months
Study Direction	28.0
Accident Initiator Identification	4.6
Event Tree Development	25.7
Initiating Event Data	3.3
System Fault Tree Development	65.8
Component Failure Data	1.0
Fault Tree Quantification	25.6
Event Tree Quantification	5.3
Common Cause Failure Analysis	22
Sensitivity Analysis	10.3
Dependency Analysis	12.2
Systems Failure Criteria	6.3
Core Response Event Trees	11.2
Containment Response Event Trees	11.2
Source Term	27
Consequence	12
Risk Analysis	9
Ex-core Sources	7
Uncertainty	7
OAET	13
Site Emergency	8
Input LCOs	7
Accident Delineation	6
On Going Risk Management	9

TABLE 3
CURRENT SCHEDULE OF UNCOMPLETED TASKS

DESCRIPTION	DUE DATE
Provide final written Accident Sequence Definition Review	3/31/83
Provide final written Radionuclide Release Analysis	12/31/83
Provide final written Uncertainty Analysis	10/31/84
Provide final written Detailed Common Cause Failure Analysis	10/31/84
Provide final written Accident Delineation Report	10/31/84
Provide final written Health Consequence Analysis	10/31/84
Provide written Risk Management Program Report	12/31/84
Provide written Operator Action Event Trees Report	12/31/84
Provide written input to Operational Procedures and Testing Interval	12/31/84
Provide written input to Site Emergency Plan	12/31/84
Provide final written Report	12/31/84

FIGURE 1

PRA FLOWCHART



11-82-2875-1-2

Figure 2

CRBRP PRA ORGANIZATION

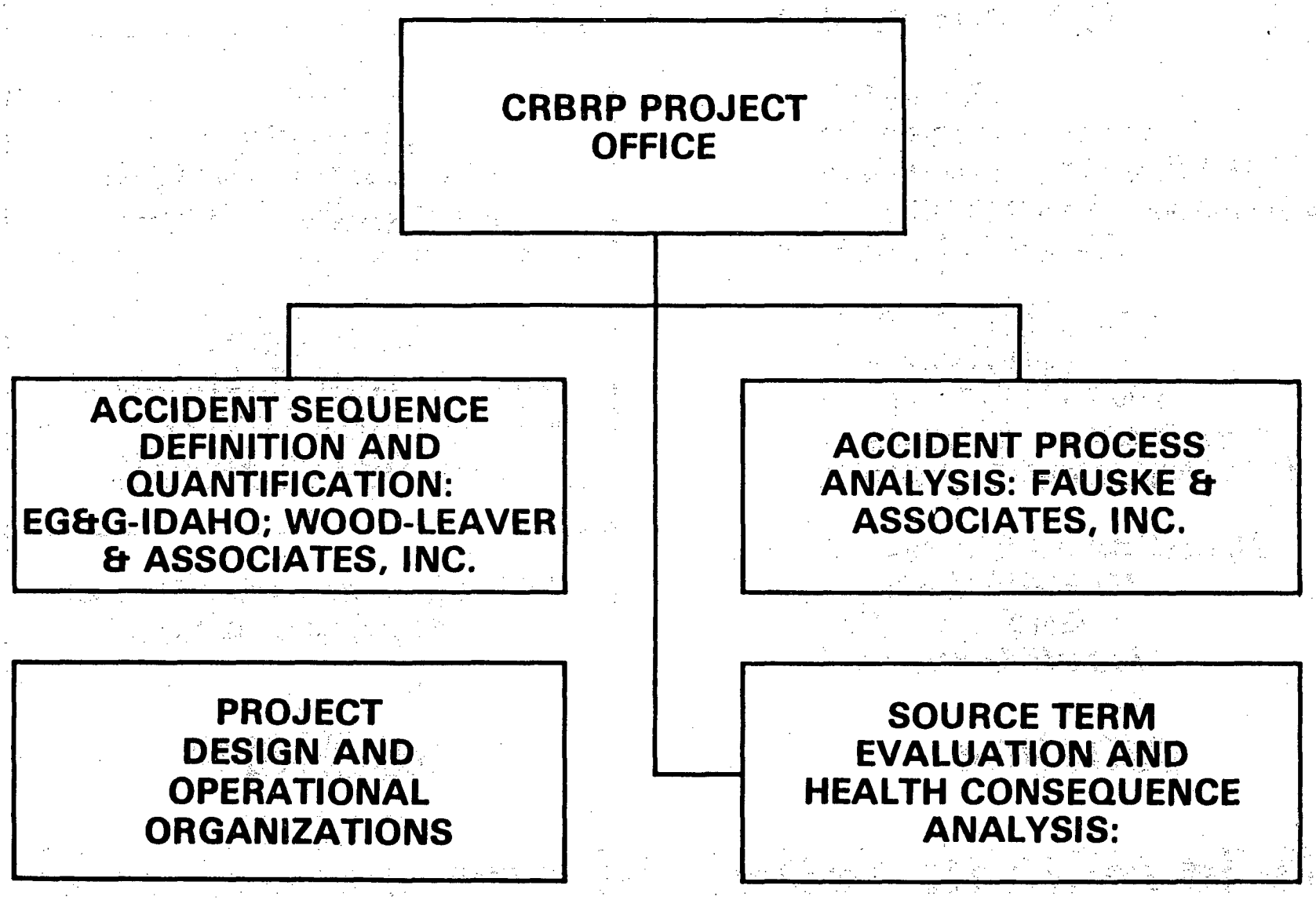


FIGURE 3

ANTICIPATED PRA INTERNAL REVIEW PLAN

ELEMENTS

INTERNAL REVIEW BY ORIGINATING ORGANIZATION

REVIEW BY PROJECT INTERFACING ORGANIZATIONS (DESIGNERS, SAFETY AND LICENSING)

OVERALL REVIEW BY DESIGNATED PROJECT COMMITTEE

OVERALL REVIEW BY PROJECT AND EXTERNAL PEER REVIEW BOARD

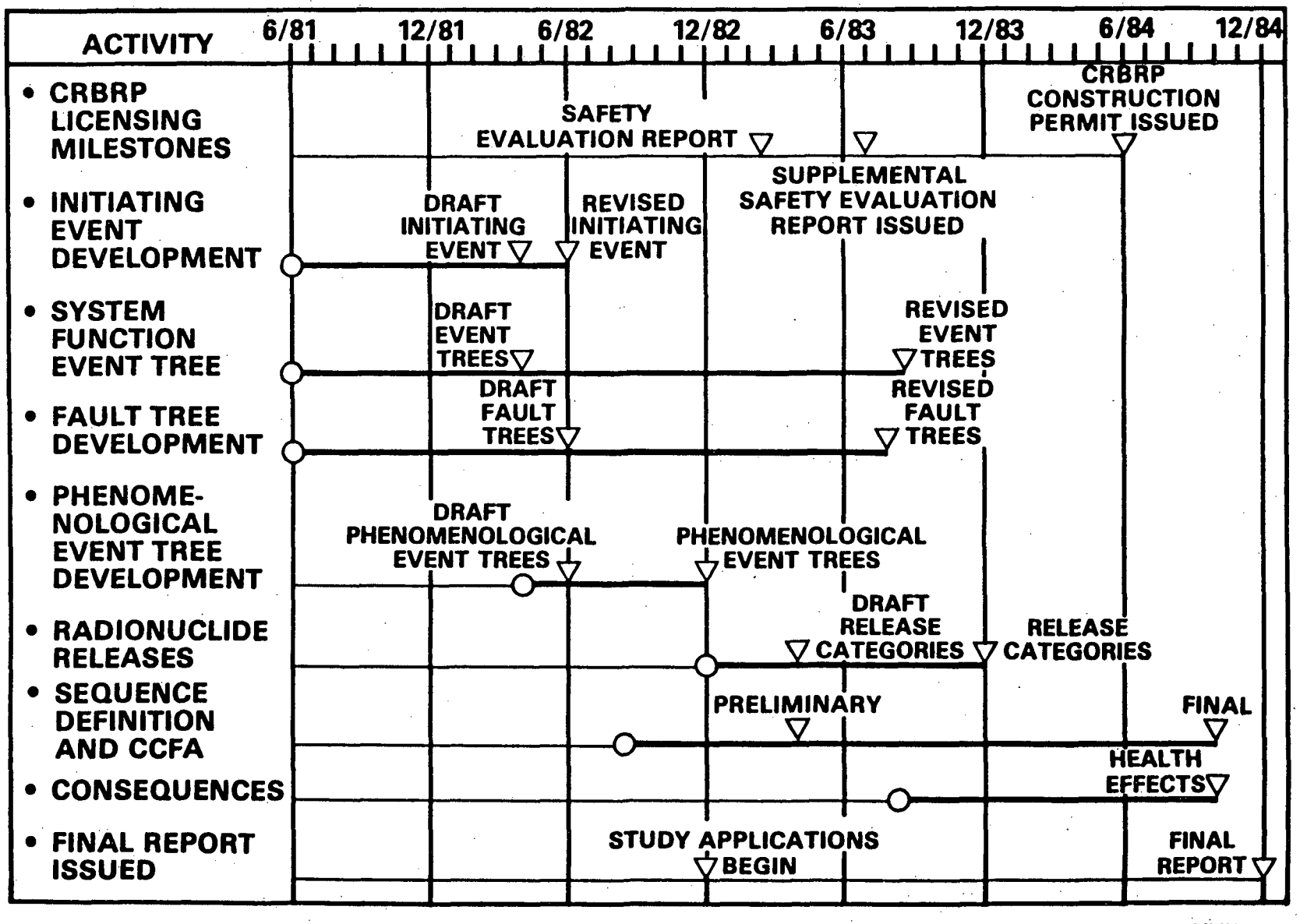
PURPOSE

- ASSURE QUALITY, CLARITY, AND CONSISTENCY OF ELEMENTS OF PRODUCT
- ASSURE CONSISTENCY OF ELEMENTS OF ANALYSIS WITH DESIGN AND OPERATIONAL CHARACTERISTICS
- ASSURE PROPER INTEGRATION OF PRODUCT ELEMENTS AND ACCURACY OF APPROACH TO INTEGRATION
- PEER REVIEW OF INTEGRATION APPROACH FOR ACCURACY AND CONSISTENCY WITH STATE-OF-THE-ART IN PRA

6-82-2875-4-1

Figure 4

EXPECTED SCHEDULE OF PRA PRODUCTS



AMENDMENT 3

LIST OF RESPONSES TO NRC QUESTIONS

REFERENCE: NRC LETTER DATED JUNE 5, 1975

NRC QUES. NO.	NRC QUES. NO.
001.1	120.5
001.15	
001.22	222.14
001.24	222.23
001.75	222.30
001.126	222.31
001.175	222.32
001.193	222.34
001.194	222.35
001.204	
001.208	310.16
001.217	310.19
001.245	
001.246	322.14
001.248	322.16
110.8	

REFERENCE: NRC LETTER DATED JUNE 11, 1975

323.43	323.44

AMENDMENT 1

LIST OF RESPONSES TO NRC QUESTIONS

REFERENCE: NRC LETTER DATED JUNE 5, 1975

NRC QUES. NO.	NRC QUES. NO.	NRC QUES. NO.	NRC QUES. NO.
310.3	310.15	331.1	331.14
310.4	310.17	331.5	331.15
310.5	310.20	331.6	331.20
310.6	310.21	331.7	
310.7	310.22	331.8	421.1
310.8	310.23	331.9	421.2
310.9	310.26	331.10	
310.10	310.30	331.11	430.1
310.13		331.12	

REFERENCE: NRC LETTER DATED JUNE 11, 1975

321.12
322.13
322.15

AMENDMENT 2

LIST OF RESPONSES TO NRC QUESTIONS

REFERENCE: NRC LETTER DATED JUNE 5, 1975

NRC QUES. NO.	NRC QUES. NO.	NRC QUES. NO.	NRC QUES. NO.
001.16	001.101	001.227	130.3
001.19	001.108	001.249	130.4
001.28	001.130	001.259	130.5
001.30	001.131		130.7
001.31	001.137	011.5	130.20
001.35	001.143	011.6	
001.36	001.188		222.36
001.37	001.189	020.11	222.37
001.44	001.190	020.20	
001.58	001.191		310.11
001.73	001.202	110.17	310.18
001.79	001.205		
001.81	001.212	120.4	331.13
001.82	001.213	120.8	331.16
001.99		120.9	331.18
001.100			

REFERENCE: NRC LETTER DATED JUNE 11, 1975

323.45

THERE ARE NO RESPONSES TO NRC QUESTIONS IN THIS AMENDMENT.

AMENDMENT 6

LIST OF RESPONSES TO NRC QUESTIONS

REFERENCE: NRC LETTER DATED JUNE 5, 1975

NRC QUES. NO.	NRC QUES. NO.	NRC QUES. NO.	NRC QUES. NO.
001.42	001.94	001.122	001.260
001.54	001.95	001.138	001.261
001.56	001.96	001.141	001.262
001.57	001.102	001.142	001.263
001.59	001.105	001.147	001.264
001.64	001.113	001.148	001.268
001.69	001.115	001.149	001.270
001.84	001.117	001.152	001.271
001.86	001.118	001.166	001.272
001.88	001.119	001.183	001.273
001.90	001.120	001.244	
001.91	001.121	001.247	020.12
001.92			
			310.4

Amendment 7

List of Responses to NRC Questions

Reference: NRC Letter dated November 19, 1974

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
322.1	323.2	323.27
322.2	323.4	323.28
322.3	323.6	323.29
322.4	323.9	323.30
322.5	323.22	323.35
322.8	323.23	323.36
322.9	323.24	323.37
322.11	323.25	323.38
322.12	323.26	323.39
		323.40
		323.41
		323.42

Reference: NRC Letter dated November 27, 1974

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
321.1	321.5	321.9
321.2	321.6	321.10
321.3	321.7	321.11
321.4	321.8	

Reference: NRC Letter dated June 5, 1975

001.41	001.136	130.17
001.43	001.165	130.18
001.45	001.199	
001.46	001.225	331.17
001.48	001.230	
001.49	001.239	

Amend. 7
Nov. 1975

AMENDMENT 8

List of Responses to NRC Questions

Reference: NRC Letter dated June 5, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.61	001.169	001.267
001.87	001.211	
001.111	001.219	110.6
001.112	001.223	
001.128	001.232	120.17
001.135	001.237	
001.146	001.241	310.15
001.150	001.245	310.29
001.154	001.250	
001.163	001.254	331.12
	001.266	331.19

AMENDMENT 9

LIST OF RESPONSES TO NRC QUESTIONS

Reference : NRC letter dated June 5, 1975

NRC Question No.	NRC Question No.	NRC Question No.
001.38	001.133	001.228
001.45	001.140	001.240
001.77	001.144	001.243
001.107	001.172	001.258
001.132	001.226	

Reference: NRC letter dated October 29, 1975

NRC Question No.	NRC Question No.	NRC Question No.
001.280	001.317	011.12
001.281	001.319	011.13
001.288	001.323	020.33
001.290	001.325	020.34
001.291	001.328	110.42
001.292	001.331	130.34
001.293	001.336	130.35
001.295		130.36
001.298	001.340	331.25
001.315	001.344	331.26
		411.6

Amendment 10

List of Responses to NRC Questions

Reference: NRC letter dated June 5, 1975

NRC Quest. No. 411.1

Reference: NRC letter dated October 29, 1975

NRC Quest. No. 411.2
411.3
411.4
411.5

Amendment 11

List of Responses to NRC Questions

Reference: NRC Letter dated June 5, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.6	001.218	130.4
001.103	001.220	310.25
001.116	001.221	310.27
001.160	001.222	421.1
001.168	001.242	
001.179	020.9	

Reference: NRC Letter dated October 29, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.286	120.48	421.6
001.359	120.54	421.7
001.360	130.28	421.8
011.11	130.29	421.9
011.18	130.30	421.10
110.46	331.23	421.11
120.31	331.24	430.2
120.36	411.3	430.3
120.37	411.4	430.4
120.44	421.3	430.5
	421.4	430.6
	421.5	430.7

Reference: NRC Letter dated December 30, 1975

NRC Ques. No.
001.448
001.457
001.463

List of Responses to NRC Questions

Reference: NRC letter dated June 5, 1975

NRC Quest.No.	NRC Quest. No.	NRC Quest. No.
001.40	001.251	222.33
001.65	020.10	310.2
001.181	120.16	310.24
001.201	130.6	310.31
001.216	130.8	422.1
001.224		

Reference: NRC letter dated October 29, 1975

NRC Quest. No.	NRC Quest. No.	NRC Quest. No.
001.278	011.15	020.31
001.279	011.16	110.24
001.299	011.17	110.45
001.337	020.27	130.25
	020.29	130.26
	020.30	130.27

Reference: NRC letter dated November 18, 1975

NRC Quest. No.	NRC Quest. No.	NRC Quest. No.
001.370	222.84	324.4
222.43	222.86	324.5
222.53	222.87	324.6
222.81	222.88	324.7
222.82	324.1	324.8
222.83	324.2	324.9
	324.3	324.10

Reference: NRC letter dated December 5, 1975

NRC Quest. No.	
323.47	(Questions renumbered to avoid duplication from NRC letter dated June 11, 1975)
323.48	

Amendment 13

List of Responses to NRC Questions

Reference: NRC letter Dated June 5, 1975

NRC Ques. No.	NRC Ques. No.
001.63	001.255
001.125	001.256
001.134	001.257
001.184	001.269
001.187	110.3
001.214	110.10

Reference: NRC Letter Dated October 29, 1975

NRC Ques. No	NRC Ques. No.	NRC Ques. No.
001.306	001.330	120.33
001.311	001.332	120.41
001.314	001.339	120.49
001.316	001.343	120.60
001.320	001.357	322.17
001.326	020.36	412.3
001.329	110.36	

Reference: NRC Letter Dated November 18, 1975

NRC Ques. No.
001.387
001.404
222.42
241.12
241.27
241.47

Reference: NRC Letter Dated December 30, 1975

NRC Ques. No.
413.5
413.6

Reference: NRC Letter Dated January 28, 1976

NRC Ques. No.
310.34

Amendment 14

List of Responses to NRC Questions

Reference: NRC letter dated June 5, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.51	001.185	310.12
001.70	130.11	323.46

Reference: NRC letter dated October 29, 1975

NRC Ques. No.	NRC Ques. No.
001.285	110.49
001.322	110.50
001.327	120.28
001.334	120.29
001.342	120.32
001.355	120.42
001.362	120.43
020.28	120.47
110.23	120.51
110.30	120.59
110.33	120.61
110.39	331.21
110.48	331.22

Reference: NRC letter dated November 18, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.374	222.56	241.33
001.378	222.68	241.35
001.382	241.4	241.37
001.383	241.6	241.38
001.384	241.9	241.42
001.385	241.11	241.43
001.389	241.16	241.44
001.391	241.18	241.49
001.392	241.20	241.50
001.393	241.24	241.51
001.394	241.25	241.53
001.395	241.28	241.55
001.397	241.31	241.59
001.403	241.32	241.60

Amendment 15

List of Responses to NRC Questions

Reference: NRC Letter dated June 5, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.4	001.184	020.17
001.23	001.185	120.6
001.153		

Reference: NRC Letter dated October 29, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.296	001.342	020.32
001.297	001.352	120.36
001.305	001.353	120.39
001.307	001.354	120.58
001.338	001.356	

Reference: NRC Letter dated November 18, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.380	222.62	241.22
001.386	222.63	241.34
001.390	222.69	241.36
001.404	222.87	241.39
	241.3	241.48
222.52	241.5	241.54
222.60	241.10	241.59
222.61	241.21	

Reference: NRC Letter dated December 30, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.421	001.527	001.559
001.503	001.534	001.564
001.505	001.536	020.37
001.508	001.541	020.40
001.510	001.549	222.93
001.521	001.551	

List of Responses to NRC Questions

Reference: NRC Letter Dated June 5, 1975

NRC Ques. No.
001.10

Reference: NRC Letter dated October 29, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.282	110.44	120.50
001.283	120.34	130.32
001.321	120.35	130.33

Reference: NRC Letter dated November 18, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.368	241.2	241.19
001.369	241.7	241.23
001.371	241.8	241.26
222.55	241.13	241.41
222.72	241.14	241.52
241.1	241.17	241.58

Reference: NRC Letter dated December 30, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.417	001.520	001.565
001.424	001.522	020.39
	001.526	040.1
001.444	001.528	040.2
001.467	001.531	110.53
001.506	001.535	110.55
001.515	001.546	130.44
001.516	001.563	222.89
001.518		222.97

Amend. 16
Apr. 1976

Amendment 17

List of Responses to NRC Questions

Reference: NRC Letter dated June 5, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.50	001.186	130.13
001.157	001.265	310.1

Reference: NRC Letter dated October 6, 1975

NRC Ques. No.
20

Reference: NRC Letter dated October 29, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.275	110.29	120.53
001.287	110.31	120.55
011.19	110.51	120.56
110.20	120.45	130.31
110.28	120.52	

Reference: NRC Letter dated November 18, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.367	222.57	241.57
001.379	241.30	241.61

Reference: NRC Letter dated December 30, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.408	001.468	001.552
001.416	001.500	001.557
001.442	001.509	001.558
001.452	001.511	001.560
001.453	001.512	040.12
001.466		

THERE ARE NO RESPONSES TO NRC QUESTIONS IN THIS AMENDMENT.

Amendment 19

List of Responses to NRC Questions

Reference: NRC Letter dated June 5, 1975

NRC Ques. No.	NRC Ques. No.
001.252	020.3
011.4	020.7

Reference: NRC Letter dated October 29, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.284	001.361	110.47
001.289	001.363	120.30
001.304	011.10	120.40
001.333	110.26	
001.349	110.35	

Reference: NRC Letter dated November 18, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.372	001.396	241.56
001.373	001.400	241.62
001.376	241.29	
001.377	241.40	

Reference: NRC Letter dated December 30, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.411	001.478	001.538
001.413	001.479	001.543
001.433	001.480	001.545
001.435	001.483	001.547
001.441	001.485	001.553
001.445	001.486	001.555
001.447	001.498	001.561
001.449	001.501	001.566
001.450	001.507	020.42
001.456	001.523	020.43
001.459	001.524	110.56
001.462	001.525	110.57
001.464	001.533	130.40
001.474	001.537	222.94
001.475		222.98

Amendment 20

List of Responses to NRC Questions

Reference: NRC letter dated June 5, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.156	222.28	331.3
001.209	331.2	331.4
001.215		

Reference: NRC letter dated October 29, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.294	001.312	020.35
001.302	001.335	110.22
001.308	001.345	110.34
001.309	001.346	110.43

Reference: NRC letter dated November 18, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.366	222.48	222.78
222.39	222.49	222.80
222.44	222.73	241.15
222.45	222.75	241.46
222.46		

Reference: NRC letter dated December 30, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.410	001.488	001.548
001.412	001.490	001.556
001.423	001.496	020.38
001.439	001.517	040.18
001.446	001.519	222.95
001.454	001.539	310.33

Amendment 21

List of Responses to NRC Questions

Reference: NRC Letter dated December 30, 1975

NRC Ques. No.
001.540

Amendment 22

List of Responses to NRC Questions

Reference: NRC letter dated June 6, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.109	001.170	222.12
001.114	001.177	310.14
001.151	222.6	

Reference: NRC letter dated October 29, 1976

NRC Ques. No.
001.324
001.341
001.351
020.26
110.27

Reference: NRC Letter dated November 18, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.365	222.59	222.90
001.388	222.66	222.92
001.399	222.70	222.96

Reference: NRC Letter dated December 30, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.415	001.489	001.542
001.418	001.491	001.550
001.430	001.492	020.41
001.437	001.494	040.17
001.440	001.497	110.54
001.443	001.499	130.42
001.470	001.513	130.45
001.473	001.514	130.46
001.487	001.529	130.47
		413.1

Amendment 23

List of Responses to NRC Questions

Reference: NRC letter dated November 19, 1974

NRC Ques. No.
323.8

Reference: NRC letter dated June 5, 1975

NRC Ques. No.
001.198
001.233

Reference: NRC letter dated October 29, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.286	001.347	001.364
001.301	001.348	110.40
001.303	001.350	110.52
001.313	001.358	120.57
001.318		

Reference: NRC letter dated November 18, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.375	222.40	222.58
001.398	222.41	222.64
001.401	222.47	222.76
222.38	222.54	222.77

Reference: NRC letter dated December 30, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.406	001.458	001.532
001.407	001.493	001.544
001.427		001.567

Amendment 24

List of Responses to NRC Questions

Reference: NRC letter dated June 5, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.159	001.210	001.253

Reference: NRC letter dated October 29, 1975

NRC Ques. No.
020.25
120.46
130.37

Reference: NRC letter dated November 18, 1975

NRC Ques. No.	NRC Ques. No.
222.51	222.67
222.65	

Reference: NRC letter dated December 30, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.425	001.469	001.554
001.426	001.471	040.6
001.428	001.472	040.8
001.460	001.477	040.10
001.465	001.482	130.38
	001.484	130.39
		310.32

Amendment 25

List of Responses to NRC Questions

Reference: NRC letter dated June 5, 1975

NRC Ques. No.
001.20
001.238

Reference: NRC letter dated October 29, 1975

NRC Ques. No.
001.277
001.300
011.14
412.1
412.2
422.2

Reference: NRC letter dated November 18, 1975

NRC Ques. No.
001.381
001.402

Reference: NRC letter dated December 30, 1975

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
001.419	001.436	001.481
001.420	001.461	001.530
001.431		130.41
001.434		130.43

Amendment 26

List of Responses to NRC Questions

Reference: NRC letter dated June 5, 1975

NRC Ques. No.
001.235
222.24
310.28

Reference: NRC letter dated October 6, 1975

NRC Ques. No.
6

Reference: NRC letter dated October 29, 1975

NRC Ques. No.
001.310
110.19
110.41

Reference: NRC letter dated November 11, 1975

NRC Ques. No.
222.50
222.71
222.74
222.79

Reference: NRC letter dated December 30, 1975

NRC Ques. No.
001.455
001.495
110.58
222.91

Reference: NRC letter dated December 30, 1975

NRC Ques. No.	
001.455	110.58
001.495	222.91

Amendment 27

List of Responses to NRC Questions

Reference: NRC letter dated June 5, 1975

NRC Ques. No.	
001.124	001.182
001.127	020.4
001.173	120.19

Reference: NRC letter dated October 6, 1975

NRC Ques. No.
12
14
15
19

Reference: NRC letter dated October 29, 1975

NRC Ques. No.	
011.20	120.27
120.25	411.4
120.26	

Reference: NRC letter dated November 18, 1975

NRC Ques. No.
001.405
222.85
241.45

Reference: NRC letter dated December 30, 1975

NRC Ques. No.
001.422
001.429
001.562
040.11

Reference: NRC letter dated August 17, 1976

NRC. Ques. No.
011.24

List of Responses to NRC Questions

Reference: NRC letter dated December 30, 1975

NRC Ques. No.
001.476

Reference: NRC letter dated June 23, 1976

NRC Ques. No.
421.12

Reference: NRC letter dated August 17, 1976

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
011.21	241.111	411.8
120.71	310.35	411.9
130.56	310.36	411.10
130.67	310.37	411.11
130.72	310.38	411.12
130.73	310.39	411.13
130.75	310.41	411.14
130.78	411.7	411.15
241.110		

Amendment 29

List of Responses to NRC Questions

Reference: NRC letter dated December 30, 1975

NRC Ques. No.
001.502
001.504
001.562

Reference: NRC letter dated August 17, 1976

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
020.45	241.78	241.109
020.47	241.79	241.112
040.25	241.80	241.114
110.59	241.82	241.120
110.63	241.83	241.123
110.70	241.87	241.125
120.70	241.88	241.126
120.73	241.89	241.127
120.74	241.92	241.130
130.50	241.94	241.133
130.58	241.95	241.134
130.60	241.96	310.47
130.70	241.97	
130.76	241.99	
130.77	241.100	
241.63	241.101	
241.69	241.102	
241.70	241.103	
241.72	241.104	
241.75	241.106	
241.76	241.107	
241.77		

Amendment 30

List of Responses to NRC Questions

Reference: NRC letter dated October 6, 1975

NRC Ques. No.
Q4

Reference: NRC letter dated December 30, 1975

NRC Ques. No.
001.438
001.451
040.3
040.7
040.9

Reference: NRC letter dated August 17, 1976

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
020.48	110.69		241.117
040.20	110.72	241.64	241.128
040.21	110.74	241.66	241.129
040.22	110.75	241.71	241.131
040.23	110.77	241.85	241.132
040.27	120.62	241.86	241.135
110.62	120.67	241.91	310.42
110.64	120.69	241.93	310.45
110.66	120.72	241.116	331.27
			331.28

Amendment 31

List of Responses to NRC Questions

Reference: NRC letter dated 10/6/75

NRC Ques. No.
Q16

Reference: NRC letter dated 12/30/75

NRC Ques. No.
Q040.5

Reference: NRC letter dated 6/23/76

NRC Ques. No.
Q421.13

Reference: NRC letter dated 8/17/76

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
Q020.46	Q120.64	Q130.61
Q040.19	Q120.65	Q130.62
Q040.26	Q130.48	Q130.65
Q110.65	Q130.49	Q130.66
Q110.73	Q130.51	Q241.81
Q120.63	Q130.54	

Amendment 32

List of Responses to NRC Questions

Reference: NRC Letter dated October 6, 1975

NRC Ques. No.
Q7
Q17
Q18

Reference: NRC Letter dated October 29, 1975

NRC Ques. No.
Q120.38

Reference: NRC letter dated August 17, 1976

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
Q001.580	Q241.67	Q241.90
Q120.66	Q241.68	Q241.98
Q120.68	Q241.73	Q241.105
Q130.55	Q241.74	Q241.108
Q130.63	Q241.84	Q241.121
Q241.65		

UNITED STATES
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
CLINCH RIVER BREEDER REACTOR PLANT PROJECT OFFICE
P. O. BOX U
OAK RIDGE, TENNESSEE 37830

809

Docket No. 50-537 File: 05.10

January 13, 1977

Mr. Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Boyd:

AMENDMENT NO. 33 TO THE PRELIMINARY SAFETY ANALYSIS REPORT FOR CLINCH RIVER BREEDER REACTOR PLANT

The application for a Construction Permit and Class 104(b) Operating License for the Clinch River Breeder Reactor Plant, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment No. 33 to the Preliminary Safety Analysis Report pursuant to 50.34 (a) of 10 CFR Part 50. This Amendment 33 to the PSAR includes: Responses to NRC requests for additional information contained in letters dated June 5, 1975, and August 17, 1976, and other updates and changes.

A Certificate of Service, confirming service of Amendment No. 33 to the PSAR upon designated local public officials and representatives of the EPA, will be filed with your office after service has been made.

Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

Anthony R. Buhl
Assistant Director for Public Safety

PS:L:77:022

Enclosure

cc: Service List
Standard Distribution
Licensing Distribution

SUBSCRIBED and SWORN to before me this 17th day of January, 1977.

Notary Public
My Commission Expires May 21, 1980

STANDARD DISTRIBUTION LIST

Mr. R. Balent, Vice President (2)
Atomics International Division
Rockwell International
P. O. Box 309
Canoga Park, California 91304

Mr. E. C. Brolin, Project Manager (2)
CRBRP Project
Burns and Roe, Inc.
700 Kinderkamack Road
Oradell, New Jersey 07649

Mr. Lochlin W. Caffey, Director (2)
Clinch River Breeder Reactor Plant Project
P. O. Box U
Oak Ridge, Tennessee 37830

Mr. James R. Chapman (2)
Senior Project Manager
Stone & Webster Engineering Corporation
P. O. Box 811
Oak Ridge, Tennessee 37830

Mr. R. W. Cochran, Chief (2)
CRBRP Program Management Branch
Division of Reactor Development
and Demonstration
U. S. Energy Research and
Development Administration
Washington, D. C. 20545

Mr. T. J. Iltis (1)
Senior Site Representative
U. S. Energy Research and Development
Administration
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, Pennsylvania 15663

Dr. W. M. Jacobi, Project Manager (2)
Clinch River Breeder Reactor Plant
Westinghouse Electric Corporation
Advanced Reactors Division
Clinch River Site
P. O. Box W
Oak Ridge, Tennessee 37830

Mr. J. E. Nolan (2)
Project Manager
CRBRP Reactor Plant
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, Pennsylvania 15663

Mr. John F. Pearson, Jr. (1)
Resident Manager
Burns & Roe, Incorporated
P. O. Box T
Oak Ridge, Tennessee 37830

Dr. R. E. Skavdahl, Manager (2)
Clinch River Project
General Electric Company
P. O. Box 5020
Sunnyvale, California 94086

Mr. J. H. Williams, Supervisor (2)
LMFBR Project Section
Tennessee Valley Authority
1300 Commerce Union Bank Building
Chattanooga, Tennessee 37401

LICENSING DISTRIBUTION

Mr. G. Williams, Jr.
Manager of Power
Tennessee Valley Authority
831 Power Building
Chattanooga, TN 37401

Mr. R. M. Little
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

Dr. Jeffrey H. Broido, Manager
Analysis and Safety Department
Gas Cooled Fast Reactor Program
P. O. Box 81608
San Diego, CA 92138

Dirk Van Hoesen, Licensing Branch, Public Safety Division

Ron Green, Acting Chief, Technology Branch, Engineering Division

SERVICE LIST

Atomic Safety & Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Atomic Safety & Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

S. Wallace Brewer, Judge
Office of County Judge
Roane County Court House
Kingston, Tennessee 37763

Dr. Thomas Cochran
Natural Resources Defense Council, Inc.
917 15th Street, N.W.
8th Floor
Washington, D. C. 20005

Docketing & Service Station
Office of the Secretary
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Geoffrey P. Gitner, Esq.
Counsel for NRC Staff
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

William B. Hubbard, Esq.
Assistant Attorney General
State of Tennessee
Office of the Attorney General
422 Supreme Court Building
Nashville, Tennessee 37219

Mr. Gustave A. Linenberger
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Marshall E. Miller, Esq.
Chairman
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Luther M. Reed, Esq.
Attorney for the City of
Oak Ridge
253 Main Street, East
Oak Ridge, Tennessee 37830

Anthony Z. Roisman, Esq.
Roisman, Kessler and Cashdan
1712 N Street, N.W.
Washington, D. C. 20036

Dr. Cadet H. Hand, Jr., Director
Bodega Marine Laboratory
University of California
Post Office Box 247
Bodega Bay, California 94923

Lewis E. Wallace, Esq.
Division of Law
Tennessee Valley Authority
Knoxville, Tennessee 37902

AMENDMENT 33

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

This thirty-third amendment to the Clinch River Breeder Reactor Plant PSAR includes responses to NRC requests for additional information contained in the NRC letters as listed below. Updated design information is also included in the amendment. Vertical margin lines on the right hand side of the page are used to identify changes resulting from NRC questions, and margin lines on the left to identify new design information.

A page replacement guide appears following the list of responses to NRC Questions.

Reference: NRC Letter dated June 6, 1975

NRC Ques. No.
Q001.234

Reference: NRC Letter dated August 17, 1976

NRC Ques. No.	NRC Ques. No.
Q020.44	Q241.113
Q130.52	Q241.115
Q130.53	Q241.118
Q130.64	Q241.119
Q130.71	Q241.122
Q130.74	

PAGE REPLACEMENT GUIDE FOR

AMENDMENT 33

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 33 to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket No. 50-537. Amendment 33 consists of new and replacement pages for the PSAR text and Question/Response Supplement pages.

The following attached sheets list Amendment 33 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

Amendment 33

List of Responses to NRC Questions

Reference: NRC Letter dated June 6, 1975

NRC Ques. No.
Q001.234

Reference: NRC Letter dated August 17, 1976

NRC Ques. No.	NRC Ques. No.
Q020.44	Q241.113
Q130.52	Q241.115
Q130.53	Q241.118
Q130.64	Q241.119
Q130.71	Q241.122
Q130.74	

UNITED STATES
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
CLINCH RIVER BREEDER REACTOR PLANT PROJECT OFFICE
P. O. BOX U
OAK RIDGE, TENNESSEE 37830

866

Docket No. 50-537 File: 05.10

February 3, 1977

Mr. Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Boyd:

AMENDMENT NO. 34 TO THE PRELIMINARY SAFETY ANALYSIS REPORT FOR CLINCH RIVER BREEDER REACTOR PLANT

The application for a Construction Permit and Class 104(b) Operating License for the Clinch River Breeder Reactor Plant, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment No. 34 to the Preliminary Safety Analysis Report pursuant to 50.34(a) of 10 CFR Part 50. This Amendment 34 to the PSAR includes: responses to NRC requests for additional information contained in letters dated October 29, December 30, 1975, August 17 and December 1, 1976; and other updates and changes.

A Certificate of Service, confirming service of Amendment No. 34 to the PSAR upon designated local public officials and representatives of the EPA, will be filed with your office after service has been made.

Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

Anthony R. Buhl
Assistant Director for Public Safety

PS:L:77:101

Enclosure

cc: Service List
Standard Distribution
Licensing Distribution

SUBSCRIBED and SWORN to before me this 28 day of January, 1977.

Dorothy S. Dare
Notary Public

My Commission Expires Dec. 26, 1977

SERVICE LIST

Atomic Safety & Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Atomic Safety & Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

S. Wallace Brewer, Judge
Office of County Judge
Roane County Court House
Kingston, Tennessee 37763

Dr. Thomas Cochran
Natural Resources Defense Council, Inc.
917 15th Street, N.W.
8th Floor
Washington, D. C. 20005

Docketing & Service Station
Office of the Secretary
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Geoffrey P. Gitner, Esq.
Counsel for NRC Staff
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

William B. Hubbard, Esq.
Assistant Attorney General
State of Tennessee
Office of the Attorney General
422 Supreme Court Building
Nashville, Tennessee 37219

Mr. Gustave A. Linenberger
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Marshall E. Miller, Esq.
Chairman
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Luther M. Reed, Esq.
Attorney for the City of
Oak Ridge
253 Main Street, East
Oak Ridge, Tennessee 37830

Anthony Z. Roisman, Esq.
Roisman, Kessler and Cashdan
1025 15th Street, NW
5th Floor
Washington, DC 20005

Dr. Cadet H. Hand, Jr., Director
Bodega Marine Laboratory
University of California
Post Office Box 247
Bodega Bay, California 94923

Lewis E. Wallace, Esq.
Division of Law
Tennessee Valley Authority
Knoxville, Tennessee 37902

STANDARD DISTRIBUTION LIST

Mr. R. Balent, Vice President (2)
Atomics International Division
Rockwell International
P. O. Box 309
Canoga Park, California 91304

Mr. E. C. Brolin, Project Manager (2)
CRBRP Project
Burns and Roe, Inc.
700 Kinderkamack Road
Oradell, New Jersey 07649

Mr. Lochlin W. Caffey, Director (2)
Clinch River Breeder Reactor Plant Project
P. O. Box U
Oak Ridge, Tennessee 37830

Mr. James R. Chapman (2)
Senior Project Manager
Stone & Webster Engineering Corporation
P. O. Box 811
Oak Ridge, Tennessee 37830

Mr. R. W. Cochran, Chief (2)
CRBRP Program Management Branch
Division of Reactor Development
and Demonstration
U. S. Energy Research and
Development Administration
Washington, D. C. 20545

Mr. T. J. Iltis (1)
Senior Site Representative
U. S. Energy Research and Development
Administration
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, Pennsylvania 15663

Dr. W. M. Jacobi, Project Manager (2)
Clinch River Breeder Reactor Plant
Westinghouse Electric Corporation
Advanced Reactors Division
Clinch River Site
P. O. Box W
Oak Ridge, Tennessee 37830

Mr. J. E. Nolan (2)
Project Manager
CRBRP Reactor Plant
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, Pennsylvania 15663

Mr. John F. Pearson, Jr. (1)
Resident Manager
Burns & Roe, Incorporated
P. O. Box T
Oak Ridge, Tennessee 37830

Dr. R. E. Skavdahl, Manager (2)
Clinch River Project
General Electric Company
P. O. Box 5020
Sunnyvale, California 94086

Mr. J. H. Williams, Supervisor (2)
LMFBR Project Section
Tennessee Valley Authority
1300 Commerce Union Bank Building
Chattanooga, Tennessee 37401

LICENSING DISTRIBUTION

Mr. G. Williams, Jr.
Manager of Power
Tennessee Valley Authority
831 Power Building
Chattanooga, TN 37401

Mr. K. Winkleblack
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

Dr. Jeffrey H. Broido, Manager
Analysis and Safety Department
Gas Cooled Fast Reactor Program
Post Office Box 81608
San Diego, CA 92138

AMENDMENT 34

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

This thirty-fourth amendment to the Clinch River Breeder Reactor Plant PSAR includes responses to NRC requests for additional information contained in the NRC letters as listed below. Updated design information is also included in the amendment. Vertical margin lines on the right hand side of the page are used to identify changes resulting from NRC questions, and margin lines on the left to identify new design information.

A page replacement guide appears following the list of responses to NRC Questions.

List of Responses to NRC Questions

Reference: NRC letter dated October 29, 1975

NRC Ques. No.
Q001.276

Reference: NRC letter dated December 30, 1975

NRC Ques. No.
Q001.409

Reference: NRC letter dated August 17, 1976

NRC Ques. No.	NRC Ques. No.
Q011.22	Q130.69
Q110.60	Q241.124
Q130.68	

Reference: NRC letter dated December 1, 1976

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
Q001.588	Q130.80	Q130.96
Q001.591	Q130.84	Q130.99
Q001.608	Q130.85	Q411.16
Q001.609	Q130.91	Q411.17

NRC Ques. No.	NRC Ques. No.
Q411.18	Q411.21
Q411.19	Q411.22
Q411.20	Q411.23
	Q411.24

PAGE REPLACEMENT GUIDE FOR

AMENDMENT 34

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 34 to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket No. 50-537. Amendment 34 consists of new and replacement pages for the PSAR text and Question/Response Supplement pages.

The following attached sheets list Amendment 34 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

Amendment 34

List of Responses to NRC Questions

Reference: NRC letter dated October 29, 1975

NRC Ques. No.
Q001.276

Reference: NRC letter dated December 30, 1975

NRC Ques. No.
Q001.409

Reference: NRC letter dated August 17, 1976

NRC Ques. No.	NRC Ques. No.
Q011.22	Q130.69
Q110.60	Q241.124
Q130.68	

Reference: NRC letter dated August 17, 1976

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
Q001.588	Q130.80	Q130.96
Q001.591	Q130.84	Q130.99
Q001.608	Q130.85	Q411.16
Q001.609	Q130.91	Q411.17

NRC Ques. No.	NRC Ques. No.
Q411.18	Q411.21
Q411.19	Q411.22
Q411.20	Q411.23
	Q411.24

AMENDMENT 35

List of Responses to NRC Questions

Reference: NRC letter dated June 5, 1975

NRC Ques. No.
Q001.236

Reference: NRC letter dated October 6, 1975

NRC Ques. No.
Q5

Reference: NRC letter dated October 29, 1975

NRC Ques. No.	
Q110.21	Q110.37
Q110.25	Q110.38
Q110.32	

Reference: NRC letter dated December 30, 1975

NRC Ques. No.
Q001.432

Reference: NRC letter dated August 17, 1976

NRC Ques. No.
Q110.61
Q110.71

Reference: NRC letter dated December 1, 1976

NRC Ques. No.		
Q001.583	Q001.586	Q001.594
Q001.584	Q001.587	Q324.11
Q001.585	Q001.592	Q324.12
		Q324.13
		Q412.4

UNITED STATES
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
CLINCH RIVER BREEDER REACTOR PLANT PROJECT OFFICE
P. O. BOX U
OAK RIDGE, TENNESSEE 37830

Docket No. 50-537 File: 05.10 807

March 11, 1977

Mr. Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Boyd:

AMENDMENT NO. 36 TO THE PRELIMINARY SAFETY ANALYSIS REPORT FOR CLINCH RIVER BREEDER REACTOR PLANT

The application for a Construction Permit and Class 104(b) Operating License for the Clinch River Breeder Reactor Plant, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment No. 36 to the Preliminary Safety Analysis Report pursuant to 50.34(a) of 10 CFR Part 50. This Amendment 36 to the PSAR includes: responses to NRC requests for additional information contained in letters dated December 30, 1975, and August 17 and December 1, 1976, and other updates and changes.

A Certificate of Service, confirming service of Amendment No. 36 to the PSAR upon designated local public officials and representatives of the EPA, will be filed with your office after service has been made.

Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

Anthony R. Buhl
Assistant Director for Public Safety

PS:L:77:240

Enclosure

cc: Service List
Standard Distribution
Licensing Distribution

SUBSCRIBED and SWORN to before me this 4th day of March, 1977.

Notary Public

My Commission Expires May 21, 1980

PAGE REPLACEMENT GUIDE FOR

AMENDMENT 36

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 36 to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket No. 50-537. Amendment 36 consists of new and replacement pages for the PSAR text and Question/Response Supplement pages.

The following attached sheets list Amendment 36 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

AMENDMENT 36

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

This thirty-sixth amendment to the Clinch River Breeder Reactor Plant PSAR includes responses to NRC requests for additional information contained in the NRC letters as listed below. Updated design information is also included in the amendment. Vertical margin lines on the right hand side of the page are used to identify new design information.

A page replacement guide appears following the list of responses to NRC questions.

List of Responses to NRC Questions

Reference: NRC letter dated August 17, 1976

NRC Ques. No.
Q310.43
Q310.46
Q310.48

Reference: NRC letter dated October 6, 1976

NRC Ques. No.
Q2

Reference: NRC letter dated October 29, 1976

NRC Ques. No.
Q001.274

Reference: NRC letter dated December 1, 1976

NRC Ques. No.
Q001.582
Q422.3

Reference: NRC letter dated December 30, 1976

NRC Ques. No.
Q040.4

AMENDMENT 36

List of Responses to NRC Questions

Reference: NRC letter dated August 17, 1976

NRC Ques. No.
Q310.43
Q310.46
Q310.48

Reference: NRC letter dated October 6, 1976

NRC Ques. No.
Q2

Reference: NRC letter dated October 29, 1976

NRC Ques. No.
Q001.274

Reference: NRC letter dated December 1, 1976

NRC Ques. No.
Q001.582
Q422.3

Reference: NRC letter dated December 30, 1976

NRC Ques. No.
Q040.4

Question 2

Provide a description of the Cumulative Damage Function concept, its application to CRBRP and the supporting data base. Estimates of uncertainties involved and a sample calculation should be included.

Response:

The information requested has been provided under separate cover in June 1976 as topical report, CRBRP-ARD-0115, "The Development and Application of a Cumulative Mechanical Damage Function for Fuel-Pin Failure Analysis in LMFBR Systems."

Question 4

Provide a description of pertinent experimental data, analytical models and plans for obtaining future data regarding fuel rod bowing.

Response:

The information requested has been provided in the topical report, "Fuel Rod Bowing," WARD-D-0150, dated November, 1976.

Question 5

Provide a description of pertinent experimental data, analytical models and plans for obtaining future data regarding fuel rod vibrations.

Response:

The information requested has been provided under separate cover in the topical report WARD-D-0166, "Flow Induced Vibration of Fuel Rods in CRBRP."

Question 7

Provide an overview of the methods used to evaluate the structural integrity of the fuel assembly including a description of all analytical methods used (e.g., PECT2) and all applicable data. The report should be in the form of a summary addressing all calculational limits (e.g., stress and deflection).

Response:

The information requested will be incorporated into the following topical reports which will be available in June 1978:

"CRBRP Fuel Rod Structural Evaluation"

"CRBRP Fuel Assembly Structural Evaluation"

Table of Contents for these reports were provided to NRC in December 1976.

Question 12

Fuel Rod Failure Criteria

Provide a detailed discussion regarding the applied understanding of the loss of fuel rod integrity including the predicted conditions for loss of integrity and the various consequences (e.g., failure location, failure sequence and fuel motion). Include in the all applied criteria (e.g., fuel damage parameter, enthalpy) and their theoretical and empirical bases.

Response:

Fuel rod failure during an HCDA most probably results from a combination of mechanisms. These postuled failure mechanisms include differential thermal expansion between fuel and clad, fission gas release from the fuel, transient induced fuel swelling, and clad melting. The importance of each failure mechanism is dependent on the nature of the transient and the physical characteristics of both fuel and cladding (including design and irradiation history). A detailed discussion of the possible interaction of these various failure mechanisms during a transient overpower accident is presented in Reference 1.

The application of postulated fuel rod failure mechanisms to the development of fuel rod failure models for HCDA analysis has followed two approaches. The first approach has been the development of mechanistic models which assume that one or more failure mechanisms dominate the failure process. The second approach has attempted to develop empirical failure correlations.

The burst pressure failure model in the SAS3A code was the principal failure model used in the analysis of TOP and sodium-in LOF accidents reported in Appendix F (see Section F6.2.3.2.1 and F6.2.3.3). The burst pressure criteria considers the mechanism of pressure loading of the fuel and clad due to fission gas release to predict failure based on temperature dependent clad failure stress data. The pressure in the fuel cavity is assumed to load the clad through the fuel whose strength is neglected. The burst pressure criteria is most applicable to irradiated fuel with a central cavity in which sufficient fission gas has been released to pressurize the cladding during the transient.

In fresh fuel, little fission gas has been released. Hence, failure is expected to be dominated by differential fuel-clad thermal expansion or substantial clad over heating. Therefore, a cladding plastic strain failure criteria was employed in TSDRIV for a parametric study (see Section F6.2.4.3.2.3 in Appendix F) to model fresh pin failure. The cladding plastic strain failure criteria predicts failure based on the mechanism of differential fuel-clad thermal expansion as calculated by the DEFORM subroutine in the SAS3A code

Whereas the burst pressure failure criteria and the clad plastic strain failure criteria each attempt to model specific physical failure mechanisms, the second approach has developed empirical failure correlations. The Damage Parameter failure criteria developed by HEDL for use in FTR TOP analysis is the best empirical fast reactor fuel pin failure correlation currently available.

A complete description of the theoretical and emperical bases for the Damage Parameter and the method in which it can be applied to predict pin failure is presented in References 1 and 2. Damaged parameter calculations were performed to predict the failure of selected pins during a 50 cents/sec transient in the CRBR BOEC and EOEC codes. A comparison of Damage Parameter failure predictions with failures predicted by the burst pressure failure criteria indicated that in general the HEDL correlation predicted slightly earlier failure times, smaller melt fractions and similar axial failure locations in the core. Full details of the comparison are presented in Sections F6.2.4.3.2 and F6.2.4.4.2.3. The good agreement between the Damage Parameter and burst pressure failure predictions provides further support to the position that the urst pressure failure criteria provides reasonable estimates of the failure time and axial location.

The energetics which would result from the Damage Parameter failure predictions were judged to be on the same order of magnitude as predicted by the use of the burst pressure criteria in the SAS analysis of the transient.

Recognizing that there is significant uncertainty associated with the use of theoretical or limited empirical failure models, parametric calculations to envelope the energetics resulting from the most adverse failure location, the core midplane, have been completed. The midplane failure energetics calculated in Section D5.1.4.3 and Section F6.2.4.4.2.4 are well within the structural design basis. Since the energetics resulting from the most adverse pin failure location have been found to be within the structural design basis, the fuel rod failure criteria used in the CRBR HCDA analysis is judged to be a critical item in the calculation of maximum HCDA accident energetics.

References

1. J.H. Scott, R.E. Baars, G.E. Culley and C.W. Menger, "Microstructural Dependence of Failure Threshold in Mixed Oxide LMFBR Fuel Pins", HEDL-TME 75-9, October 1974, pp. 7-38.

2. A.E. Waltar, N.D. Wilburn, D.C. Molesar, L.D. O'Dell, A. Padilla, Jr., L.N.Steward, W.L. Partain, "An Analysis of the Unprotected Transient Overpower Accident in the FTR", HEDL-TME 75-50, June 1975, pp. 3-21 to 3-29.

Question 14

To a great extent, the steady state design is predicated upon an understanding of the exposure dependent cladding deformations of irradiation induced swelling and creep mechanisms. Provide a summary description of the applied understanding of this phenomenon including an identification of its limitations and bases for its adoption.

Response:

The information requested has been provided under separate cover in document WARD-D-0146, "Exposure Dependent Cladding Deformations."

Question 15

Provide a discussion which evaluates and establishes the validity of design criteria, methodology, failure modes, failure criteria, and performance of cell liners proposed for CRBRP. Include in your discussion the consideration given to liner performance criteria and to construction features. Specific information is required on the following structural aspects of potential sodium spills: equipment cell liner response and design, sodium concrete reactions and structural properties of concrete at elevated temperatures.

Response:

The requested information has been included in the cell liner report and Third Level Thermal Margin report. These reports were transmitted to NRC in May 1976 and April 1976, respectively.

Question 16 (D5 and F6.2)

Provide a complete description of the PRIMAR-II and FCI routines used in the SAS code. Include in the discussion the details of the mathematical models, computer logic description and numerical results. Regarding cladding relocation experiments, provide the results of experiments conducted to date (including all relevant experimental conditions) the pre-measurement analysis, if any, and a schedule of additional work planned. Include in your discussion of cladding relocation the impact of the experimental results on CDA analyses by addressing the implications of and your conclusions for the experimental work performed to date. Provide similar information in the area of fuel motion (especially the basis for the SLUMPY model).

Response:

The requested information on the PRIMAR-II and FCI subroutines of SAS3A has been provided to the NRC in References 16-24 and 16-25, respectively.

The experimental evidence on the subject phenomena of cladding and fuel relocation during loss of flow events will be discussed separately. However, it should be borne in mind that these phenomena can be coupled through the reactor thermal-hydraulic and neutronic design characteristics. Thus, to address this topic for the CRBRP additional experimental information related to sodium voiding and two-phase phenomena, as well as physics measurements on material displacement and neutronic effects will be introduced.

Cladding relocation experiments have been performed both in and out-of-pile for a range of thermal-hydraulic conditions. These experiments consist of the TREAT reactor L (Ref. 16-1 through 16-4), R (Ref. 16-5 through 16-7) and F (Ref. 16-8) series and the out-of-pile Argonne flooding and entrainment measurements (Ref. 16-9).

Of equal importance are the complementary set of OPERA (Ref. 16-9) and TREAT R series sodium voiding and two-phase flow regime measurements. These experiments substantiate the general conceptions of one and two-dimensional sodium voiding behavior under LMFBR thermal-hydraulic conditions and their representation within the SAS code (Ref. 16-10, 16-5, 16-9) for accident analysis. Thus, the initial conditions and forces which affect cladding relocation can be reasonably established.

Results from most of these experiments have been documented in the open literature and specifically described to Regulatory personnel at the ERDA sponsored Base Technology Briefings (Ref. 16-11, 16-12) during 1974. Tables 16-1 and 16-2 list those experiments considered pertinent to cladding relocation dynamics and indicate new data which were not covered in the Base Technology Briefings.

It should be noted that a large field of experimental data on material properties, flow dynamics and stability criteria, sodium superheat and boiling phenomena, etc., is being drawn upon to support and form the basis for interpretation of the experiments. This supportive literature is documented in the primary references used herein. Further, the ongoing experiments in foreign countries (e.g., Ref. 16-13, 16-14) are not being addressed due to a lack of detailed knowledge of their experimental conditions and results. Such information will be applied within the CRBRP as possible.

The key tests and conclusions which were drawn in regard to cladding motion in the CRBRP are discussed in what follows. The impact of these conclusions is best represented by examples from Appendix F which provides a best estimate scenario of these events.

The OPERA seven pin experiments and supporting calculations are employed to conclude that the characteristics of sodium voiding in an LMFBR 217 pin fuel assembly will be (Ref. 16-9):

- Initial vapor formation at low or zero bulk superheat ($<10^0$C) near the assembly outlet, in its interior subchannels.
- Inlet flow reversal will occur with limited upstream voiding of the assembly due to pressurization of the vapor producing region (even with subcooled flow channels existing at the assembly radial boundary).
- Upstream voiding of the assembly is related primarily to heat capacity constraints.
- A liquid sodium film remains on the cladding with an average subchannel void fraction of approximately 85-90 percent.
- At LMFBR power densities, liquid sodium is not expected to re-enter the flow channel after voiding in the LOF sequence.

Comparisons of the OPERA results with R series measurements and SAS calculations are satisfactory (Ref. 16-5, 16-15). The SAS analytic model has been shown to employ an incorrect two-phase flow regime prior to inlet flow reversal but the effects of this in a 217 pin assembly will be quite minimal.

Additionall, the experiments and three dimensional calculations show that at the time of sodium voiding (for pump coastdown rates similar to CRBRP, a radial distribution of cladding temperatures exists in the assembly due to overcooling in the outer three rings of pins. Quantitatively, approximately half of the pins will have cladding which, radially, will be 200-400°F colder than the central assembly pin cladding (Ref. 16-9).

Thus, these basic sodium voiding experiments and calculations with SAS and other codes provide confidence in predicting the initial conditions in the fuel assembly at the time of cladding melting. Another basic factor, which is reactor dependent, is the neutronics effect of sodium voiding the active core region. Neutron leakage and spectral effects will change the Doppler coefficient and the net multiplication factor resulting in a change in the reactor power.

The reactor power is important because it determines the axial location of initial clad melting and the time interval available to obtain substantial fuel melting at the same location. For the CRBRP, the reactor power is from one to four times nominal during the cladding melting (0.5 to 2 seconds) prior to significant fuel melting (CRBRP PSAR, Appendix F, Section F6.2.4). The basic L and R series experiments (L2, 3 4 and R3, 5, 5, 6) were performed at approximately nominal power (150-190 watt/gm of fuel). Experiments L5 and R7 were conducted with power bursts to 5 and 16 times nominal at the SAS pre-calculated occurrence of clad and substantial fuel melting, respectively. These experiments are more applicable to combined fuel and cladding motions. Post-test analyses and examinations are still in progress such that detailed conclusions cannot be drawn at this time. A substantial experimental and analytical effort is being made to verify the neutronics effects due to sodium voiding, and steel and fuel relocation in the CRBRP core (Ref. 16-16).

The Argonne experiments on gas-liquid film concurrent flow stability are key results in extending available information from traditional engineering fluids (e.g., air-water) to LMFBR materials. These experiments resulted in physical models for criteria and quantitative data on inception of flooding and entrainment in a liquid metal film-gas flow system, which by dimensional similitude, can be applied to the liquid steel-sodium vapor flow system (Ref. 16-17).

These physical ideas were incorporated into the CLAZAS subroutine of the SAS code and successfully applied in pre and post-test analyses of the R series experiments to predict cladding relocation (Ref. 16-15). The principal error in the calculations was to overpredict the upward plugging capacity of steel. The principal experiment with regard to steel relocation is R5, in which the TREAT energy was terminated after clad melting but prior to significant fuel meltings based upon SAS pre-test predictions. Post-test examination revealed some upper steel blockage and deposition in the unheated section, but not as massive as the SAS prediction (Ref. 16-18) and confirmed a large pre-test predicted steel blockage below the heated section. Bearing in mind that the R-series test section is designed to reflect a flat radial temperature profile (models inner region of LMFBR fuel assembly), it was concluded from these R-series tests, the flooding experiments and the SAS calculations that the basic phenomena of clad relocation in a one-dimensional, radially coherent, test section could be adequately described with SAS.

A final conclusion relative to CRBRP, which is made from all of the above relates to a prior statement that the CRBRP fuel assembly is <u>not</u> radially coherent at the inception of cladding melting, and that the flooding process is known to result in a concentrated pressure drop. Thus, at the onset of cladding flooding in the central region, the local pressure drop is greatly increased and the sodium vapor flow will be diverted to the outer assembly regions, leading to an unflooding of the central region. This process is considered to result in an axial oscillatory motion of cladding until addi0 additional radial melting occurs. The time interval for the latter to occur is equivalent to the onset of fuel motion. Therefore, in the Appendix F base case analyses, the cladding and fuel were considered to relocate together.

Additional out-of-pile experiments are planned for FY77 using a 28 pin, full length pie sector representation of a full fuel assembly with radial melting effects to verify these conclusions on cladding relocation.

Because these supporting experiments were not yet concluded, analyses were presented in the PSAR (Appendix F6.2.4) to reflect possible non-confirmatory experimental results. The CLAXAS one-dimensional model was allowed to relocate steel based upon sodium vapor drag forces. Additionally, the sodium vapor and steel were purposely decoupled and the steel allowed to drain under gravity forces. These cases resulted in increased thermal energy generation relative to the base cases, but were still reasonably within the Structural Design Basis Loadings.

As previously stated, the motions of cladding and fuel are coupled via the reactor design thermal hydraulic and neutronic conditions. Thus, the experiments discussed in regard to cladding relocation (and listed in Tables 16-1, 16-2) all contribute toward establishing the thermal hydraulic and neutronic conditions at inception of fuel relocations. Most of the experimental measurements already discussed continued into the fuel relocation regime. Two out-of-pile information sources, not yet discussed, are the HEDL tungsten heated capsule (Ref. 16-19) presented in a Regulatory Staff Briefing (Ref. 16-20) and the Argonne Direct Electrical Heating (DEH) apparatus, described in Reference 16-21.

Discussions of the key test results and conclusions which were used for the CRBRP PSAR analyses follow. Again, Appendix F analyses best reflect the conclusions drawn from the experiments.

The HEDL tungsten capsule irradiated fuel test series demonstrated physical effects of gas release and resulted in the following HEDL conclusions (Ref. 16-19):

1) Significant fission gas release occurs from solid fuel at temperatures of approximately $\geq 2000^{\circ}C$.

2) Fuel fragmentation can occur in high gas content regions during rapid heating ($250^{\circ}C$/sec).

3) Extensive gas bubble precipitation and coalescence can occur between $2500^{\circ}C$ and melting for time periods on the order of 1 to 2 seconds.

4) Gas release between $2000^{\circ}C$ and melting can produce fuel motion, i.e., dispersal for high heating rates and/or high gas concentrations, or fuel swelling and foam formation for low heating rates and/or low gas concentrations.

The impact of the above on the CRBRP analyses and the SLUMPY subroutine (Ref. 16-22) was the explicit representation of time-dependent gas release estimates (correlated, in part, to experiment FGR-15) and the drag forces resulting from the slip of the gas past fuel particles. In irradiated fuel channels, the mechanism tends to predict dispersal of the fuel after "slumping" initiation from its original axial location.

The ANL DEH apparatus is being employed to determine the factors influencing the onset of fuel relocation, pellet column integrity and cladding effects (Ref. 16-23). These experiments are expected to provide verification of current CRBRP modeling and aid in the interpretation of in-pile experiments.

The slow collapse of fresh fuel modeled in the CRBRP PSAR is based in part on out-of-pile observations and more directly on hodoscope data from the TREAT L2 experiment. At the inception of fuel motion (hodoscope), the experimental data were interpreted to indicate an upper steel blockage with fluted-tube melting. The hodoscope shows a "plastic" (i.e., slow, 10 cm/sec) complete collapse of the upper third of the fuel. A quiescent period of approximately four seconds was abruptly ended by a eructation of fuel upward and outward (∿50 cm/sec). The upward displaced material partially fell slowly back down to be followed by another eructation some two seconds after the first and before TREAT power termination (Ref. 16-1).

Such eructations have not been seen during out-of-pile experiments with fresh fuel, in the R6 hodoscope data, nor in the R4 test data (Ref. 16-5, 16-6). However, these R-series experiments are still in post-test destructive examination and conclusions on the mechanisms of such eructations cannot yet be inferred by the Project.

TREAT experiment R7 examined the effect of a power burst (16 x nominal) during the LOF sequence. The burst was set by pre-SAS analysis and R6 data to occur during fuel melting but prior to fuel relocation. Preliminary results (Ref. 16-7) indicated an inlet blockage formation during the burst and fuel dispersal into the upper reflector region. This is in contrast to the preliminary hodoscope data from R6 (constant power) which indicated a gradual downward collapse of fuel.

The primary experiments on irradiated fuel relocation are the TREAT L and F series. In particular, L3, L4 and F1 are at constant power, while L5 and F2 include power bursts. The experimental conditions and fuel employed are characterized in Table 16-1. Only L3 and L4 are considered to be completely analyzed at this time. These experiments have been employed in conjunction with the conclusions stated earlier from the HEDL tungsten capsule experiments to model the fuel behavior for the CRBRP PSAR.

Hodoscope data from L3 and L4 indicate very sudden, rapid fuel dispersal (eructations). In contrast to L2, prior downward slumping was not seen in the irradiated fuel and material did not fall back down after dispersal. It was the opinion of the experimentors that the mechanism for fuel eructations is the entrapment and vaporization of steel. However, rapid fuel dispersals were seen in the HEDL tests where steel was not present (Ref. 16-19) and eructations were not indicated in either the R series or the F1, single pin, experiment. The L5 and F2 experiment results, where steel and fuel are in proximity during rapid fuel melting, are considered too preliminary for inference on this subject.

For the CRBRP PSAR analyses, steel vapor pressures did not play a significant role in fuel relocations.

SAS post-test analyses of the L-series experiments are sparsely documented with respect to cladding and fuel motions. Again, the radial incoherencies (power/flow), short-heated section, and tight flow loop coupling make the L-series test vehicle a difficult analysis for the SAS-SLUMPY assumptions.

The R-series test vehicle provides a cleaner analysis capability and post-test analysis documentation shows good agreement with the R4 post-test neutron radiograph of fuel location (Ref. 16-15). Since the R4 (and R6) post-test destructive exam is still unavailable, conclusions on the SAS analysis would be premature.

It is expected that pre-test analyses of the F series experiments (Ref. 16-22), and the DEH apparatus will provide the most definitive test and development opportunity for the SLUMPY module.

References

16-1. J.G. Eberhart, et al., "Final Report on the L2 Loss-of-Flow Experiment", ANL/RAS 74-10, July 1974.

16-2. E.W. Barts, et al., "Fuel Dynamics Loss-of-Flow Test L3 (Final Report)", ANL/RAS 75-2, January 1975.

16-3. J.G. Eberhart, et al., "Final Report on Test L4, A Loss-of-Flow Experiment", ANL/RAS 75-16, August 1975.

16-4. R. Simms, et al., "Loss-of-Flow Test L5 on FFTF-Type Irradiated Fuel (Interim Report)", ANL/RAS 75-30, August, 1975.

16-5. M.A. Grolmes, et al., "R-Series Loss-of-Flow Safety Experiment in TREAT", Proc. ANS Fast Reactor Safety Meeting, Beverly Hills, Calif., CONF-740401, P. 279, April 1974.

16-6. B.W. Spencer, et al., "TREAT Loss-of-Flow Safety Experiment R6", Trans. ANS, 21, 287, June 1975.

16-7. B.W. Spencer, et al., "TREAT Loss-of-Flow Safety Experiment R7", Trans ANS, 22, 424, November 1975.

16-8. R.G. Palm, et al., "Loss-of-Flow Test F1 on a FFTF-Type Fuel Element", Trans ANS 22, 427, November, 1975.

16-9. H.D. Fauske, et al., "An Assessment of Voiding Dynamics In Sodium-Cooled Fast Reactors", ANL/RAS 74-20, August 1974.

16-10. F.E. Dunn, et al., "The SAS3A LMFBR Accident Analysis Computer Code", ANL/RAS 75-17, April 1975.

16-11. Regulatory Briefing, "Safety Technology Meeting on TREAT Fuel Behavior Experiments", Argonne National Laboratory, August 1, 1974.

16-12. Regulatory Briefing, "Safety Technology Meeting on Coolant and Cladding Dynamics and Thermal Interactions", Bethesda, Maryland, September 24, 1974.

16-13. H. Kwast, "The behavior of Fast Reactor Fuel Pins Under Loss of Coolant Flow Conditions - (RCN), Proc. of Fast Reactor Safety Meeting, Beverly Hills, Calif., CONF-740401, p. 550, April 1974.

16-14. A. Alexas, et al., "Out-of-Pile Simulation of Fuel Pin Behvaior in Sodium Cooled Fast Breeder Reactors by UO_2 Rods Heated by Electrical Current - (KFK)", ibid, p. 535.

16-15. G. Höppner, "SAS3A Analysis of R-Series Experiments", ANL/RAS 74-14, August 1974.

16-16. Critical Experiments and Analysis, Thirteenth Quarterly Report, GEAP 13371-13, January 1975.

References (Cont'd.)

16-17. Ishii and Grolmes, "Inception Criteria for Droplet Entrainment In Two-Phase Concurrent Film Flow", ANL/RAS 74-25, October 1974.

16-18. Reactor Development Program Progress Report, ANL-RDP-43, Section VII, September 1975.

16-19. Stagle, Hinman and Weber, "Experiments on Melting and Gas Release Behavior of Irradiated Fuel", HEDL-TME 74-17, 1974.

16-20. Regulatory Staff Briefing, "Fuel Pin Transient Behavior Technology Applied to Safety Analyses", HEDL-SA-771, Hanford, Wash., November 1974.

16-21. B.J. Wrona, et al., "First Report on Apparatus to Similate In-Reactor Transient Heating Conditions in Oxide Fuel Columns", Nucl. Technology, 20, 114, 1973.

16-22. W.R. Bohl, "SLUMPY: The SAS3A Fuel Motion Model for Loss-of-Flow", ANL/RAS 74-18, August 1974.

16-23. B.J. Wrona, et al., "Thermal-Shock Cracking in UO_2 During Power Transients", and "Transient Response of Clad UO_2" Trans. ANS 22, p. 419 - 421, San Francisco, November, 1975.

16-24. F.E. Dunn, J.R. Travis, L.L. Smith, "The PRIMAR-II Primary Loop Module for the SAS3A Code", ANL/RAS 76-5, March 1976.

16-25. L.L. Smith, "SAS/FCI: The SAS3A Fuel Coolant Interaction Model", ANL/RAS 75-33, December 1975.

Table 16-1

EXPERIMENTS PERTINENT TO CLADDING AND/OR FUEL RELOCATION

(IN-PILE)

EXPERIMENT/ FACILITY	IRRADIATION HISTORY	MICRO-STRUCTURE (Note 1)	NO. PINS	LENGTH (cm)	AVERAGE SPECIFIC POWER (Watt/gm)(Note 2)	COMMENTS
L2 MARK II A	Fresh	---	7	34	190	Cladding blockage indicated prior to fuel motion.
L3 MARK II A	PNL-17 In EBR II 9-11 KW/Ft to 3.5% BU	I	7	34	155	Upper and lower cladding blockages, some fuel above upper incomplete blockage.
L4 MARK II A	NUCMEC-F In EBR II 11-12 KW/Ft to 4.3% BU	H	7	34	171	Substantial steel blockages not considered to exist at time of fuel eructation.
L5 MARK II C (Note 3)	HEDL-59 In GETR 8-11 kw/Ft to 8% BU	I	3	86	190	Five times power burst timed to occur at SAS predicted clad melting.
R3 R Loop	Fresh	---	1	91	161	Test section designed for radial temperature coherency.
R4 R Loop	Fresh	---	7	91	170	Substantial lower steel blockage - no hodoscope results.
R5 R Loop	Fresh	---	7	91	170	Specifically aimed at cladding motion; transient ended based upon SAS pretest prediction of no fuel motion.
R6 R Loop (Note 3)	Fresh	---	7	91	170	Repeat of R4 to get hodoscope results and determine reproducability.
R7 R Loop (Note 3)	Fresh	---	7	91	150	Sixteen times power burst timed to occur at SAS predicted fuel melting.
F1 F Capsule (Note 3)	NUMEC E in EBR II 12.7 KW/Ft to 2.6% BU	H	1	34	204	Dry capsule - only preliminary results.
F2 F Capsule (Note 3)	PNL-17A in EBR II 11.6 KW/Ft to 0.34% BU	H	1	34	225	Dry capsule - only preliminary results. Ten times power burst timed to occur at SAS predicted unrestructured fuel melting.

NOTES

(1) I ≡ intermediate, columnar grains, but without central void;
H ≡ high, full restructuring with developed central void.

(2) Flat top power; equivalent power for out-of-pile experiments.

(3) Experiment not covered in 1974 Regulatory Briefings.

33

Table 16-2

EXPERIMENTS PERTINENT TO CLADDING AND/OR FUEL RELOCATION

(OUT-OF-PILE)

EXPERIMENT	OBJECTIVE	TEST CONDITIONS	AVERAGE SPECIFIC POWER (watt/gm)(Note 1)	COMMENTS
OPERA	Sodium boiling characteristics and two-phase flow regimes in prototypic geometry.	Full length thermal-hydraulic similarity with internal region of CRBRP/FFTF fuel assembly. (Seven pin triangular array)	~170	Well instrumented for experimental objective. Shares similarity with R-series test section. Demonstrates low to zero bulk superheat, annular-slug flow transition at flow reversal, and average void fractions of 80-90 percent.
Clad Relocation	Determine two-phase flooding and entrainment criteria in gas-liquid film system.	Single and multipin geometry. Internal steam heating of Woods metal "cladding" sections with Angon gas flow.	(Note 3)	Visible, photographic record of two-phase flow conditions. Demonstrates gravity drainage, flooding and entrainment flow conditions. Twenty-eight pin test planned to verify radial, non-coherent cladding flooding - flow effects.
CRBR Engineering Mockup Critical (EMC)-ZZPR 5 (Note 2)	Measure reactivity characteristics of CRBRP design configuration and HCDA related gross material displacements. Verify and improve analytic methods employed in design and safety reactivity calculations.	Zero power critical facility	---	Beginning and end of equilibrium cycle control system influences represented. Gross displacements of sodium, cladding and fuel to support PSAR and FSAR calculations. Analysis of data in progress.
HEDL Tungsten Capsule	Determine effects of rapid heating on irradiated fuel, including fission gas release and motions.	Single pin, uncladded fuel segments (~2 inches high) are heated by radiation from the tungsten wall at 50-250°C/sec. Static flow environment.	17-87	Demonstrated fission gas release from solid fuel and effects of gas on fuel motion pre and post melting. Fuel relocation continuously observed via gamma radiography.
ANL DEH (Note 2)	Determine consequences of fresh and irradiated fuel stack heating to failure.	Single fuel pellet stacks with possible quartz "cladding" heated by electrical current. Vertical gas flow for cooling.	100-750	Visual, photographic record available of fuel conditions. Demonstrated column integrity dependence on transient heating cracking, effect of radial cladding restraint.

(1) Flat top power; equivalent power for out-of-pile experiments.

(2) Experiment not covered in 1974 Regulatory Briefings

(3) Estimated maximum of 12.8 KW/Ft.

Question 17

Provide a discussion of the reactivity effects of bubble collapse (Behren's effect) on CDA energetics as calculated by the VENUS II disassembly code.

Response:

The information requested has been provided under separate cover in a document entitled, "An Analysis of the Reactivity Effects of Bubble Collapse in a Boiled-up Molten Pool in CRBRP."

Question 18

Provide a complete description of the DEMO Code including details of the mathematical models, the equations used, subroutines and supporting data. Provide a discussion of the limitations of the code, and indicate the method, if any, used for validation.

Response

A complete description of the DEMO Code, including details of the mathematical models, the equations used, subroutines and supporting data is herein provided through the Model Description Report, WARD-D-0005, Rev. 4.

Validation of the DEMO Code has been carried out in the following areas:

1. Natural Circulation
2. Pipe Rupture
3. Normal Scram

The first two of these conditions are severe tests of the performance of the Code which should test to a great degree the limitations of thermal and hydraulic modelling of the DEMO Code and their associated numerical procedures.

Natural Circulation

A comparison of the core hot channel clad temperature for 3 to 3 natural circulation event was made using DEMO and FORE-II computer codes. The results of this comparison showed very good agreement between DEMO and FORE-II for this event. This validation against an independent analysis contributes to verification of the reactor response for low-flow and natural circulation conditions. As a consequence of the importance of natural circulation as a backup decay heat removal path for pony motor flow, a plan has been drafted to further verify CRBRP natural circulation capability, including verification of the DEMO Code natural circulation performance. This proposed program plan has been transmitted to the NRC, June 1976.

Pipe Rupture

Validation of the DEMO Code for primary heat transport pipe rupture analysis has been performed by:

(1) Comparison of DEMO primary loop hydraulics against the Westinghouse Proprietary IANUS code primary loop hydraulics, and

(2) Comparison of DEMO Reactor Thermal Hydraulic response with that of the FORE-II Code.

1. DEMO-IANUS Hydraulics Comparison

A comparison of the reactor inlet flow transient following a primary inlet pipe rupture was made using DEMO and IANUS. The results showed a close comparison which provides a validation of DEMO hydraulic modelling against an independent analysis.

Discharge coefficients for the IANUS pipe rupture analysis have been conservatively derived from the test series of Reference 2. Since the IANUS hydraulics modelling has also been checked against the SEFOR tests for natural circulation conditions, this combination lends confidence to the IANUS, and hence DEMO, hydraulics model validity.

2. DEMO-FORE II Reactor Comparison

A comparison of the reactor exit temperature response of the DEMO and FORE-II codes for a postulated inlet pipe rupture in the primary system was also made. Here again, the results indicated good agreement, thus supporting the validity of DEMO core average thermal-hydraulics and neutronics modelling. The FORE-II core model has also been verified against the IANUS model by the FFTF Project.

Other test work is in the planning phase that is expected to obtain test data usable for pipe rupture analysis verification, specifically:

Primary Piping Crack Opening Tests, (Reference 3)

This test is to support the piping integrity investigation. While it is primarily intended to examine critical pipe wall crack sizes and crack propagation, if any, the proposed test will include high-speed instrumentation. This will allow use of test results to establish effective orifice size for the simulated ruptures. The information obtained will be incorporated into the DEMO pipe rupture analysis to confirm the rupture discharge coefficient modelling.

A further discussion of analysis verification for the pipe rupture case can be found in the response to NRC Question 001.419.

Normal Scram

A comparison of DEMO and FORE-II was also made for the normal scram event. The average channel, peak channel and hot channel were examined. The results indicated that the two codes were in good agreement for the scram transient.

The DEMO Code has also been the subject of a careful independent review by the Argonne National Laboratory. The results presented in

Reference 4 support the validity of the fueled assembly hot channel thermal model.

In summary, the DEMO calculations have been, and will continue to be examined for their appropriateness and accuracy. DEMO results have been compared to other calculations and indirectly to test data. This has given confidence in the code's basic calculational scheme and its ability to represent system and component real or required performance. On a continuing basis, as the design and construction progresses and component and system test data becomes available, applicable portions of the code will be compared to this information.

A discussion of the limitations of the code can be found in Appendix E and applicable sections of the text of WARD-D-0005, Rev. 4. Appendix E addresses the major assumptions of the code, and, where applicable, gives a justification for the assumption. Attachment I to this response summarizes DEMO Code limitations.

REFERENCES

1. Verification of Natural Circulation in Clinch River Breeder Reactor A Plan, submitted to NRC June 1976

2. Idel'chik, I.E., Handbook of Hydraulic Resistance Coefficients of Local Resistance and of Friction, AEC-TR-6630 (page 425).

3. Proposed Primary Piping Carck Opening Tests, ARD DAD 02.07.

4. ANL-CT-75-23, "A Study on the Nodal Approximation of the CRBRP Core Simulation," D. Saphier, L. W. Kirsch, T. P. Mulcahey, January 1975.

DEMO LIMITATIONS

1. Identification:

   DEMO Code, ARD overall plant transient analysis program.

2. Applicability:

   (a) General overall-plant transient analysis.

   (b) Determine forcing boundary value transients to input other detailed subsystem analysis programs.

3. Program Capabilities/Limitations:

   (a) Reactor

      1. Power
         - Fission Power - Point reactor kinetics with 6 groups of delayed neutrons
         - Decay Power - Input tables from separate detailed calculations (WARD-D-0090)

      2. Inlet Plenum
         - Complete mixing of inlet fluid. One node metal-sodium heat transfer

      3. Average Channel(s)
         - Equivalent thermal-hydraulics of active core and radial blanket regions
         - Input and output coupled with rest of model
         - Thermal model - 7 axial nodes, 5 radial nodes in core. Sodium restricted to liquid phase.
         - Flow pressure drop - calculated as function of flow. Based on detailed FLODISC calculations.
         - Constant flow fractions - transient redistribution between core, radial blanket and bypass not considered.

4. Hot Channels

- For core and radial blanket information - output temperatures not coupled to rest of plant.
- Constant power peaking, flow orificing and maldistribution, and pessimistic (to 3 σ level of confidence) thermal properties.
- No inter-assembly flow redistribution, intra-assembly flow mixing, or inter-assembly heat transfer.

5. Natural Head

- Calculated from nodal sodium densities.
- Nodal density calculated from sodium temperature in lower plenum, average core (5 axial nodes), and upper plenum.

6. Upper Plenum

- Stratification - Simple model for low flow and low core exit temperatures.
- Heat Transfer - One metal-sodium heat transfer node.

(b) Sodium Coolant Loops

1. Piping

- Nodalized sequential transport model
- Thermal head sum of nodal density-times-elevation
- Pipe metal heat storage not considered
- Heat losses not considered
- Pipe stratification not considered (other than that inherent in sequential transport model)
- Pipe pressure losses regarded as square-law
- Transition to laminar not considered

2. Pumps

- Pump characteristics fitted for head vs speed, flow, for RNPSH and for torque

- Stopped-rotor loss considered as form loss
- Cavitation modelled as head effect only (without column separation or sonic effects)

3. Check Valve
   - Head vs flow loss characteristic is fitted for valve forward and reversed flow. Includes inherently effect of disc hang angle vs flow but is not dynamic.

4. IHX
   - 45 nodes on each side
   - Thermal head is nodal density times elevation
   - Includes tube metal heat storage
   - Includes shell and structure heat storage
   - Single equivalent channel on each side (hence, flow maldistribution or other non-uniform effects are not depicted).

5. Vessel Levels
   - Reactor and pump tank levels calculated dynamically, including effects of cover gas pressure and overflow/makeup
   - Overflow a function of RV level; makeup at specified rate. Makeup handled as a level effect only; thermal effect not considered.

6. Cover Gas
   - Limited dynamic system model using pressure drop network and specified gas makeup rates.

7. Steam System
   - Heat exchange modules represented by seven nodes on each side. Includes subcooled, nucleate boiling, transition boiling, film boiling and superheat representation on water/steam side. Single equivalent path per side.
   - Water side transport time in pipe not considered.
   - Recirculation pump modelled by simple affinity square-law. Cavitation not modelled. Stopped-rotor loss includes preliminary windmill - effect model.

- Two evaporators in each loop are lumped into an equivalent single model.
- Perfect check valves on water side.
- Feedwater assumed not to mix in drum (full stratification in current model).
- Perfect steam separation model.
- Single momemtum equation for evaporator loop (mass flow assumed same throughout). Similar assumption for superheater.
- Recirculation loop natural driving head is density times elevation around loop.
- Tube metal heat storage included. Shell heat storage neglected.
- Water side pipe heat storage neglected.

Question 19

Provide a complete description of the PECT code including details of the mathematical models, the equations used, subroutines and supporting data. Provide a discussion of the limitations of the code and indicate the method, if any, used for validation.

Response:

The PECT computer code has been deleted from the PSAR in Amendment 27 and will not be used for PSAR safety analysis.

Question 20

Provide a complete description of the SPRAY I code including a detailed description of the Theoretical basis, mathematical models and the experimental verifications, if any, with an identification of the Limiting conditions of applicability.

Response:

This information was provided in response to Question 001.224.

Amendment 37

List of Responses to NRC Questions

Reference: NRC letter dated June 5, 1975

NRC Ques. No.
Q001.174

Reference: NRC letter dated October 6, 1975

NRC Ques. No.
Q9

Reference: NRC letter dated December 30, 1975

NRC Ques. No.
Q040.16

Reference: NRC letter dated August 17, 1976

NRC Ques. No.
Q130.57
Q130.59

Reference: NRC letter dated December 1, 1976

NRC Ques. No.	NRC Ques. No.	NRC Ques. No.
Q001.590	Q130.83	Q130.92
Q001.593	Q130.86	Q130.93
Q130.79	Q130.87	Q130.94
Q130.81	Q130.88	Q130.95
Q130.82	Q130.90	Q130.97
		Q130.98

Amendment 38

There are no new NRC Questions in Amendment 38

UNITED STATES
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
CLINCH RIVER BREEDER REACTOR PLANT PROJECT OFFICE
P. O. BOX U
OAK RIDGE, TENNESSEE 37830

Docket No. 50-537 File: 05.10 812

May 27, 1977

Mr. Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Boyd:

AMENDMENT NO. 39 TO THE PRELIMINARY SAFETY ANALYSIS REPORT FOR CLINCH RIVER BREEDER REACTOR PLANT

The application for a Construction Permit and Class 104(b) Operating License for the Clinch River Breeder Reactor Plant, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment No. 39 to the Preliminary Safety Analysis Report pursuant to 50.34(a) of 10 CFR Part 50. This Amendment No. 39 to the PSAR includes: responses to NRC requests for additional information contained in letters dated August 17, 1976, and March 22, 1977; a revised set of General Arrangement Drawings to Section 1.2; and other updates and changes.

A Certificate of Service, confirming service of Amendment No. 39 to the PSAR upon designated local public officials and representatives of the EPA, will be filed with your office after service has been made.

Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

D R Ashley for

Anthony R. Buhl
Assistant Director for Public Safety

PS:L:77:572

Enclosure

cc: Service List
Standard Distribution
Licensing Distribution

SUBSCRIBED and SWORN to before me this 23rd day of May, 1977.

Inez F. Minor

Notary Public
My Commission Expires May 21, 1980

SERVICE LIST

Atomic Safety & Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Atomic Safety & Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

S. Wallace Brewer, Judge
Office of County Judge
Roane County Court House
Kingston, Tennessee 37763

Dr. Thomas Cochran
Natural Resources Defense Council, Inc.
917 15th Street, N.W.
8th Floor
Washington, D. C. 20005

Docketing & Service Station
Office of the Secretary
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Geoffrey P. Gitner, Esq.
Counsel for NRC Staff
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

William B. Hubbard, Esq.
Assistant Attorney General
State of Tennessee
Office of the Attorney General
422 Supreme Court Building
Nashville, Tennessee 37219

Mr. Gustave A. Linenberger
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Marshall E. Miller, Esq.
Chairman
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Luther M. Reed, Esq.
Attorney for the City of
Oak Ridge
253 Main Street, East
Oak Ridge, Tennessee 37830

Anthony Z. Roisman, Esq.
Roisman, Kessler and Cashdan
1712 N Street, N.W.
Washington, D. C. 20036

Dr. Cadet H. Hand, Jr., Director
Bodega Marine Laboratory
University of California
Post Office Box 247
Bodega Bay, California 94923

Lewis E. Wallace, Esq.
Division of Law
Tennessee Valley Authority
Knoxville, Tennessee 37902

CRBRP CORRESPONDENCE

August 30, 1976

STANDARD DISTRIBUTION LIST

Mr. R. Balent, Vice President (2)
Atomics International Division
Rockwell International
P. O. Box 309
Canoga Park, California 91304

Mr. E. C. Brolin, Project Manager (2)
CRBRP Project
Burns and Roe, Inc.
700 Kinderkamack Road
Oradell, New Jersey 07649

Mr. Lochlin W. Caffey, Director (2)
Clinch River Breeder Reactor Plant Project
P. O. Box U
Oak Ridge, Tennessee 37830

Mr. James R. Chapman (2)
Senior Project Manager
Stone & Webster Engineering Corporation
P. O. Box 811
Oak Ridge, Tennessee 37830

Mr. R. W. Cochran, Chief (2)
CRBRP Program Management Branch
Division of Reactor Development
and Demonstration
U. S. Energy Research and
Development Administration
Washington, D. C. 20545

Mr. T. J. Iltis (1)
Senior Site Representative
U. S. Energy Research and Development
Administration
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, Pennsylvania 15663

Dr. W. M. Jacobi, Project Manager (2)
Clinch River Breeder Reactor Plant
Westinghouse Electric Corporation
Advanced Reactors Division
Clinch River Site
P. O. Box W
Oak Ridge, Tennessee 37830

Mr. J. E. Nolan (2)
Project Manager
CRBRP Reactor Plant
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, Pennsylvania 15663

Mr. John F. Pearson, Jr. (1)
Resident Manager
Burns & Roe, Incorporated
P. O. Box T
Oak Ridge, Tennessee 37830

Dr. R. E. Skavdahl, Manager (2)
Clinch River Project
General Electric Company
P. O. Box 5020
Sunnyvale, California 94086

Mr. J. H. Williams, Supervisor (2)
LMFBR Project Section
Tennessee Valley Authority
1300 Commerce Union Bank Building
Chattanooga, Tennessee 37401

LICENSING DISTRIBUTION

Mr. G. Williams, Jr.
Manager of Power
Tennessee Valley Authority
831 Power Building
Chattanooga, TN 37401

Mr. R. M. Little
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

Dr. Jeffrey H. Broido, Manager
Analysis and Safety Department
Gas Cooled Fast Reactor Program
P. O. Box 81608
San Diego, CA 92138

Dirk Van Hoesen, Licensing Branch, Public Safety Division

Ron Green, Acting Chief, Technology Branch, Engineering Division

Amendment 39

Clinch River Breeder Reactor Plant

Preliminary Safety Analysis Report

(Docket No. 50-537)

This thirty-ninth amendment to the Clinch River Breeder Reactor Plant PSAR includes responses to NRC requests for additional information contained in the NRC letters as listed below, revised sets of General Arrangement Drawings to Section 1.2, and other updates. Vertical margin lines on the left hand side of the page are used to identify new design information or updates to the design. Vertical margin lines on the right hand side of the page are used to identify question response information.

A page replacement guide appears following the list of responses to NRC questions.

List of Responses to NRC Questions

Reference: NRC letter dated August 17, 1976

NRC Ques. No.
310.40
310.51

Reference: NRC letter dated March 22, 1977

NRC Ques. No.
001.610
001.611
011.612
001.613
001.614

PAGE REPLACEMENT GUIDE FOR

AMENDMENT 39

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 39 to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket No. 50-537. Amendment 39 consists of new and replacement pages for the PSAR text and Question/Response Supplement pages.

The following attached sheets list Amendment 39 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

Amendment 39

List of Responses to NRC Questions

Reference: NRC letter dated August 17, 1976

NRC Ques. No.
310.40
310.51

Reference: NRC letter dated March 22, 1977

NRC Ques. No.
001.610
001.611
001.612
001.613
001.614

Amend 39
May 1977

UNITED STATES
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
CLINCH RIVER BREEDER REACTOR PLANT PROJECT OFFICE
P. O. BOX U
OAK RIDGE, TENNESSEE 37830

Docket No. 50-537 File: 05.10

July 15, 1977

Mr. Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Boyd:

AMENDMENT NO. 40 TO THE PRELIMINARY SAFETY ANALYSIS REPORT FOR CLINCH RIVER BREEDER REACTOR PLANT

The application for a Construction Permit and Class 104(b) Operating License for the Clinch River Breeder Reactor Plant, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment No. 40 to the Preliminary Safety Analysis Report pursuant to 50.34(a) of 10 CFR Part 50. This Amendment No. 40 to the PSAR includes: responses to NRC requests for additional information contained in letters dated October 7 and December 1, 1976; the annual update to Chapter 17, Quality Assurance, with the exception of Appendix I to Chapter 17 which will be updated in Amendment 41; and other updates and revisions.

A Certificate of Service, confirming service of Amendment No. 40 to the PSAR upon designated local public officials and representatives of the EPA, will be filed with your office after service has been made.

Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

Anthony R. Buhl
Assistant Director for Public Safety

PS:L:77:698

Enclosure

cc: Service List
Standard Distribution
Licensing Distribution

SUBSCRIBED and SWORN to before me this 7th day of July, 1977.

Notary Public
My Commission Expires May 21, 1980

SERVICE LIST

Atomic Safety & Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Atomic Safety & Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

S. Wallace Brewer, Judge
Office of County Judge
Roane County Court House
Kingston, Tennessee 37763

Dr. Thomas Cochran
Natural Resources Defense Council, Inc.
917 15th Street, N.W.
8th Floor
Washington, D. C. 20005

Docketing & Service Station
Office of the Secretary
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Geoffrey P. Gitner, Esq.
Counsel for NRC Staff
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

William B. Hubbard, Esq.
Assistant Attorney General
State of Tennessee
Office of the Attorney General
422 Supreme Court Building
Nashville, Tennessee 37219

Mr. Gustave A. Linenberger
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Marshall E. Miller, Esq.
Chairman
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Luther M. Reed, Esq.
Attorney for the City of
Oak Ridge
253 Main Street, East
Oak Ridge, Tennessee 37830

Anthony Z. Roisman, Esq.
Roisman, Kessler and Cashdan
1712 N Street, N.W.
Washington, D. C. 20036

Dr. Cadet H. Hand, Jr., Director
Bodega Marine Laboratory
University of California
Post Office Box 247
Bodega Bay, California 94923

Lewis E. Wallace, Esq.
Division of Law
Tennessee Valley Authority
Knoxville, Tennessee 37902

CRBRP CORRESPONDENCE

August 30, 1976

STANDARD DISTRIBUTION LIST

Mr. R. Balent, Vice President (2)
Atomics International Division
Rockwell International
P. O. Box 309
Canoga Park, California 91304

Mr. E. C. Brolin, Project Manager (2)
CRBRP Project
Burns and Roe, Inc.
700 Kinderkamack Road
Oradell, New Jersey 07649

Mr. Lochlin W. Caffey, Director (2)
Clinch River Breeder Reactor Plant Project
P. O. Box U
Oak Ridge, Tennessee 37830

Mr. James R. Chapman (2)
Senior Project Manager
Stone & Webster Engineering Corporation
P. O. Box 811
Oak Ridge, Tennessee 37830

Mr. R. W. Cochran, Chief (2)
CRBRP Program Management Branch
Division of Reactor Development
and Demonstration
U. S. Energy Research and
Development Administration
Washington, D. C. 20545

Mr. T. J. Iltis (1)
Senior Site Representative
U. S. Energy Research and Development
Administration
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, Pennsylvania 15663

Dr. W. M. Jacobi, Project Manager (2)
Clinch River Breeder Reactor Plant
Westinghouse Electric Corporation
Advanced Reactors Division
Clinch River Site
P. O. Box W
Oak Ridge, Tennessee 37830

Mr. J. E. Nolan (2)
Project Manager
CRBRP Reactor Plant
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, Pennsylvania 15663

Mr. John F. Pearson, Jr. (1)
Resident Manager
Burns & Roe, Incorporated
P. O. Box T
Oak Ridge, Tennessee 37830

Dr. R. E. Skavdahl, Manager (2)
Clinch River Project
General Electric Company
P. O. Box 5020
Sunnyvale, California 94086

Mr. J. H. Williams, Supervisor (2)
LMFBR Project Section
Tennessee Valley Authority
1300 Commerce Union Bank Building
Chattanooga, Tennessee 37401

LICENSING DISTRIBUTION

Mr. G. Williams, Jr.
Manager of Power
Tennessee Valley Authority
831 Power Building
Chattanooga, TN 37401

Mr. R. M. Little
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

Dr. Jeffrey H. Broido, Manager
Analysis and Safety Department
Gas Cooled Fast Reactor Program
P. O. Box 81608
San Diego, CA 92138

D. Elias, Chief, Technology Branch, Engineering Division

PAGE REPLACEMENT GUIDE FOR

AMENDMENT 40

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 40 to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket No. 50-537. Amendment 40 consists of new and replacement pages for the PSAR text and Question/Response Supplement pages.

The following attached sheets list Amendment 40 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

Amendment 40

List of Responses to NRC Questions

Reference: NRC letter dated October 6, 1975

NRC Ques. No.
Q3

Reference: NRC letter dated October 7, 1976

NRC Ques. No.
001.581

Reference: NRC letter dated December 1, 1976

NRC Ques. No.	NRC Ques. No.
001.595	001.602
001.596	001.603
001.597	001.604
001.598	001.605
001.599	001.606
001.600	001.607
001.601	130.89
	130.100

Amendment 41

List of Responses to NRC Questions

Reference: NRC letter dated October 6, 1975

NRC Ques. No.
Q1
Q8
Q10
Q13

Reference: NRC letter dated August 17, 1976

NRC Ques. No.
310.50

Reference: NRC letter dated March 30, 1977

NRC Ques. No.
001.695

Question 1

A description of the LIFE III Code is required with details of all the submodels contained in the code as well as verification of the code with applicable data. The verification should include a description of, and data for, all the uses to which LIFE III will be put in the safety analysis of the CRBRP fuel. Sample calculations should be included for the type of fuel behavior analyzed for CRBRP under various conditions. Flow charts and brief discussions of the mathematical methods should be included. Topics listed elsewhere in this enclosure for which there is extensive and sufficient description in the LIFE code support documentation need not be presented separately.

Response:

A description of the LIFE III code and its application for CRBRP was provided under seperate cover in the report entitled, "LIFE III Fuel Element Performance Code User's Manual". The verification of the LIFE III code will be documented in Reference 4, 5, and 6 of the user's manual and will be provided to NRC prior to FSAR submittal.

Question 6

Provide a description of pertinent experimental data, analytical models, and plans for obtaining future data regarding fuel rod-wire wrap interactions.

Response:

This information requested has been provided under seperate cover as Topical Report WARD-D-0149,"Wire Wrap-Classing Interaction in LMFBR Fuel Rods." 41

Question 8

Provide a summary description of the various aspects of fuel temperature calculations including a complete listing of all correlations used (fission gas release, gap conductance, fuel relocation, etc.) along with the data base for each correlation and its range of applicability.

Response:

Detailed calculations have been performed in the thermal-hydraulic analyses of the CRBRP core to characterize the thermal behavior of fuel and blanket assemblies. The hot (maximum cladding temperature) rod in each fuel and blanket assembly has been analyzed throughout its equilibrium lifetime, i.e., at the beginning and end of each of three consecutive equilibrium cycles, thus providing a lifetime history of cladding temperature and fission gas pressure necessary for rod structural analyses. Additionally, fuel temperatures and linear power ratings have been calculated at the time of maximum to assure that the limiting power-to-melt is not exceeded, and thus, no fuel centerline melting occurs. Material properties and correlations have been used in the thermal-hydraulic studies to represent the physical behavior of the fuel rod. Since Question 8 was transmitted on October 6, 1975, the analyses have been discussed in detail in PSAR Section 4.4, and Topical Report WARD-D-0054, "Predicted Preliminary Thermal Hydraulic Performance of CRBRP Fuel and Blanket Assemblies," was transmitted to NRC on March 9, 1976.

Rather than collate and repeat what is already reported in the PSAR and WARD-D-0054, the following identifies, for the convenience of the reviewers, the various sections of PSAR pertaining to each of the items in the topical report table of contents.

Fuel Temperature Calculations

PSAR Section 4.4.3.3 provides the step-by-step procedure used to calculate fuel temperatures, plus a discussion of the uncertainties to be considered in the power-to-melt studies.

Analyses

In addition to Section 4.4.3.3, the following PSAR sections apply:

a) 4.4.3.2 provides an overview of the uncertainties analysis plus a detailed discussion of the combination of gap and fuel uncertainties in calculating the fuel centerline temperature and a comparison with HEDL P-19 experimental data;

b). 4.4.3.3.1 discusses in detail fuel thermal conductivity uncertainities;

c). 4.4.3.3.2 provides an overview of uncertainities in gap conductance calculations, which are discusses in greated detail in 4.4.2.6.5; and

d). 4.4.3.3.3 and 4.4.3.3.4 are a discussion of experimental uncertainities in melting temperature and power-to-melt, respectively.

CORRELATIONS AND PROPERTIES

A complete discussion of T&H correlations and properties is presented in PSAR Section 4.4.2.6 as follows:

• Sodium Properties	4.4.2.6.1
• Film Heat Transfer Coefficient	4.4.2.6.2
• Cladding Thermal Conductivity	4.4.2.6.3
• Gap Conductance (this section provides a very detailed discussion of the gap conductance model, uncertainities, limitations and future analyses planned for final design and FSAR)	4.4.2.6.5
• Fuel Thermal Conductivity	4.4.2.6.7
• Fuel Density	4.4.2.5.11
• Fuel Restructing Parameters	4.4.2.6.15
• Fuel Melting Temperature (this section also included a discussion of actinides redistribution and its effects on solidus temperature)	4.4.2.6.12
• Fuel Linear Power-to-Melt (in addition to other sections previously mentioned - 4.4.3.3, 4.4.3.4)	4.4.2.6.14
• Fission Gas Release (which includes a discussion of the fission gas release correlation, its uncertainities and fission yield values)	4.4.2.6.16

RESULTS

Results of the core thermal analyses for fuel and blanket assemblies are reported in Sections 4.4.3.3.5 (for equilibrium core at Plant Thermal-Hydraulic Design conditions), 4.4.3.3.6 (for first core at Plant Thermal-Hydraulic Design conditions), and 4.4.3.3.7 (for equilibrium core at Plant Expected Operating

Amend. 41
Oct. 1977

conditions). These results are also reported in great detail in Topical Report WARD-D-0054.

Conclusion

The various secitons of amended PSAR Section 4.4 listed in the above provide a detailed discussion of the state-of-the-art for the core thermal-hydraulic analyses. Improvements have been and will continue to be developed, e.g., closure, continous fuel restructing models as function of burnup. All these improvements will be factored into the final analyses and reported in the FSAR.

Question 10

Provide a description of fuel/cladding chemical interactions including uniform and intergrannular fission product attack. External cladding wastage in the sodium should also be discussed. Include a description of all experimental data and plans for obtaining data in the future to confirm your conclusions in this matter.

Response:

The information requested has been provided under separate cover in the Topical Report WARD-D-0147, "Internal/External Cladding Degradation".

Question 13

Provide a discussion of continued operation with failed fuel with particular emphasis given to the experimental data base and the extent of reliance on this data.

Response:

Reference 1 provides an analysis of fuel failure propagation and a discussion of operation with failed fuel based on both domestic and foreign experience.

Reference:

1. Letter, A. R. Buhl to R. S. Boyd, "Additional Information on Subassembly Faults," S:L:1628, dated September 20, 1976.

contaminated sodium. In these tests, sodium was doped with from 10 to 233 ppm PuO_2 or Na_4PuO_5 and then ignited in air at temperatures of 500 to 550°C. The aerosol released from this burning pool of sodium was collected and analyzed for plutonium content. For the sodium containing PuO_2, nine experiments resulted in plutonium release fractions ranging from 1×10^{-6} to 3.6×10^{-5}. The average release fraction was 1.1×10^{-5}. For sodium-plutonate, the results indicate the release fractions are several orders of magnitude less than for PuO_2. The application of these data to the evaluation of sodium fire consequences would essentially eliminate the Pu contribution to exposures and thus significantly reduce predicted consequences. However, to insure a conservative assessment of potential radiological consequences, the PSAR analyses include the Pu contribution by pessimistically assuming equal aerosol/liquid concentrations. This assumption overestimates the Pu exposure contribution by several orders of magnitude and consequently provides conservative doses to the limiting organ (i.e. bone) that contain margins for uncertainties in the fractional releases of more volatile species.

To summarize, the PSAR sodium fire analyses are based on the assumption of equal aerosol/liquid radionuclide concentrations. This assumption leads to an overestimate of several orders of magnitude of the contribution of Pu, which is the controlling nuclide dose contributor. This approach results in a conservative assessment of potential radiological consequences and provides ample margin for any uncertainties associated with the preferential release of volatile isotopes. Alternately, the experimental sodium burning-release data could be directly applied to the PSAR sodium fire evaluations. This would result in essentially the complete elimination of the plutonium source term, a factor of three increase in the iodine source, and a factor of ten increase for cesium. These latter two factors have already been shown to be conservative. The overall result of this approach, as shown in the above discussion, would be a net decrease in the controlling off-site exposures, leading to an assessment which is enveloped by that currently presented in the PSAR.

References

Q310.50-1 "Fission Product Retention in Sodium - A Summary of Analytical and Experimental Studies at Atomics International", NAA-SR-11766, 1966, pp. 73-79.

Q310.50-2 Reference Q310.50-1, pp.93-95.

Q310.50-3 Pollock, B.D., Silberberg, M., and Koontz, R.L., "Vaporization of Fission Products From Sodium", ANL-7520, Part I, 1968, pp. 549-554.

Q310.50-4 Saroul, J., "Investigation on the Behaviour of Fission Products in Sodium and Argon - PIRANA Experiments", in Proceedings of the International Conference on the Safety of Fast Reactors, Aix-en-Provence, September 19-22, 1967, pp. Vb-1-1-3.

Q310.50-5 "Summary Report for Laboratory Experiments on Sodium Fires", TR-707-130-007, August 1973, pp. 21-22.

Q310.50-6 Kawahara, S., et. al., "Behavior of Sodium Oxide and Fission Products Under Sodium Pool Fire", in Engineering of Fast Reactors for Safe and Reliable Operation - International Conference, Karlsruhe, Germany, October 9-12, 1973. Vol III, pp. 1406-1422.

Q310.50-7 NAA-SR-12296, "Quarterly Technical Progress Report - AEC Unclassified Programs, October - December 1966", pp. 60-61.

Q310.50-8 E. J. Chatfield, J. Fuel Material, 32, 228 (1969).

Q310.50-9 AI-ERDA-13166, "Quarterly Technical Progress Report: Nuclear Safety, Characterization of Sodium Fires and Fast Reactor Fission Products, October - December 1975", February 15, 1976.

Question 001.695

Your response to Item 001.310 requires clarification. Confirm that the last sentence in the first paragraph of your response may be regarded as a commitment.

Response:

Yes, the Project is committed to provide design modification if necessary to assure common side wall integrity between the IHTS and SGS cells in the event of a sodium spray fire. However, as indicated in the response to Question 001.310, it is anticipated that conservative analyses will demonstrate that the present design is adequate.

Department of Energy
Clinch River Breeder Reactor
Plant Project Office
P.O. Box U
Oak Ridge, Tennessee 37830

881

Docket No. 50-537

File: 05.10

November 4, 1977

Mr. Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Boyd:

AMENDMENT NO. 42 TO THE PRELIMINARY SAFETY ANALYSIS REPORT FOR CLINCH RIVER BREEDER REACTOR PLANT

The application for a Construction Permit and Class 104(b) Operating License for the Clinch River Breeder Reactor Plant, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment No. 42 to the Preliminary Safety Analysis Report pursuant to 50.34(a) of 10 CFR Part 50. This Amendment No. 42 to the PSAR includes changes which revise the reactivity feedback component of the overall power coefficient during a limited portion of the startup range. Under worst-case conditions used for analysis purposes, it is shown that the overall power coefficient in this limited region may be slightly positive. The accompanying change pages present the Project position on this matter. Your review of the Project's position is requested so that early NRC/Project discussion can commence if necessary. Responses to NRC requests for additional information contained in a letter dated March 30, 1977, and other updates and revisions are also included in this amendment.

A Certificate of Service, confirming service of Amendment No. 42 to the PSAR upon designated local public officials and representatives of the EPA, will be filed with your office after service has been made.

Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

Anthony R. Buhl
Assistant Director for Public Safety

PS:L:77:889

Enclosure

cc: Service List
Standard Distribution
Licensing Distribution

SUBSCRIBED and SWORN to before me this 28th day of October, 1977.

Notary Public

My Commission Expires May 21, 1980

SERVICE LIST

Atomic Safety & Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Atomic Safety & Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

S. Wallace Brewer, Judge
Office of County Judge
Roane County Court House
Kingston, Tennessee 37763

Dr. Thomas Cochran
Natural Resources Defense Council, Inc.
917 15th Street, N.W.
8th Floor
Washington, D. C. 20005

Docketing & Service Station
Office of the Secretary
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Geoffrey P. Gitner, Esq.
Counsel for NRC Staff
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

William B. Hubbard, Esq.
Assistant Attorney General
State of Tennessee
Office of the Attorney General
422 Supreme Court Building
Nashville, Tennessee 37219

Mr. Gustave A. Linenberger
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Marshall E. Miller, Esq.
Chairman
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Luther M. Reed, Esq.
Attorney for the City of
Oak Ridge
253 Main Street, East
Oak Ridge, Tennessee 37830

Anthony Z. Roisman, Esq.
Roisman, Kessler and Cashdan
1712 N Street, N.W.
Washington, D. C. 20036

Dr. Cadet H. Hand, Jr., Director
Bodega Marine Laboratory
University of California
Post Office Box 247
Bodega Bay, California 94923

Lewis E. Wallace, Esq.
Division of Law
Tennessee Valley Authority
Knoxville, Tennessee 37902

CRBRP CORRESPONDENCE August 30, 1976

STANDARD DISTRIBUTION LIST

Mr. R. Balent, Vice President (2)
Atomics International Division
Rockwell International
P. O. Box 309
Canoga Park, California 91304

Mr. E. C. Brolin, Project Manager (2)
CRBRP Project
Burns and Roe, Inc.
700 Kinderkamack Road
Oradell, New Jersey 07649

Mr. Lochlin W. Caffey, Director (2)
Clinch River Breeder Reactor Plant Project
P. O. Box U
Oak Ridge, Tennessee 37830

Mr. James R. Chapman (2)
Senior Project Manager
Stone & Webster Engineering Corporation
P. O. Box 811
Oak Ridge, Tennessee 37830

Mr. R. W. Cochran, Chief (2)
CRBRP Program Management Branch
Division of Reactor Development
and Demonstration
U. S. Energy Research and
Development Administration
Washington, D. C. 20545

Mr. T. J. Iltis (1)
Senior Site Representative
U. S. Energy Research and Development
Administration
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, Pennsylvania 15663

Dr. W. M. Jacobi, Project Manager (2)
Clinch River Breeder Reactor Plant
Westinghouse Electric Corporation
Advanced Reactors Division
Clinch River Site
P. O. Box W
Oak Ridge, Tennessee 37830

Mr. J. E. Nolan (2)
Project Manager
CRBRP Reactor Plant
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, Pennsylvania 15663

Mr. John F. Pearson, Jr. (1)
Resident Manager
Burns & Roe, Incorporated
P. O. Box T
Oak Ridge, Tennessee 37830

Dr. R. E. Skavdahl, Manager (2)
Clinch River Project
General Electric Company
P. O. Box 5020
Sunnyvale, California 94086

Mr. J. H. Williams, Supervisor (2)
LMFBR Project Section
Tennessee Valley Authority
1300 Commerce Union Bank Building
Chattanooga, Tennessee 37401

LICENSING DISTRIBUTION

Mr. G. Williams, Jr.
Manager of Power
Tennessee Valley Authority
831 Power Building
Chattanooga, TN 37401

Mr. R. M. Little
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

Dr. Jeffrey H. Broido, Manager
Analysis and Safety Department
Gas Cooled Fast Reactor Program
P. O. Box 81608
San Diego, CA 92138

Dirk Van Hoesen, Licensing Branch, Public Safety Division

Ron Green, Acting Chief, Technology Branch, Engineering Division

Amendment 42

List of Responses to NRC Questions

Reference: NRC letter dated March 30, 1977

NRC Ques. No.	
001.693	001.699
001.694	001.702
001.696	001.703
001.697	001.704
001.698	

Question 001.693

Your response to Item 001.232 appears to be unsatisfactory. We are concerned about the method used of bracketing the venting rate with calculations at 700,000 cfm and zero. At zero venting rate, the pressures would certainly rise (in the abiabatic assumption case) to unsatisfactory levels. On the other hand, in the case of the huge venting rate required to maintain the pressure at atmospheric, the non-radiological toxic effects of sodium oxide should be considered.

Provide an evaluation of the venting rate, the pressure rise in the regions considered, and the toxicity level of the non-radioactive sodium released during the venting process.

Response:

In response to this question, the results of an evaluation performed with the SPRAY 3 code, modified for venting, is presented to establish cell venting rate, cell pressure, and the sodium aerosol release rate as a function of time for the IHTS pipe rupture accident in the intermediate bay (IMB) described in Section 15.6.1.5 of the PSAR. All assumptions regarding cell dimensions, initial cell conditions (e.g. temperature and pressure), spray characterization parameters, and sodium discharge rate are the same as those used in the PSAR analysis. It should be noted that, although the IMB vents to the steam generator bay (SGB), it has been conservatively assumed, from the standpoint of aerosol release that gases and sodium aerosol are vented directly to the environment. Consequently, no credit for attenuation of aerosols in the SGB has been taken. The venting rate for this direct-release flow model is based on the flow equation for an ideal, compressible fluid; a vent area of 12 ft^2 is required to ensure that the cell design pressure of 3 psig is not exceeded. Based on preliminary design information, this vent area is well below the available IMB vent area. The direct release flow model is considered to be realistic for the IMB cell pressurization analysis because the venting capacity of the SGB is an order of magnitude greater than that of the IMB; thus it presents an insignificant increase in flow resistance during IMB venting.

From Figure Qoo1.693-1, it is seen that a peak cell venting rate of 440,000 cfm is reached in about 12 seconds. This is less than the peak value of 700,000 cfm calculated by the preliminary analysis referred to in the question. Consequently, the PSAR analysis provides conservative results for peak cell vent rates. At 30 seconds, the vent rate has decreased an order of magnitude from the peak value. Figure Q001.693-2 shows a peak cell pressure of 2.7 psig which is reached at about 10 seconds.

Responses to NRC Questions 001.233 and 310.51 provide the exclusion boundary levels of sodium hydroxide associated with the postulated IHTS pipe rupture as currently analyzed in PSAR Section 15.6.1.5. That section will be updated in a future amendment to reflect the information contained herein including updated SPRAY 3 analysis. Preliminary indications are that the updated concentrations will involve an approximately 20% reduction in the levels presently provided in the responses referenced.

Amend. 42
Nov. 1977

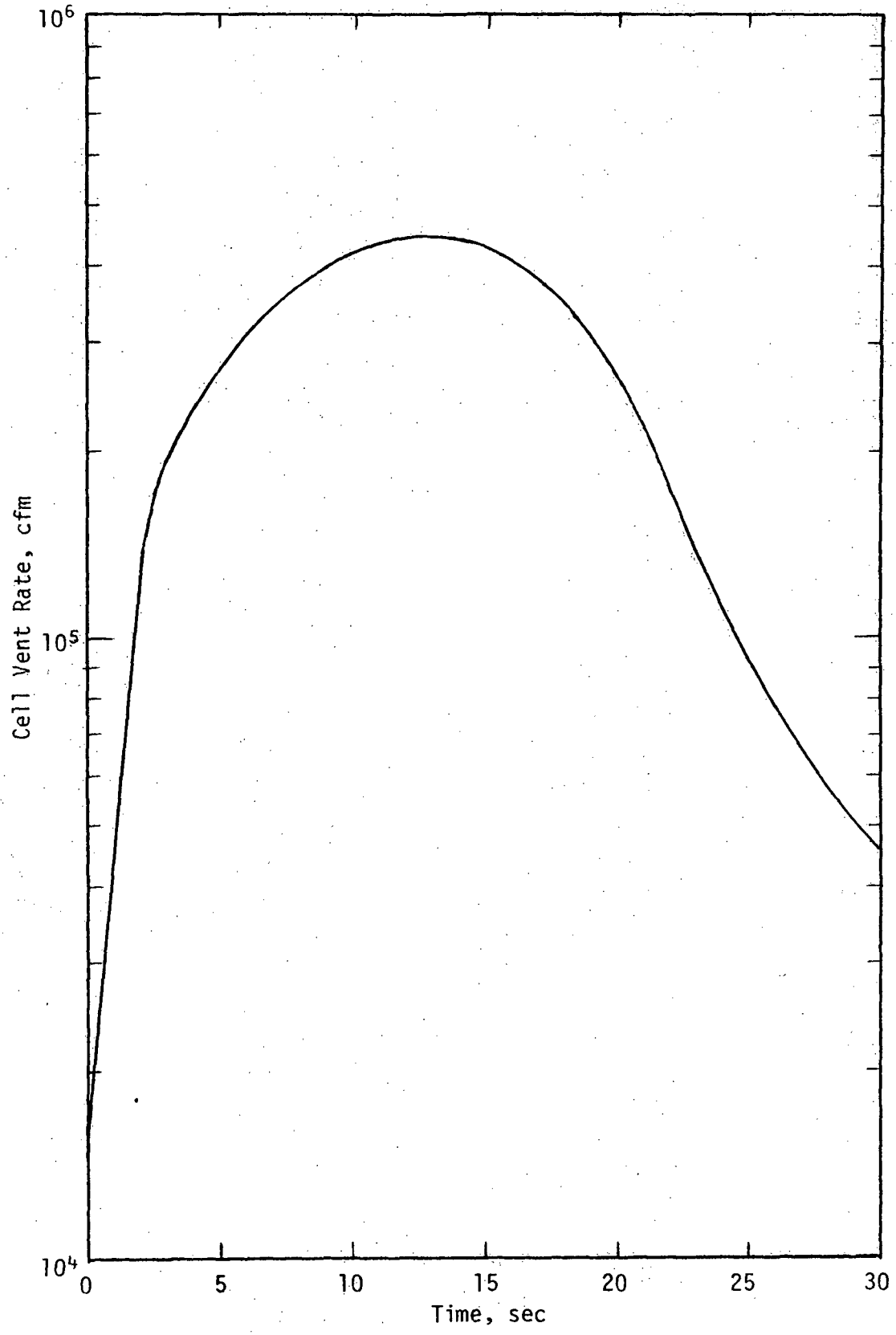


Figure Q001.693-1. IMB Cell Vent Rate vs. Time for Spray Fire Following Double-Ended Rupture in IHTS Piping

Amend. 42
Nov. 1977

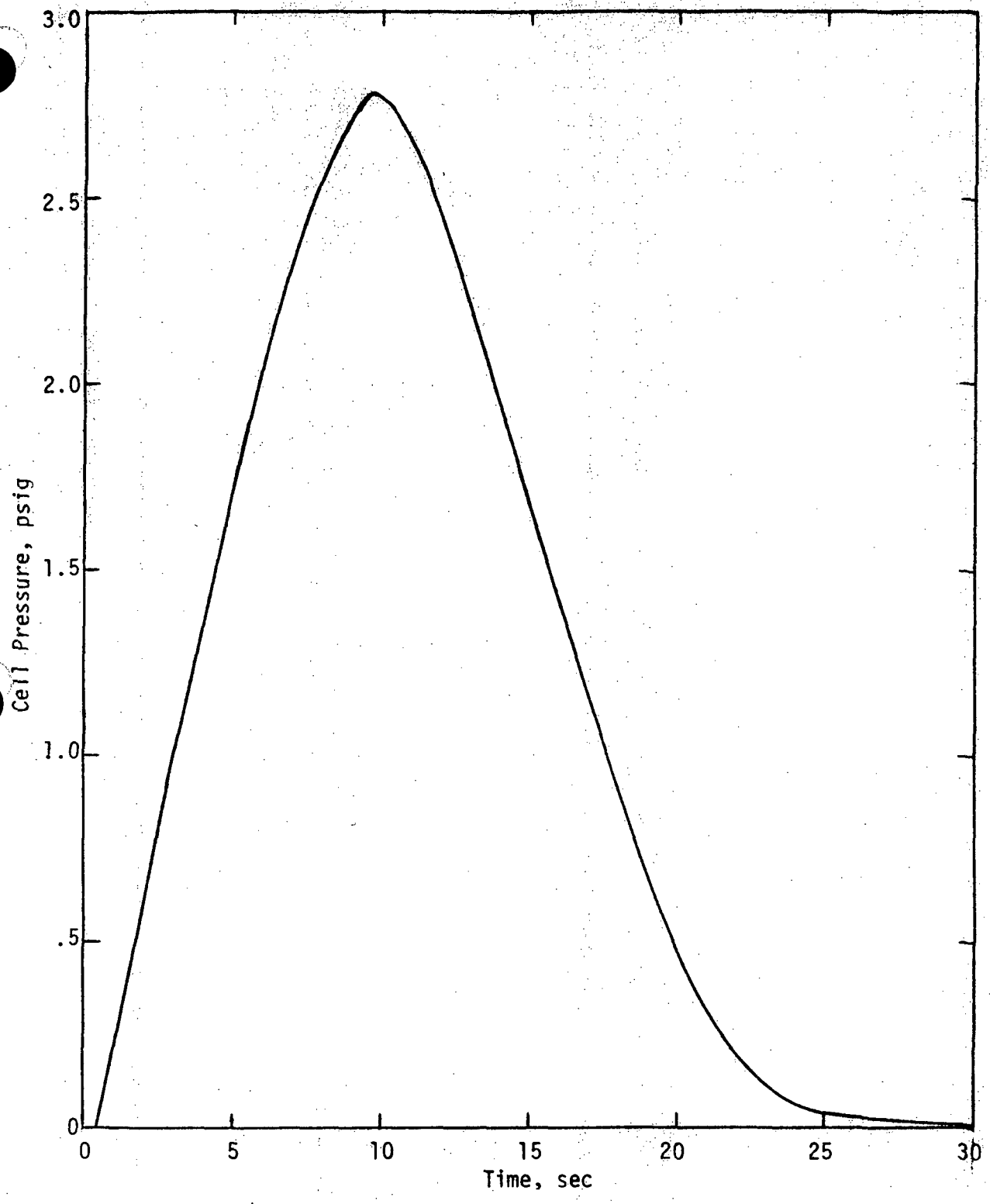


Figure Q693-2. IMB Cell Pressure vs. Time for Spray Fire Following Double-Ended Rupture in IHTS Piping

Amend. 42
Nov. 1977

Question 001.694

Your response to Item 001.241 regarding heat transfer coefficients in SOFIRE is not acceptable. Although Brown's work is cited for the development of heat transfer coefficients in the code, no indication is given of how these heat transfer coefficients apply in the region of large Grashof numbers, beyond the range tested by Brown. Our position is that, in calculations where the Grashof number is beyond the experimental range by a decade or more, unless the justification requested in Item 001.697 below can be provided, free circulation leading to the same atmosphere in the two regions should be assumed. If sufficient justification for the applicability of Brown's correlation cannot be provided, furnish the results of a calculation that treats the two cells as one.

Response:

The response to Question 001.697 provides the justification for the use of Brown's correlation at the high Grashof numbers encountered in CRBRP cells.

Amend. 42
Nov. 1977

Question 001.696

One of the slides presented by your representatives at the March 5, 1976 meeting concerning sodium fires showed a comparison of SOFIRE II predictions with data from AI LTV Test 4. The SOFIRE II results, which were calculated assuming 60% Na_2O and 40% Na_2O_2 formation, under-predicted the peak pressure by approximately 20%. On the basis of the limited empirical data presented we are concerned that the use of stoichiometric relationships for sodium-oxygen reactions which are less conservative than 100% Na_2O formation has not been sufficiently justified for one or two cell analytical models of sodium fires. Provide technical justification for your use of the above stated assumption.

Response:

For sodium pool fires, the AI LTV pool tests 4 and 5 support the fact that sodium peroxide formation will in fact ensue as a combustion product. However, because the available data (References Q001.696-1 and 2) may not adequately justify a specific stoichiometric relationship for sodium oxygen reactions, the CRBRP Project will conservatively assume 100% monoxide formation for the entire duration of its sodium pool fire events analyzed.

References

Q001.696-1 L. Baurmash, et al., "Summary Report for Laboratory Experiments on Sodium Fires", TR-707-130-007, Atomics International, August, 1973.

Q001.696-2 L. Baurmash, et al., "Test Report Two-Cell Sodium Pool Fire No. 2", TR-707-130-005, Atomics International, January 1973.

Question 001.697

In the two-cell version of SOFIRE II, the most important part of the analytical model is the mass transfer correlation for natural convection flow of air through the cell opening. The Brown correlation used in the code (W. G. Brown, International Journal of Heat and Mass Transfer, 5:869-78, 1962) was developed over a limited range of Grashof numbers in the original work. We are concerned about the extrapolation of the correlation, particularly if different flow regimes are encountered.

Provide a comparison of the range of Grashof numbers at the cell opening for the Brown experiments, AI two-cell sodium pool fire tests, and postulated two-cell pool fires for a range of configurations, including the PHTS cell opening for CRBRP. Justify the applicability of the Brown correlation for the range of accident cases. Provide the results of a study of the sensitivity of containment building pressure versus time for important parameters in SOFIRE II over a range of two-cell pool fires in CRBR.

Response:

Figure Q001.697-1 shows the sensitivity of the CRBRP containment pressure to important SOFIRE II parameters; namely, the emissivity-view factor from the sodium pool to the cell walls (AF) and to the cell atmosphere (AF3), the leading coefficient to the oxygen diffusion equation (D), and the fraction of the combustion products which is sodium monoxide. The values used for AF, AF3, and percent Na_2O bound the range of expected values; the lower value used for D was determined from matching experimental data and the larger value was selected to give conservative results. The accident scenario (described in Section 15.6.1.1 of the CRBRP PSAR) is a rupture of the sodium in-containment storage tank during maintenance. The general shape of the pressure curves as well as the values of peak pressures do not change significantly over the range of values used for the parameters. It should be noted that the values of these parameters used in CRBRP safety analyses are conservative relative to those values which have been determined experimentally or used to fit code results to the experimental data.

The Brown correlation used in the SOFIRE II code to evaluate natural convection through openings in horizontal partitions is based on theory and experiments (Ref. Q001.697-1). In addition to Brown's experiments, two-cell sodium pool fire tests were conducted at Atomics International (AI). The SOFIRE II code results agree well with the results of these AI pool tests (Ref. Q001.697-2). Table Q001.697-1 provides a comparison of the range of Grashof numbers considered in the Brown experiments and the AI pool fire tests. The table also includes the anticipated range of Grashof numbers for postulated two-cell pool fires in CRBRP. It can be seen that the Grashof numbers for typical CRBRP pool fires are several orders of magnitude higher than the Grashof numbers at which the code was verified.

However, existing experimental data suggest that extrapolation of the Brown's correlation to the higher Grashof numbers for CRBRP will not introduce significant error. The experimental results of Emery and Chu (Ref. Q001.697-3) showed only a slight increase in the slope of the Nusselt vs. Grashof-Prandtl log-log curve for (Grashof) x (Prandtl) values at about 10^7, indicative of the transition from laminar to turbulent flow. The data of Dropkin and

Somerscales (Ref. Q001.697-4) showed a linear trend in the Nusselt vs. Rayleigh number log-log plot for Rayleigh numbers as high as 10^9 with no indication of an upward trend in the slope of the curve. Likewise, the work of Jakob (Ref. Q001.697-5) for free convection heat transfer from horizontal plates showed only modest changes in the slope of the Nu vs. Gr-Pr log-log curve at Gr-Pr of about 10^8. Consequently, it is felt that the use of the Brown correlation in SOFIRE II for CRBRP cell conditions in conjunction with the conservative values assumed for code input parameters (e.g. oxygen diffusion coefficient) will provide conservative results for cell pressures and temperatures.

References:

Q001.697-1 W. G. Brown, et al., "Natural Convection Through Rectangular Openings in Partitions", Int. J. Heat Mass Trans., S, (869) (1962).

Q001.697-2 B.U.B. Sarma, et al., "Review of Current Sodium Fire Analytical Models", GEAP-14148, General Electric, September, 1976.

Q001.697-3 A. Emery and N. C. Chu, "Heat Transfer Across Vertical Layers", Transactions of the ASME, February, 1965.

Q001.697-4 D. Dropkin and F. Somerscales, "Heat Transfer by Natural Convection in Liquids Confined by Two Parallel Plates which are Inclined at Various Angles with Respect to the Horizontal", Journal of Heat Transfer, February, 1965.

Q001.697-5 M. Jakob, "Free Heat Convection Through Enclosed Plane Gas Layers," Trans. ASME, Vol. 68, 1946, P. 180.

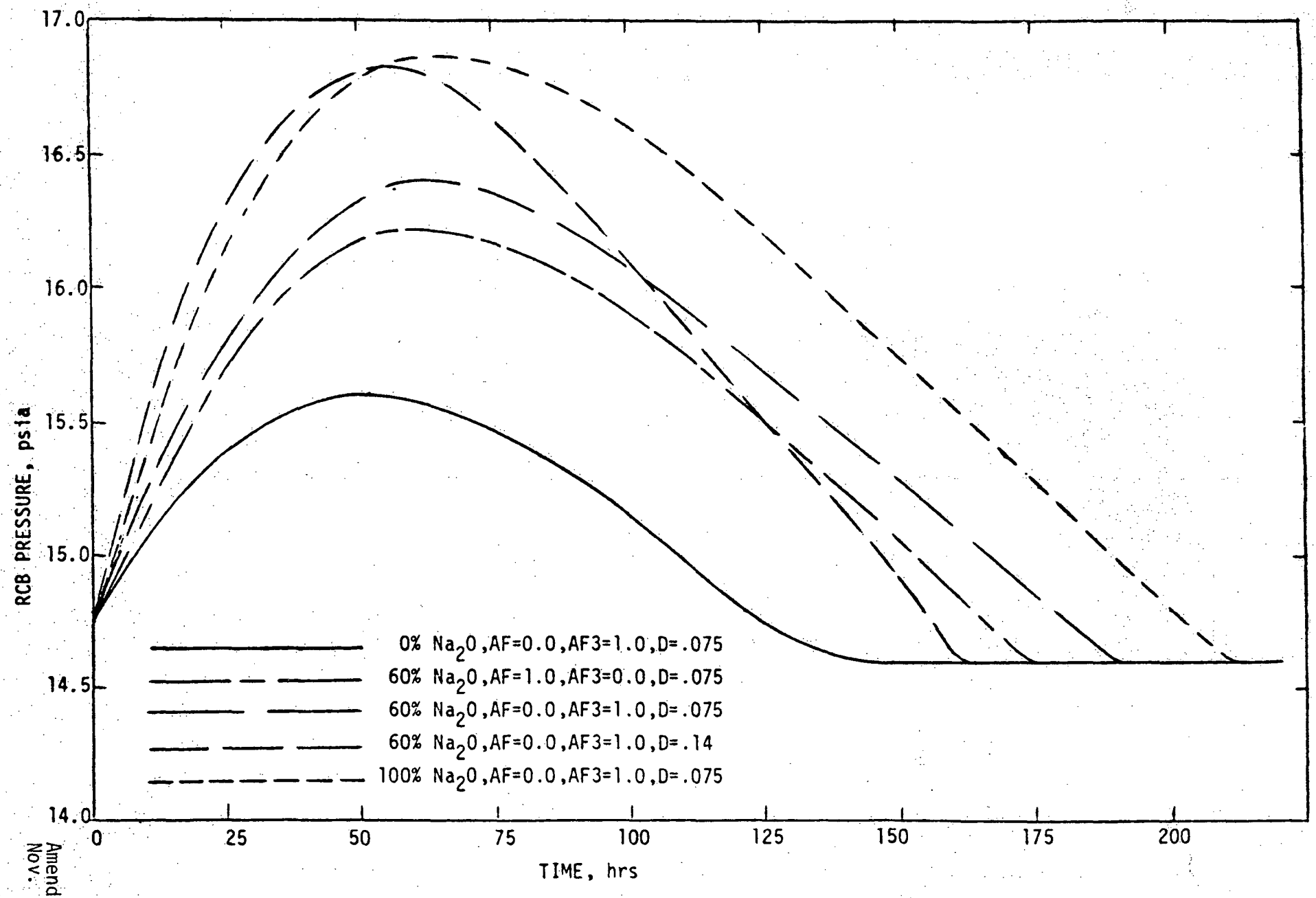
Amend. 42
Nov. 1977

Table Q001.697-1

Comparison of Ranges of
Grashof Numbers for Application
of Brown's Correlation

Application	Grashof Number Range
Brown Experiments	$3 \times 10^4 - 4 \times 10^7$
AI Pool Fire Experiments	$7 \times 10^6 - 2 \times 10^8$
Postulated CRBRP Two-Cell Pool Fires	$10^{10} - 10^{11}$

Amend. 42
Nov. 1977



CONTAINMENT PRESSURE VERSUS TIME CALCULATED BY SOFIRE II AS A FUNCTION OF KEY INPUT PARAMETERS

Figure Q001.697-1

Question 001.698

At the March 5, 1976 meeting, your representatives presented an intercomparison of experimental data on sodium burning rate for both forced and natural convection air flows from AI and Karlsruhe. We are concerned that this intercomparison was not done on a consistent and complete basis.

Provide a correlation of the experimental burning rate data for sodium pools (U.S. and Europe) prepared in a sufficiently consistent and complete manner such that pool dimensions (characteristic lengths), forced and natural convection flow and other important conditions investigated in the tests can be meaningfully compared with similar parameters in the SOFIRE II calculations for bounding accident cases. Compare and discuss any differences in flow regimes and ranges with respect to verification of the applicability of the oxygen mass-transfer correlation used in SOFIRE II for the sodium pool. (The peak burning rates and Na_2O formation should be used as a basis for comparison in tests and calculations.)

Response:

The "intercomparison" of experimental burning rate data presented at the March 5, 1976 meeting was not intended as a means of determining or verifying SOFIRE II code burning rate models. The presentation of these data was merely intended to give order-of-magnitude values of sodium pool burning rates under different experimental conditions.

Available French, (Ref. Q001.698-1), German, (Ref. Q001.698-2), and U.S., (Ref: Q001.698-3) experimental data on burning rates for sodium pool fires were reviewed again. Because the amount of data is insufficient, no correlation of burning rates with significant parameters such as Grashof number, oxygen concentration and pool characteristic length can be obtained. The data are not only limited but the quantitative data available are for different experimental conditions. For example, such key variables as flow condition (forced or natural convection) and cell and pool geometry were all varied between experiments. Consequently, no correlation of burning rate with key burning rate parameters could be obtained, and therefore a meaningful comparison with SOFIRE II model parameters could not be made.

In the SOFIRE II code, the transport of oxygen to the burning zone (pool surface) is obtained from film theory. The correlation for mass transfer of oxygen to the burning zone is based on the mass transfer analogy to heat transfer for turbulent heat transfer from horizontal square plates in an infinite medium (Ref. Q001.698-4). It should be noted that this correlation predicts higher turbulent heat (and therefore mass) transfer coefficients than the correlation for turbulent heat transfer coefficients for horizontal surfaces in enclosed air spaces which is a better representation (Ref. Q001.698-5). The correlation used has been shown to predict higher than measured burning rates for sodium pool fires in air cells (Ref. Q001.698-6). Consequently, the existing burning rate model in SOFIRE II code should provide conservative results for burning rates for CRBRP cells.

Amend. 42
Nov. 1977

References

Ref. Q001.698-1 A. Chalot, et.al., "Consequences Thermiques d'un Feu de Sodium sur un Reacteur a Neutrons Rapides - Application au Projet de Reacteur Super-Phenix," Radiation Protection - International Conference, Conf750500, May 1974.

Ref. Q001.698-2 F. Huber, et.al., "Behavior of Sodium Area Conflagrations and Suitable Protecting Systems," Institute für Reaktorentwicklung Kernforschugnzentrum Karlsruhe.

Ref. Q001-698-3 L. Baurmash, et.al., "Summary Report for Laboratory Experiments on Sodium Fires," TR-707-130-007, Atomics International, August 1973.

Ref. Q001-698-4 W. H. McAdams, Heat Transmission, 3rd Edition, 1954, page 180.

Ref. Q001.698-5 W. H. McAdams, Heat Transmission, 3rd Edition, 1954, pages 175, 182.

Ref. Q001.698-6 B. Sarma, et.al., Review of Current Sodium Fire Analytical Models, GE-FBRD, September 1976 (GEAP-14148).

Amend. 42
Nov. 1977

Question 001.699

At the March 5, 1976 meeting, the slide labeled Figure 9 showed a comparison of AI LTV two-cell Test No. 2 data with SOFIRE II results. In this comparison, the code overpredicted the peak pressure for the case of 100% Na_2O formation, whereas for the 100% Na_2O_2 formation case, it underpredicted the peak pressure. Such results would normally be reassuring. However, the marked dissimilarity between the shapes of the pressure rise versus time curves raises concerns about the fidelity of the mechanistic description of the processes modeled in the code.

Provide an explanation for the observed behavior of the peak pressure in LTV two-celled Test No. 2 compared with peak pressure computed by SOFIRE II, in particularly the higher initial rate of rise of pressure in the test and the early peaking of pressure. Indicate whether this behavior is scale-dependent for possible limitations of the LTV two-cell test with respect to scaling laws for important sodium fire parameters for such configurations.

Response:

The pressure versus time data for LTV two-cell pool Test No. 2 performed at Atomics International (AI) (Ref. Q001.699-1), which were present at the March 5, 1976 meeting, are shown in Figure Q001.699-1. In the same figure are two curves calculated with the SOFIRE II code for either 100% sodium monoxide formation or 100% sodium peroxide formation (Reference Q001.699-2). It is observed in Figure Q001.699-1 that for times greater than about 15 minutes both of the SOFIRE II calculated curves give conservative results (higher pressures) relative to experimental data. However over the first 5 minutes, the SOFIRE II curves do not follow the measured variation and the SOFIRE II pressures are less than the measured pressures.

The rapid pressure rise (start-up transient) observed experimentally is attributed to (1) sodium droplet formation caused by splashing and stripping of droplets from the discharge jet by shear forces, and (2) sensible heat transfer from the sodium to the cell atmosphere during discharge. The first process will lead to higher sodium burning rates and heat transfer rates relative to those of a stagnant pool owing to the resulting higher sodium surface area. The latter process, which is also enhanced by the larger surface area generated during the spill, is judged to be the main reason for the initial pressure spike.

The SOFIRE II code is designed to analyze the long-term effects, typically over a period of several hundred hours, of large sodium pool fires in high and low-oxygen environments. For these fires, the peak pressure should occur several hours after the initial spill. The code is not designed to model the short-term start-up effects described above. As evidenced in Figure 001.699-1, the peak cell pressure predicted by SOFIRE II for 100% monoxide formation, which is assumed

in CRBRP pool fire calculations, is much higher (and occurs much later) than that resulting from start-up transient behavior. The SOFIRE II code model, which assumes a stagnant pool throughout the accident, should provide a conservative evaluation of actual accident consequences for cases where splashing and break-up are minor. It is expected that start-up transients or effects are not strictly scale dependent; however, there may be a configuration dependency. For example, spillage of sodium from tanks at high elevations could result in significant splashing and break-up of sodium. Conversely, spillage of sodium from points close to the chamber floor should produce insignificant splashing and break-up. Consequently, the contribution of the initial transients to the overall pool fire consequence would depend on the discharge configuration. For cases where significant break-up is postulated, the discharge phase should be analyzed with the spray fire code.

References:

Q001.699-1 L. Baurmash, et al., "Test Report Two-Cell Sodium Pool Fire No. 2", TR-707-130-005, Atomics International, January 1973.

Q001.699-2 L. Baurmash, et al., "One-Cell and Two-Cell Sodium Pool Fire Tests", Conference on Fast Reactor Safety, Beverly Hills, Calif., April, 1974.

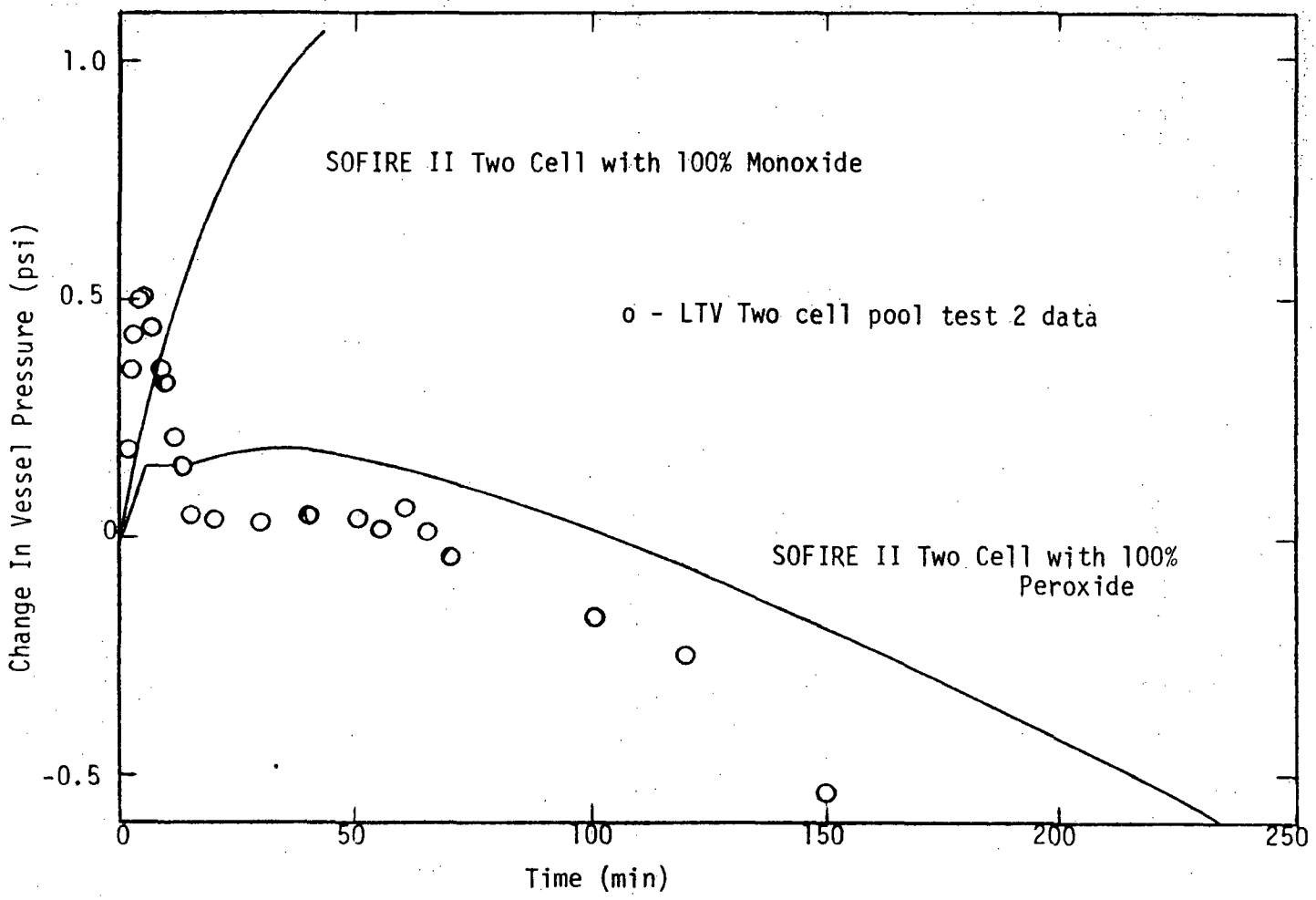


FIGURE Q001.699-1 Change in Vessel Pressure Versus Time for LTV Two-Cell Pool Fire Test No. 2

Amend. 42
Nov. 1977

Question 001.702

The two tests presented at the March 5, 1976 meeting for verification of the SPRAY code for the air case have limitations in their applicability to verification for large scale spray fires in air. Provide the results of a comparison of SPRAY code predictions with data from AI LTV sodium dispersal tests Nos. 1 and 2. This comparison should be evaluated in terms of its implications for use of the code for accident analyses and for possible need for additional verification bases.

Response:

Two sodium spray fire tests, Jet Tests Nos. 1 and 2 were performed in an air environment in the Large Test Vessel (LTV) at Atomics International (AI) (Ref. Q001.702-1). Those tests were preliminary in nature and were primarily used by AI to set and check test system components. Consequently, spray droplet sizes (a required SPRAY code input parameter) were not measured as in later tests, and the spray configuration did not correspond to the model in the SPRAY code. For example, an unknown amount of burning sodium was observed flowing down the wall in a sheet. Consequently sufficient data are not available to specify SPRAY code input parameters, and the distribution of sodium in the tests does not satisfy the criteria required in order to properly investigate the scaling aspects of spray fire analyses performed with the SPRAY code. Therefore, SPRAY code calculations of Jet Tests Nos. 1 and 2 have not been performed. However, the following information is provided to address the conservatism of the SPRAY III code for scale up to larger cell sizes.

Good agreement between SPRAY-III code predictions and experimental data for spray fires has been obtained in both low and high oxygen environments in small cells. It is expected that the application of the SPRAY-III code to larger-size CRBRP cells would yield more conservative results relative to experimental data. The main scale dependent part of the code model is the gas recirculation model. In SPRAY-III gas recirculation is described by a single toroidal vortex which involves all of the cell atmosphere. Thus, oxygen is transferred to the burning zone by convection. Spray fires in large cells may result in more complex recirculation patterns involving more than one vortex. In this case, the mass transfer of oxygen to the burning zone involves diffusion and convection. The principal exchange between vortices should be by diffusion. Since mass transfer by diffusion is slower than by convection, the SPRAY-III model should result in more oxygen-rich air reaching the burning zone and higher sodium burning rates. Consequently, the application of SPRAY III to larger sized cells should produce conservative results relative to results that may be obtained with a more complex convection model or from actual tests.

References

Q001.702-1 L. Baurmash, et al., "Summary Report for Laboratory Experiments on Sodium Fires", TR-707-130-007, Atomics International, August, 1973.

Amend. 42
Nov. 1977

Question 001.703

Provide the results of studies for key SPRAY code parameters for a spray fire following a range of partial pipe breaks in the CRBR steam generator building.

Response:

Figures Q001.703-1 through Q001.703-5 are curves of cell pressure, average gas temperature, vent rate, total sodium burned, and total sodium discharged, as a function of time for pipe break sizes of 1-inch, 2-inch, 4-inch, 9.8-inch, and 24-inch diameter. The analyses presented in the figures were carried out for time durations adequate to predict and present the peak pressure assessments. The range of partial pipe breaks presented herein are judged to represent a spetrum of events that envelope the on-going efforts to establish the design basis leak for the steam generator building. These results were obtained using SPRAY 3 code, a recently revised version of the SPRAY code. The breaks were assumed to occur in the steam generator bay (SGB) at the worst piping location, i.e. the lowest to the cell floor, from the standpoint of sodium spray release. Table Q001.703-1 gives the SPRAY 3 code input values for spray characterization parameters and cell geometry used in generating these curves. The spray conversion efficiency was assumed constant at 100% for pinhole through 4-inch diameter leaks and to vary linearly and inversely with break area from a value of 100 percent for 4-inch diameter leaks to 20 percent for a large area leak (break diameter = 24-inches). Time dependent sodium discharge rates were calculated using basic orifice equations assuming forced coolant flow up to the point of pump cavitation after which gravity feed was assumed.

Table Q001.703-1

INPUT PARAMETERS TO SPRAY 3 CODE
FOR ALL BREAK SIZES

Input Parameter	Break Diameter 1-inch	2-inch	4-inch	9.8-inch	24-inch
Droplet diameter, in.	0.18	0.18	0.18	0.18	0.18
Liner area, ft^2	36852.	36852.	36852.	36852.	36852.
Initial cell pressure, psia	14.7	14.7	14.7	14.7	14.7
Initial cell gas temp, ^oF	100.	100.	100.	100.	100.
Sodium discharge temp., ^oF	800	800	800	800	800
Cell volume, ft^3	569968	569968	569968	569968	569968
Impact diameter, ft	38.83	38.83	38.83	38.83	38.83
Spray cone angle, rad.	0.35	0.35	0.35	0.35	0.35
Initial oxygen conc., %	21	21	21	21	21
Cell height, ft	106.	106.	106.	106.	106.
Spray conversion efficiency, %	100	100	100	85	20
Cell vent area*, ft^2	2.0	8.0	26.0	50.	26.0

* Cell vent area is that needed to limit the pressure build-up to less than the cell design pressure of 3 psig.

Amend. 42
Nov. 1977

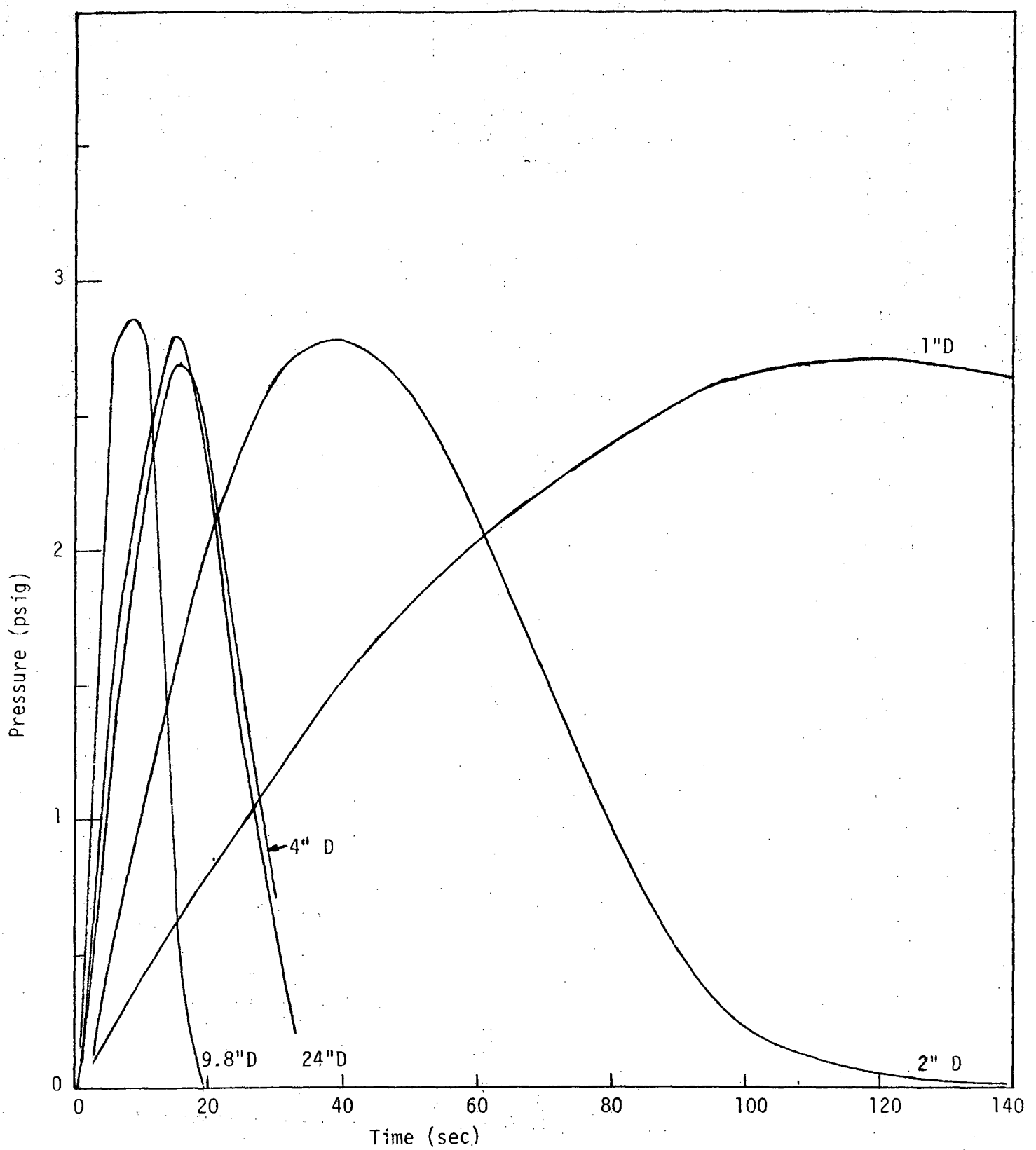


Figure 0001.703-1

SGB Pressure vs. Time Following Sodium Spray Fire For Different Break Sizes

Amend. 42
Nov. 1977

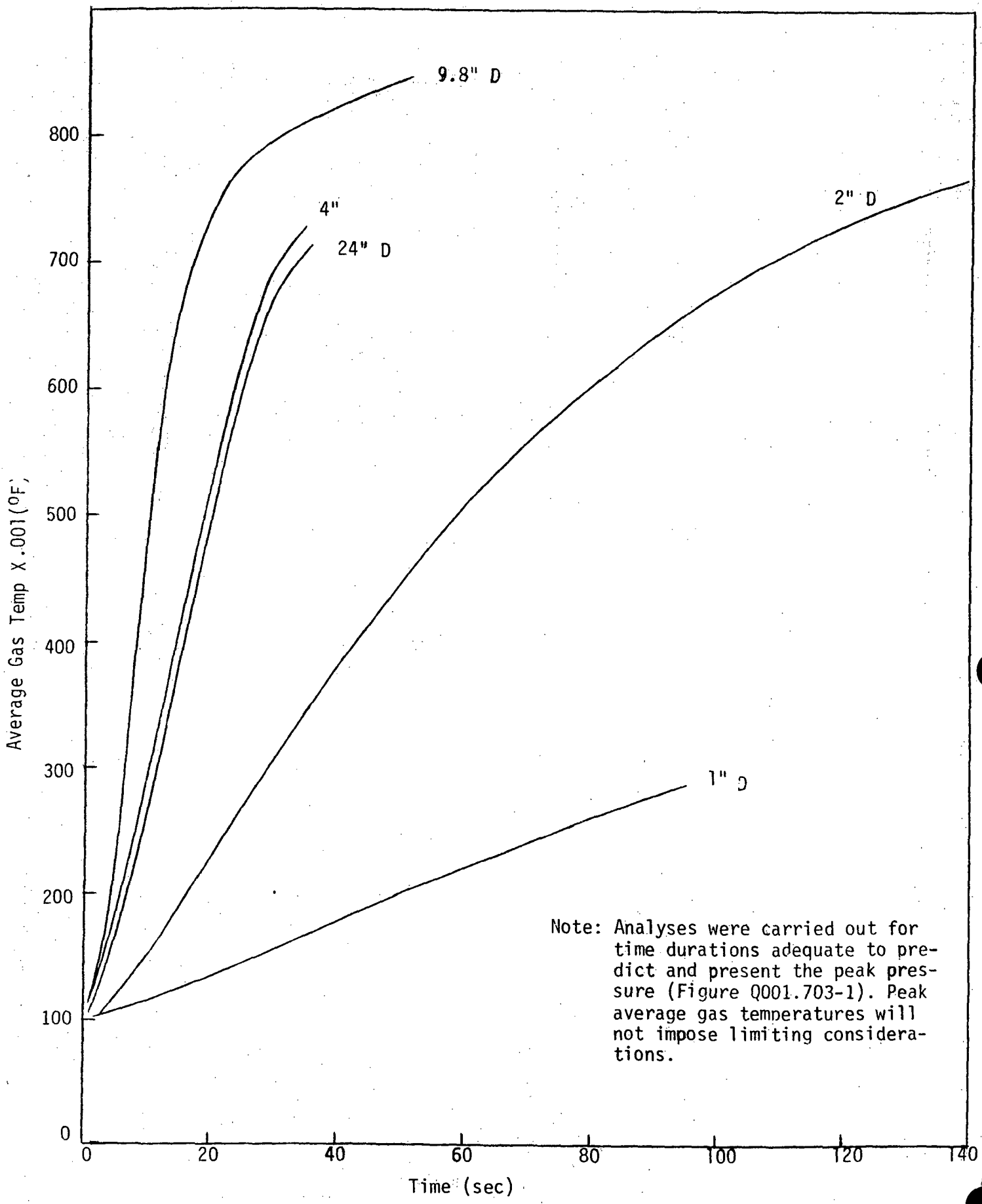
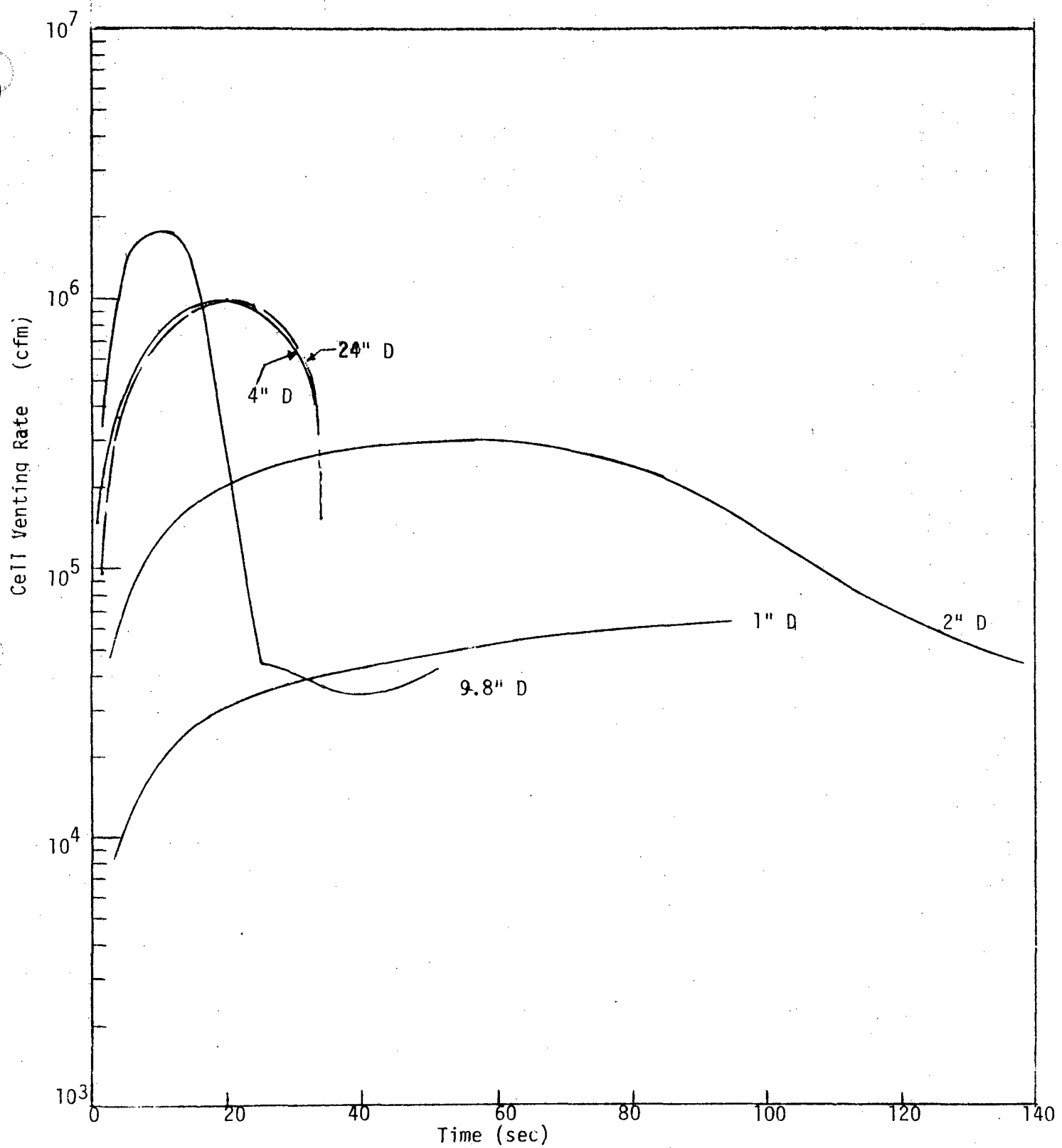


Figure Q001.703-2

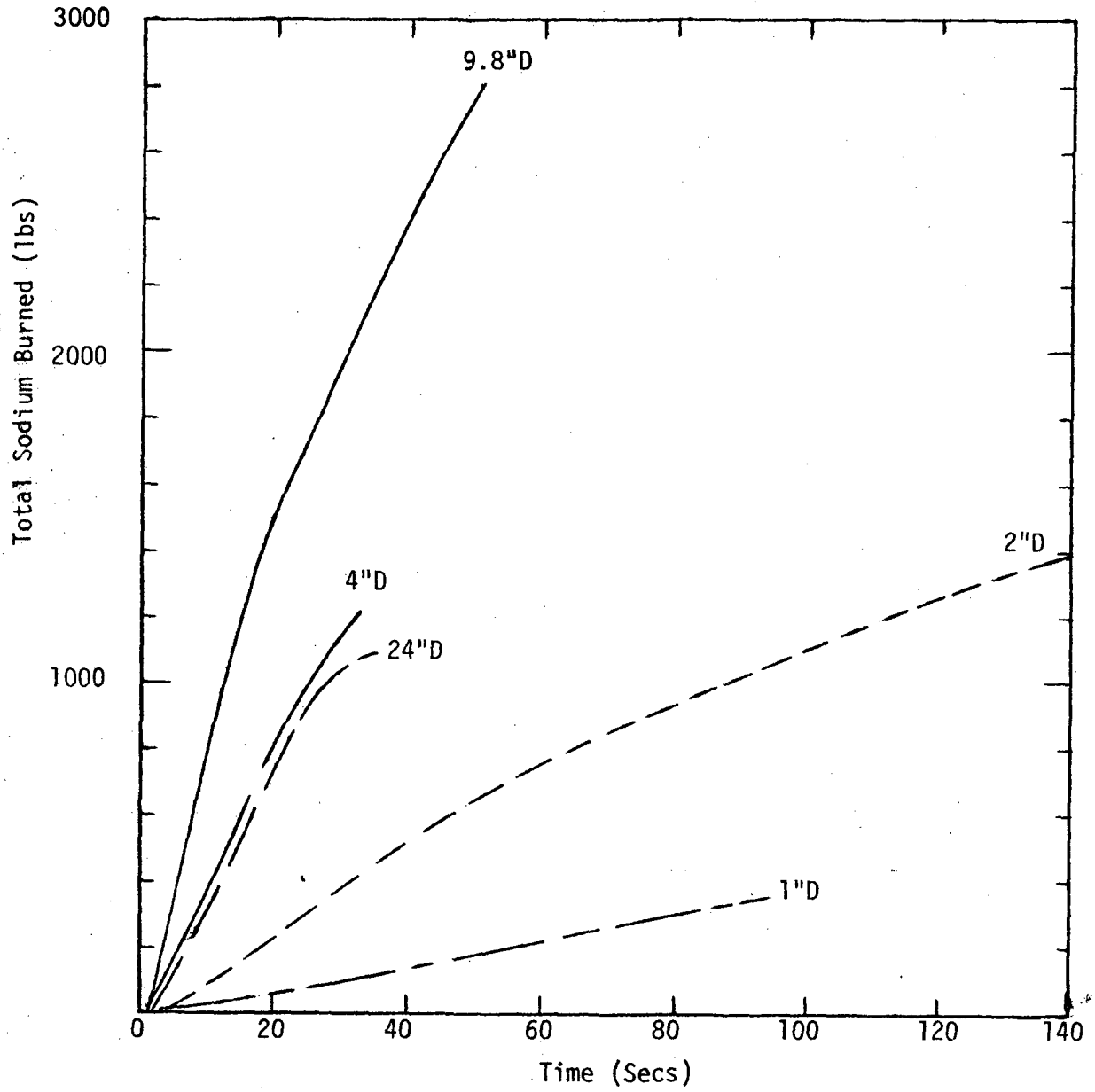
SGB Gas Temperature vs. Time Following Sodium Spray Fire For Different Break Sizes



SGB Venting Rate vs. Time Following Sodium Spray Fire For Different Break Sizes

Figure Q001.703-3

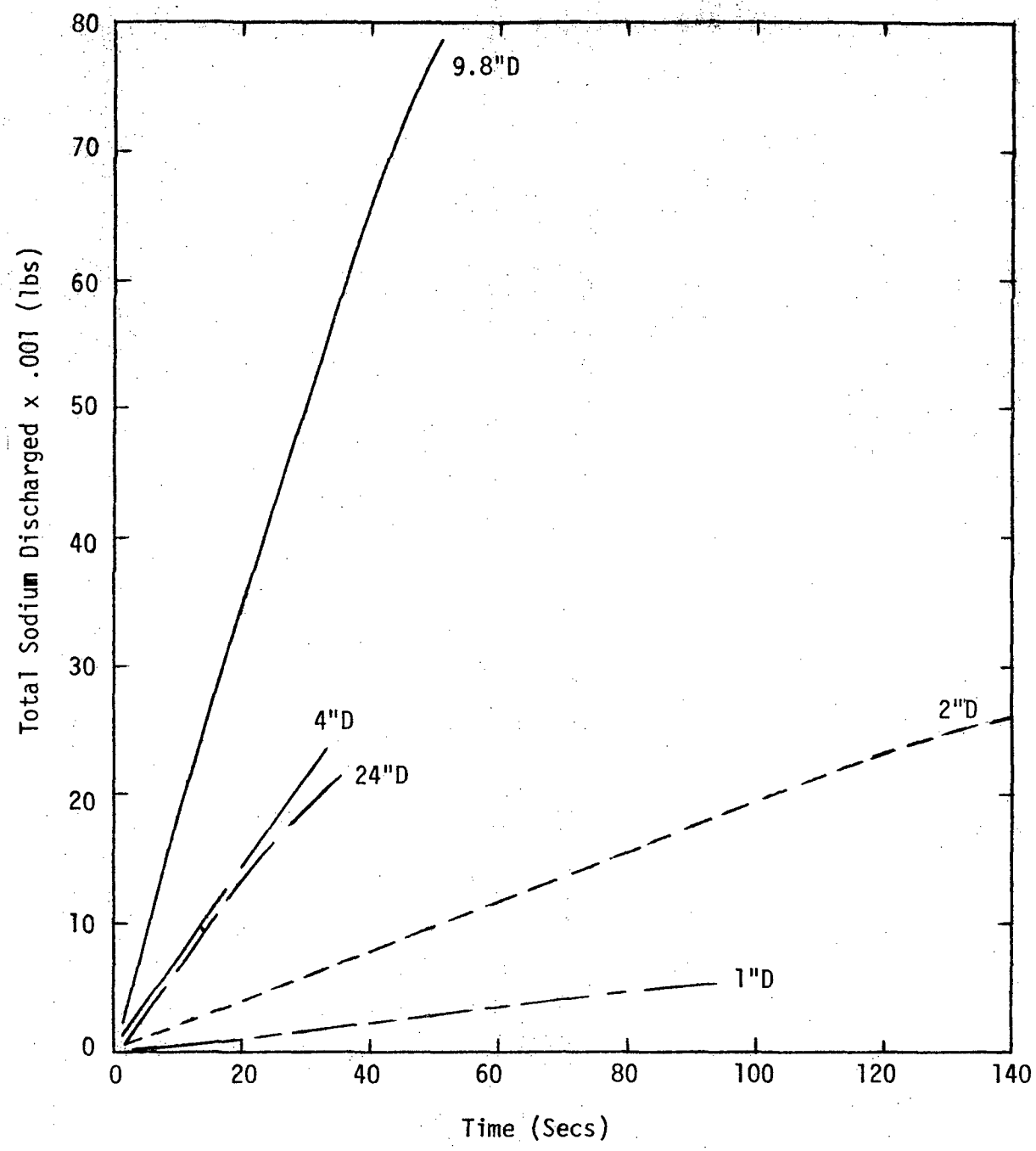
Amend. 42
Nov. 1977



Total Sodium Burned in SGB vs. Time
For Different Break Sizes

Figure 001.703-4

Amend. 42
Nov. 1977



Total Sodium Discharged to SGB vs. Time
For Different Break Sizes

Figure Q001.703-5

Amend. 42
Nov. 1977

Question 001.704

Discuss under what conditions secondary or explosive fragmentation mechanisms ("star bursts") could occur for sodium spray droplets. Discuss whether this aspect has been explored. Discuss how would such mechanisms be accounted for in the SPRAY code.

Response:

Secondary or explosive fragmentation of a liquid drop would be driven by large internal drop pressures which could result from vaporization of liquid inside the drop as a result of heating from an internal heat source or from radiation heat. For the case of sodium burning in air, this kind of explosive fragmentation is considered highly unlikely since sodium burning is a surface phenomenon, and consequently it does not constitute an internal heat source. Since sodium is not transparent, radiative transport can be neglected. Furthermore, since the burning zone temperature is highter than the temperature anywhere in the sodium drop, and since the droplet temperature cannot exceed the boiling point of sodium, it is expected that vaporization of sodium will always occur at the surface of the drop. Consequently, internal generation of vapors which would lead to explosive fragmentation of the drop is precluded.

Experimental evidence for explosive fragmentation (i.e. total breakup of the sodium droplet into small drops) of sodium spray droplets has not been observed. Single droplet burning tests conducted at AI (Ref. Q001.704-1) have indicated a shedding of small droplets when the primary sodium droplet exceeded a certain size (0.8-0.9 cm diameter). This shedding of droplets from the drop surface continued until the drop reached a limiting diameter of approximately 0.75 cm. The main drops and shedded drops continued to burn at a controlled rate.

On the basis of the above theoretical and experimental bases, the assumption of explosive breakup of sodium spray droplets is considered inappropriate.

Explosive fragmentation of sodium droplets has not been accounted for in the SPRAY code because it is not phenomenologically realistic. This is supported by the recent single drop experiments noted.

Reference:

Q001.704-1 "Test Report for the Sodium Drop Burning Tests for Somix", No. N707TR130017, Rockwell International - Atomics International Division, February 16, 1977.

Amendment 43

List of Responses to NRC Questions

Reference: NRC letter dated August 17, 1976

NRC Ques. No.
110.67
110.68

Question 110.67 (3.9.1.3)

The response to item 110.23 is insufficient. It is the staffs' position that additional detailed information on the comprehensive program of analysis, component testing, component inspection and model testing must be provided to show how these program parts will be integrated to demonstrate the structural integrity of the reactor internals. Stipulate in the PSAR your commitment to conduct an inspection program that is consistent with the intent of item C.2.3 of the Regulatory Guide 1.20 and applicable to the CRBRP. Also stipulate in the PSAR your commitment to document the results of your program consistent with the intent of item C.2.4 of Regulatory Guide 1.20 and applicable to the CRBRP. Provide a schedule of the following: (1) vibration measurement and inspection phases of the comprehensive vibration assessment program, (2) a summary of the vibration analysis program and (3) preliminary and final reports consistent with item 2.4 of Regulatory Guide 1.20.

Response:

The CRBRP approach, to assure that the reactor can satisfactorily operate without impairment from flow induced vibrations, is different than the Light Water Reactor approach. This inherent difference is largely due to the fact that the sodium coolant in CRBRP has significantly different characteristics than water. Some of these differences are 1) sodium is opaque and clings to surfaces, 2) sodium becomes radioactive during operation, 3) sodium wetted components require inert environments for handling and cleaning with subsequent requalification for insertion into the reactor, and 4) high thermal stresses occur in CRBRP components during transients.

For these reasons, the comprehensive program developed by the Project to meet the intent of the LWR Reg. Guide 1.20 differs from LWR practice because of the recognized difference between LMFBR and LWRS. This program is one of analysis, design, and testing, to assure that excessive flow induced vibrations will not occur. This program is described in amended section 3.9.1.3 of the PSAR. The extensive model test program serves the function of a prototype, which eliminates the need for extensive plant vibration testing. The model tests are heavily instrumented for vibration with over 100 accelerometers used in one test. This program is considerably more extensive than that conducted for LWR's. The scale is large (0.248), the features are modeled conservatively, key design parameters and tolerances are varied, model components are examined after testing and the flow parameters are varied over a range larger than that to which the plant will be exposed.

Due to the characteristics of sodium mentioned above, it is not possible to examine components in the detail specified in Section C.2.3. For this reason, the extensive test program, described in the previous paragraphs, is being performed to assure that no serious flow induced vibrations of Reactor Internals exist. In regions of this experimental model, where significant vibration is measured, examinations, design changes, if required, and retesting are being carried out.

The CRBRP will document the results of the above described program, as required by Item C.2.4 of Regulatory Guide 1.20.

A description of the vibration measurement and inspection phases of the comprehensive vibration assessment program will be provided a minimum of 90 days prior to plant sodium system testing. A summary of the vibration analyses program will be provided a minimum of 60 days prior to the submittal of the description of the vibration measurement and inspection program. The preliminary and final reports consistent with Item 24 of Regulatory Guide 1.20 will be provided within 60 and 180 days respectively of the completion of the vibration testing phase of the plant sodium system testing.

Question 110.68 (3.9.1.3)

It is stated in Section 3.9.1.3 that it is planned to locate two tri-axial accelerometers on the Upper Internals Structure with the intent of measuring its vibrational behavior. This is a very complicated structure and the staff questions the adequacy of only two accelerometers to measure its vibration behavior.

Outline and justify your program to measure the vibrational behavior of the Upper Internals Structure. Indicate where these accelerometers and other instrumentation will be located in the structure.

Response:

The requested information is provided in amended Section 3.9.1.3.

Question 6

Provide a description of pertinent experimental data, analytical models, and plans for obtaining future data regarding fuel rod-wire wrap interactions.

Response:

This information requested has been provided under seperate cover as Topical Report WARD-D-0149,"Wire Wrap-Cladding Interaction in LMFBR Fuel Rods." 41 43

Question 8

Provide a summary description of the various aspects of fuel temperature calculations including a complete listing of all correlations used (fission gas release, gap conductance, fuel relocation, etc.) along with the data base for each correlation and its range of applicability.

Response:

Detailed calculations have been performed in the thermal-hydraulic analyses of the CRBRP core to characterize the thermal behavior of fuel and blanket assemblies. The hot (maximum cladding temperature) rod in each fuel and blanket assembly has been analyzed throughout its equilibrium lifetime, i.e., at the beginning and end of each of three consecutive equilibrium cycles, thus providing a lifetime history of cladding temperature and fission gas pressure necessary for rod structural analyses. Additionally, fuel temperatures and linear power ratings have been calculated at the time of maximum to assure that the limiting power-to-melt is not exceeded, and thus, no fuel centerline melting occurs. Material properties and correlations have been used in the thermal-hydraulic studies to represent the physical behavior of the fuel rod. Since Question 8 was transmitted on October 6, 1975, the analyses have been discussed in detail in PSAR Section 4.4, and Topical Report WARD-D-0054, "Predicted Preliminary Thermal Hydraulic Performance of CRBRP Fuel and Blanket Assemblies," was transmitted to NRC on March 9, 1976.

Rather than collate and repeat what is already reported in the PSAR and WARD-D-0054, the following identifies, for the convenience of the reviewers, the various sections of PSAR pertaining to each of the items in the topical report table of contents.

Fuel Temperature Calculations

PSAR Section 4.4.3.3 provides the step-by-step procedure used to calculate fuel temperatures, plus a discussion of the uncertainties to be considered in the power-to-melt studies.

Analyses

In addition to Section 4.4.3.3, the following PSAR sections apply:

a) 4.4.3.2 provides an overview of the uncertainties analysis plus a detailed discussion of the combination of gap and fuel uncertainties in calculating the fuel centerline temperature and a comparison with HEDL P-19 experimental data;

b). 4.4.3.3.1 discusses in detail fuel thermal conductivity uncertainties; | 43

c). 4.4.3.3.2 provides an overview of uncertainties in gap conductance calculations, which are discusses in greater detail in 4.4.2.6.5; and | 43

d). 4.4.3.3.3 and 4.4.3.3.4 are a discussion of experimental uncertainties in melting temperature and power-to-melt, respectively. | 43

CORRELATIONS AND PROPERTIES

A complete discussion of T&H correlations and properties is presented in PSAR Section 4.4.2.6 as follows:

• Sodium Properties	4.4.2.6.1	
• Film Heat Transfer Coefficient	4.4.2.6.2	
• Cladding Thermal Conductivity	4.4.2.6.3	
• Gap Conductance (this section provides a very detailed discussion of the gap conductance model, uncertainties limitations and future analyses planned for final design and FSAR)	4.4.2.6.5	43
• Fuel Thermal Conductivity	4.4.2.6.7	
• Fuel Density	4.4.2.6.11	43
• Fuel Restructuring Parameters	4.4.2.6.15	
• Fuel Melting Temperature (this section also included a discussion of actinides redistribution and its effects on solidus temperature)	4.4.2.6.12	
• Fuel Linear Power-to-Melt (in addition to other sections previously mentioned - 4.4.3.3, 4.4.3.4)	4.4.2.6.14	
• Fission Gas Release (which includes a discussion of the fission gas release correlation, its uncertainties and fission yield values)	4.4.2.6.16	43

RESULTS

Results of the core thermal analyses for fuel and blanket assemblies are reported in Sections 4.4.3.3.5 (for equilibrium core at Plant Thermal-Hydraulic Design conditions), 4.4.3.3.6 (for first core at Plant Thermal-Hydraulic Design conditions), and 4.4.3.3.7 (for equilibrium core at Plant Expected Operating

Amend. 43
Jan. 1978

conditions). These results are also reported in great detail in Topical Report WARD-D-0054.

Conclusion

The various sections of amended PSAR Section 4.4 listed in the above provide a detailed discussion of the state-of-the-art for the core thermal-hydraulic analyses. Improvements have been and will continue to be developed, e. g., improved representation of fuel behavior, such as gap conductance, gap closure, continuous fuel restructuring models as function of burnup. All these improvements will be factored into the final analyses and reported in the FSAR. 43

Question 001.65 (5.2.1.1)

It is not clear how it will be demonstrated that the quantity of accident products reaching the lower head will be insufficient to cause melt-through. Explain the basis for this statement.

Response:

This statement was made based on the expectation that the core support structure and inlet modules would be designed to enhance post accident cooling capability to the extent that the accident products from major core damage could be cooled within the reactor vessel.

Design features including an extended inlet module design were evaluated to determine if they would be effective in enhancing the post accident cooling capability. These evaluations showed that practical design features would not substantially augment the third level thermal margins, due to uncertainties related to postulated meltfront uniformity, boil-up, and the determination of the capability of a thin stainless steel container to contain debris. Consequently, the sentence referred to in Section 5.2.1.1 is no longer considered accurate, and is being deleted. The Project completed an assessment of the thermal capability of the plant for downward debris retention which is reported in Reference 10 of PSAR Section 1.6. 43

Question 001.115 (5.3.3.6)

It is stated that the potential for longitudinal and circumferential crack growth will be investigated. What criteria will be used for judging acceptability and when will the results of this investigation be made available.

Response:

The criteria for judging acceptability is that no failures due to crack growth lead to loss of core coolable geometry. Loss of core coolable geometry is conservatively defined as sodium boiling. PSAR Section 1.6, Reference 2, provides a detailed discussion of the crack growth investigations and the consideration of the results in the CRBRP design. 43

Question 020.44

The response to item 020.2 is not complete. Revise the PSAR to indicate the tornado protection provided for the emergency cooling tower fans and motors.

Response:

Emergency cooling towers have been re-instituted into the Emergency Plant Service Water System Design. Section 3.5, Table 3.5-1 has been revised to indicate the tornado protection provided for the emergency cooling tower fans and motors. 43

Question 110.51 (3.9, 5.3, 5.4, 5.6)

Provide a statement or table which verifies that the design rules including stress limits for all ASME Code Class 1, 2 and 3 component supports are no less conservative than those of Sub-section NF of ASME Code Section III. Include the stress limits for all supports subjected to component operating conditions identified as normal, upset, emergency and faulted.

Response:

PSAR Section 3.9 presents the design criteria for all ASME Code Class 1, 2 and 3 component supports. All ASME component supports constructed to the rules of Section III, Class 1, 2 or 3 will be designed to satisfy the requirements of Subsection NF, Class 1, 2, or 3, respectively. For Class 2 and 3 component supports explicit criteria for all conditions are specified in Table 3.9-3 (revised in response to Question 110.10).

PSAR Section 5.3.3.1.8 has been amended to provide the required statement for PHTS components.

PSAR Section 9.1.4.3.2 and Table 3.2-5 have been amended to provide a description of the codes and standards used for the EVTM support. 43

Amend. 43
Jan. 1978

Question 120.74

Report WARD-D-0127 (Primary Pipe Integrity Status Report-December 1975) discusses lower bound critical crack lengths determined in model elbow tests and states that additional tests are planned to investigate the potential rupture area opening sizes at the plant operating temperature. We require that information and analysis developed from these tests be submitted for our review. Your analysis should include and justify maximum anticipated rupture areas (leak sizes) which can be developed in the primary and intermediate heat transport systems considering the system environment and design conditions.

Response:

The development test program discussed in Subsection 4.6.5 of the referenced report has been deleted from the Piping Integrity Assurance Program. The test of large crack openings is not considered necessary since Reference 2 of PSAR Section 1.6 shows conclusively that such cracks will not occur in CRBRP. Reference 2, WARD-D-0185, "CRBRP Integrity of Primary and Intermediate Heat Transport System Piping in Containment", was submitted to NRC on September 30, 1977 and supercedes WARD-D-0127. 43

Question 222.46 (C.1.3.4, Q222.32)

The integrated reliability program plan is considered to be a key element in the overall evaluation of the CRBRP reliability program. When will the integrated reliability program plan be submitted to NRC?

Response:

The integrated reliability program plan providing an overview of the Project reliability activities and their relationship to other Project activities was submitted to NRC on January 13, 1976. This was considered the first submittal of a working document.

That version of the program plan has been superceded by the program plan provided in PSAR Appendix C (Amendment 36) dated March 1977. Any further modifications of the plan will be reflected as amendments to Appendix C. 43

Question 222.95 (3.8.4.1.8)

In Section 3.8.4.1.8 you state that the electrical manholes of your onsite distribution system are Seismic Category I. Specify what systems this provision covers, and describe your flood protection features for these electrical manholes.

Response:

The Seismic Category I Electrical Manholes, as described in Section 3.8.4.1.7, are provided for supplying standby onsite power to the Emergency Cooling Towers in the yard. Flood protection features for these electrical manholes will be discussed in a future amendment.

33 43

Amend. 43
Jan. 1978

TABLE Q310.51-1

SUMMARY OF ENVIRONMENTAL REPORT EVENTS RELEASING SODIUM REACTION PRODUCTS TO THE ATMOSPHERE

PARAMETERS	Accident 4.1	Accident 4.2	Accident 5.2	Accident 8.1	Accident 8.2	Accident 8.3	Accident 8.4	Q000.28	PSAR 15.6	
Sodium Spill (LBS) Pool Area (FT^2) Location	130 2500 SGB-Open Cell During Maintenance	250 280 RSB-Open Cell During Maintenance	337* N/A Steam Generator Sodium Water Reaction Pressure Relief System	240,000 830 RCB-Open Cell During Maintenance	193,000 1450 RCB-Closed PHTS Cell	600,000 2500 SGB-Closed Cell	57,000 280 RSB-Closed EVST Cell	5,000 285 RCB-Closed Cold Trap Cell	4,000,000 6818 SGB	43 42 43
Max. Burning Rate ($LB/HR\text{-}FT^2$)	2.30	2.30	N/A	4.90	0.46	0.20	0.30	0.33	4.8	43
Release Pathway (leakage rate)	Direct Release to Atmosphere	Direct Release to Atmosphere	SWRPRS Exhaust direct to Atmosphere	Leakage from RCB to Atmosphere (Calculated** based on 0.1%/day @10 psig)	Leakage from Cell (Calculation ** based on 100%/day @ 10 psig) to RCB; then leakage from RCB to atmosphere (Calculation ** based on 0.1%/day @ 10 psig)	Leakage from Cell to Atmosphere (Calculated** based on 100%/day @ 10 psig)	Leakage from Cell to Atmosphere (Calculated** based on 100%/day @ 10 psig)	Leakage from Cell (Calculation** based on 100%/day @ 10 psig to RCB; then leakage from RCB to atmosphere (Calculation ** based on 0.1%/day @ 10 psig)	Direct Release to Atmosphere	43
Duration of Aerosol Release to Atmosphere	2 minutes	30 minutes	15 seconds	145 hours	720 hours	192 hours	96 hours	720 hours	8 min.	43 42
Total Aerosol Released to Atmosphere (kg of Na)	16	32	9.2	1.2	0.10	2.23	0.5	0.003	307	43

*Accident 5.2 considers rupture of a steam generator tube. The 337 pounds of sodium is the quantity of intermediate sodium that reacts with steam generator water.

**Assume square root pressure-leakage relationship

TABLE Q310.51-2

DOWNWIND NaOH CONCENTRATIONS

Accident	Milligrams NaOH/Cubic Meter		
	Control Room Air Intake	Exclusion Boundary	LPZ
ER 4.1	9.4×10^{2}	5.2×10^{-1}	0.14
ER 4.2	1.2×10^{2}	6.9×10^{-2}	2.0×10^{-2}
ER 5.2	4.0×10^{3}	2.4	0.63
ER 8.1	1.5×10^{-4}	0.9×10^{-8}	2.4×10^{-6}
ER 8.2	2.6×10^{-4}	1.5×10^{-7}	3.9×10^{-9}
ER 8.3	2.2×10^{-2}	1.3×10^{-5}	3.3×10^{-6}
ER 8.4	9.4×10^{-3}	5.7×10^{-6}	1.5×10^{-6}
ER Q000.28	8.0×10^{-6}	4.5×10^{-9}	1.2×10^{-9}
IHTS Pipe Leak (PSAR Sect. 15.6.1.5)	3.5×10^{3}	2.4	0.63

42

Amend. 42
Nov. 1977

Amendment 45

List of Responses to NRC Questions

Reference: NRC Letter Dated August 17, 1976

NRC Ques. No.
110.76
310.49

Question 110.76 (3.9, 4.1.7, 5.2, 5.3.3.1)

The responses to items 110.27 and 110.40 are not completely acceptable. In addition to the information presented in Section 3.9, 4.1.7, 5.2 and 5.3 of the PSAR it is the staff's position that the design control measures, as required by 10 CFR Part 50, Appendix B.III, that will be employed to demonstrate the applicability and validity of these computer programs shall conform to the criteria outlined in NRC Standard Review Plan, Section 3.9.1, paragraph II.2. Identify those programs which, in your opinion, satisfy paragraph II.2.a of NRC Standard Review, Section 3.9.1. For the remainder of the programs, provide a summary comparison of the results as outlined in paragraph II.2 of NRC Standard Review Plan, Section 3.9.1. The mere fact that a program is in the public domain does not necessarily mean that the program has been properly verified and does not satisfy paragraph II.2.a of Standard Review Plan Section 3.9.1.

The above requirement is applicable to all computer programs which will be used by vendors or the reactor manufacturer in addition to those described in the CRBRP PSAR.

Response:

As indicated in response to question 110.40, the Project recognizes its obligation to verify the computer codes used to demonstrate the adequacy of CRBRP systems and components to protect the public health and safety. Further, the criteria in SRP 3.9.1, Section II.2 are recognized as appropriate for implementing the requirements of 10CFR50, Appendix B in this regard. Any specific codes whose verification is by means other than as specified in the SRP will be so identified. Currently, it is not anticipated that such alternate verification will be necessary for codes used by the Project.

Appendix A of the PSAR has been revised to identify those codes which satisfy the requirements of SRP 3.9.1, Section II.2.a. The Project does not consider availability to the public as sufficient to meet this verification criterion. Appendix A of the PSAR has been further revised to identify summary comparisons of certain code results with other data as specified in SRP 3.9.1, Section II.2. As indicated in the Appendix, the required comparisons for many codes are contained in the references cited for the codes. Where no comparisons (or reference to comparisons) are provided, efforts are planned to document the code verification. The documentation will be available to NRC as it is completed.

Question 310.49(15.7.7)

Provide an analysis of the radiological consequences of a cold trap fire based on the following assumptions:

1. failure of the crystallizer,
2. 1% failed fuel,
3. 100 ppb of plutonium,
4. complete volatilization of volatile species, unless conservative partition coefficients can be justified (see 312.51),
5. for the primary sodium cold trap, assure the design basis leak rate of the containment (after isolation), and direct exhaust via the RCB ventilation system if the containment is continually purged.
6. For the EVS cold trap, assume leakage of the cell at tech spec. rate.
7. State all other assumptions.

Response

The response to the question is presented in two parts, the first addresses the Primary Sodium Cold Trap and the second the Ex-Vessel Storage (EVS) Sodium Cold Trap.

Primary Sodium Cold Trap

The cold trap proper consists of an economizer and crystallizer tank. Primary sodium impurities, including radionuclides released to the coolant from failed fuel pins, are condensed and deposited on collector sites in the crystallizer. The sodium volume in the crystallizer is approximately 100 ft^3 (5000 lbs.). The crystallizer sodium temperature is purposely low (<300°F) to insure condensation of coolant impurities. For a postulated crystallizer rupture, the low sodium temperature coupled with the inerted (2% O_2) cell atmosphere would result in only a minor pressure and temperature transient. The cold trap cell is located below, and is closed to, the upper Reactor Containment volume. All interior surfaces of the cell are steel lined.

An extremely conservative assessment of the consequences of a postulated primary cold trap fire has been conducted by assuming a sodium spray release and subsequent pool fire. It should be noted that failure of the crystallizer resulting in loss of the sodium inventory is not expected to release a significant portion of the cold trap radioactive inventory,

Amend. 45
July 1978

since this inventory consists of condensed solids within the trap. The decay heat generated in the cold trap is insufficient to significantly raise the temperature of its contents. Thus, loss of sodium should stabilize the condensed solids in the crystalizer at a lower temperature. Nevertheless, for extreme conservatism the entire trap radioactive inventory is assumed evenly dispersed in the sodium discharged to the cold trap cell.

The postulated sodium leak is a 15 gpm leak for 20 minutes. One-hundred percent (100%) of this leak is assumed to react as a spray in the cold trap cell. Additional sodium is released as a pool on the floor of the cell. A total of 6950 pounds of sodium is contained in the pool.

For additional conservatism, all the sodium released to the cell is assumed to be at an initial temperature of 830°F, the operating temperature of the inlet sodium, even though this is considerably higher than the crystallizer sodium temperature.

The radiological consequences of this event were analyzed as follows:

1. The sodium reacts with available oxygen in the cell. The burning releases Na_2O as aerosol.

2. The radionuclide concentrations in the sodium are established assuming the entire cold trap radioactive inventory is evenly distributed in the sodium discharged to the cell. The sodium nuclide concentration also includes the design basis (1% failed fuel) nuclide concentration in the primary sodium.

3. The cold trap radioactive inventory is based on continuous plant operation for 15 years, with 1% failed fuel. The primary sodium inventory is also based on the design basis condition at 1% failed fuel, including 100 ppb of plutonium. The cold trap inventory is itemized in Table 12.1-10, and the primary sodium nuclide concentrations are provided in Table 12.1-6.

4. The radionuclide concentrations in the aerosol, generated during sodium combustion, are the same as the initial concentrations in the sodium.

5. Radioactive decay during the accident is neglected.

6. All the aerosol generated in the cold trap cell is immediately released to the RCB. No credit for plate-out, settling, or retention of the aerosol in the cold trap cell is taken.

7. Leakage of aerosol from the RCB to the confinement annulus was computed based on the containment design basis leak rate (0.1% Vol/Day).

8. The Containment Isolation System automatically isolates the RCB and prevents direct release of aerosol through the exhaust system.

9. Fallout (cloud depletion) of radioactive material during downwind transport is conservatively neglected.

10. Release from the confinement annulus would be through the annulus filtration system. The attenuating effect of the annulus filters was conservatively neglected and consequently the predicted off-site doses are conservative by more than an order of magnitude.

Using the dose calculational methodology described in Chapter 7 of the ER, and the 95% X/Q's (Table 2.6-38A), potential off-site doses have been determined. Doses at the Exclusion Boundary and the Low Population Zone are itemized in Table Q310.49-1. As the table indicates, large margins exist between the potential doses and the applicable guideline values.

Table Q310.49-1

OFF-SITE DOSES FROM PRIMARY
SODIUM COLD TRAP FAILURE

	Dose (Rem)		
Organ	10CFR100	Exclusion Boundary (2 hr. - 0.42 mi.)	Low Population Zone (30 days - 2.5 mi.)
Bone	150*	7.4×10^{-3}	1.0×10^{-2}
Lung	75*	6.3×10^{-4}	8.7×10^{-4}
Thyroid	300	4.1×10^{-4}	5.7×10^{-4}
Whole Body**	25	1.3×10^{-3}	1.8×10^{-3}

*Equivalent to 10CFR100 guidelines; see "Site Suitability Report", Office of Nuclear Regulation, USNRC, Docket No. 50-537, March 4, 1977.

**Includes inhalation and external semi-infinite cloud beta and gamma exposures.

EVS Sodium Cold Trap

A conservative assessment of the consequence of a postulated EVS cold trap fire has been conducted by assuming a sodium spray and subsequent pool fire. In the scenario which is analyzed herein, it was assumed that a sodium leak occurs in the cold trap cell, causing the (non-safety-related) cold trap cooling system to fail. The plant operators would be alerted by several sodium leakage, temperature, smoke, radiation and pressure detectors, but it is assumed that they fail to isolate the cold trap circuit with the isolation valves which are located in a separate environment and thus not affected by the fire. The leaked sodium burns all the oxygen available in the cell. It should be noted that a leak in this cell is not expected to release the radioactive contents of the cold trap inventory since this inventory consists of condensed solids in the trap. Nevertheless, it is conservatively assumed that the radionuclide concentration in the leaked sodium is that which would exist if all of the cold trap inventory was redissolved in the EVS sodium. In fact, most of the redissolution would take place long after the available oxygen was consumed.

For a sodium leak, the resultant cell temperature, and thus the cold trap temperature, would not rise to a value above that of the leaked sodium. Analysis for the Primary Cold Trap shows that it takes tens of hours to redissolve the cold trap contents, even if the cold trap temperature rises to 600°F. Similar analysis for the EVS cold trap has not been done, but similar behavior would occur. In actuality, the EVS cold trap release would be expected to be slower since its normal inlet sodium temperature is less than 500°F. The temperature rises to 600°F infrequently, only during operations to remove sodium surface impurity buildup in the EVST.

The decay heat generated in the cold trap is not sufficient to cause a significant temperature rise in the cold trap. If the trap were cut off from both its cooling system and the sodium process stream, the trap temperature would not rise above its normal operating temperature.

The radiological consequences of this event were analyzed as follows:

1. The total inventory of the cold trap is redissolved into the EVS sodium. The radioactivity within the 60,000 gal. of EVS sodium includes the normal inventory plus that from the cold trap. It was obtained from the cold trap inventory as given in Table 12.1-24 and the distributions ratio between the cold trap and the EVS sodium as given in Table 12.1-7. The concentration was assumed uniform throughout the volume.

2. Following failure of the EVS cold trap, 8000 gal. of sodium spills into the cold trap cell. This is the maximum amount that can spill due to the arrangement of the lines from the EVST. With the conservative assumptions followed, the radiological consequences are insensitive to the total volume of sodium spilled.

3. All of the oxygen within the cold trap cell and the adjoining cells which share a common atmosphere reacts with the sodium. The total volume of these cells is 33000 cubic feet. These cells contain a maximum of 2 w/o oxygen in nitrogen.

4. All of the sodium oxide produced is Na_2O and all of it is airborne as an aerosol.

5. All of the aerosol generated is immediately released to the atmosphere without credit for plate-out or settling within the cold trap cell or the Reactor Service Building (RSB), depletion along the leak paths from the cold trap cell to the RSB or filtration through the RSB exhaust system.

6. The concentration of radioactivity in the aerosol is the same as that in the spilled sodium. The discussion forwarded in response to Question 310.50 reviewed the partitioning of fission products in the aerosol resulting from burning of the spilled sodium. It was shown that the assumption of equal activity in the pool and aerosol sodium was conservative.

7. Radioactivity decay during the accident is neglected.

8. Fallout (cloud depletion) of radioactive material during downwind transport is conservatively neglected.

Using the dose calculational methodology described in Chapter 7 of the ER and 95% X/Q's, potential offsite doses have been determined. Doses at the Exclusion Boundary and the Low Population Zone are itemized in Table Q310.49-2.

Table Q310.49-2

OFF-SITE DOSES FROM EVS
SODIUM COLD TRAP FAILURE

45

Organ	10CFR100	Dose (Rem) Exclusion Boundary (2 hr. - 0.42 mi.)	Dose (Rem) Low Population Zone (30 days - 2.5 mi.)
Bone	150*	3.8	1.1
Lung	75*	.21	.056
Thyroid	300	.12	.033
Whole Body	25	.26	.070

*Equivalent to 10CFR100 guidelines; see "Site Suitability Report", Office of Nuclear Regulation, USNRC, Docket No. 50-537, March 4, 1977.

Department of Energy
Clinch River Breeder Reactor
Plant Project Office
P.O. Box U
Oak Ridge, Tennessee 37830
Docket No. 50-537

807

File: 05.10

November 30, 1978

Mr. Roger S. Boyd, Director Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Boyd:

AMENDMENT NO. 47 TO THE PRELIMINARY SAFETY ANALYSIS REPORT FOR CLINCH RIVER BREEDER REACTOR PLANT

The application for a Construction Permit and Class 104(b) Operating License for the Clinch River Breeder Reactor Plant, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment No. 47 to the Preliminary Safety Analysis Report pursuant to 50.34(a) of 10 CFR Part 50. This Amendment No. 47 includes: an update to Section 13.7, "Industrial Security" and the security portions of Section 9.11, "Communication System"; an update to Section 9.10, "Compressed Gas System"; an update to Section 3.8A, "Buckling Stress Criteria" (steel containment vessel); an update to Appendix B, "General Plant Transient Data"; and other updates and revisions, as well as a response to NRC's request for additional information contained in a letter dated December 1, 1976.

A Certificate of Service, confirming service of Amendment No. 47 to the PSAR upon designated local public officials and representatives of the EPA, will be filed with your office after service has been made.

Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

Raymond L. Copelan

Raymond L. Copelan
Acting Assistant Director for
Public Safety Enclosure

PS:78-301
Enclosure

cc: Service List
Standard Distribution
Licensing Distribution

SUBSCRIBED and SWORN to before me this 17th day of November, 1978.

Inez H. Minor

Notary Public
My Commission Expires May 21, 1980

SERVICE LIST

Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Atomic Safety & Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

S. Wallace Brewer, Judge
Office of County Judge
Roane County Court House
Kingston, TN 37763

Dr. Thomas Cochran
Natural Resources Defense Council, Inc.
917 15th Street, NW
8th Floor
Washington, DC 20005

Docketing & Service Station
Office of the Secretary
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Counsel for NRC Staff
U. S. Nuclear Regulatory Commission
Washington, DC 20555

William B. Hubbard, Esq.
Assistant Attorney General
State of Tennessee
Office of the Attorney General
422 Supreme Court Building
Nashville, TN 37219

Mr. Gustave A. Linenberger
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Marshall E. Miller, Esq.
Chairman
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Luther M. Reed, Esq.
Attorney for the City of Oak Ridge
253 Main Street, East
Oak Ridge, TN 37830

Anthony Z. Roisman, Esq.
Natural Resources Defense Council
917 15th Street, NW
Washington, DC 20036

Dr. Cadet H. Hand, Jr., Director
Bodega Marine Laboratory
University of California
P. O. Box 247
Bodega Bay, CA 94923

Lewis E. Wallace, Esq.
Division of Law
Tennessee Valley Authority
Knoxville, TN 37902

STANDARD DISTRIBUTION

Mr. W. J. Roberts, Prog. Mgr. (2)
Atomics International Division
Rockwell International
P. O. Box 309
Canoga Park, CA 91304

Mr. M. M. Hoyle (2)
Project Manager, CRBRP
Burns and Roe, Inc.
700 Kinderkamack Road
Oradell, NJ 07649

Mr. Lochlin W. Caffey (2)
Director
Clinch River Breeder Reactor Plant
P. O. Box U
Oak Ridge, TN 37830

Mr. W. I. Clifford (2)
Project Manager, CRBRP
Stone & Webster Engineering Corp.
P. O. Box 811
Oak Ridge, TN 37830

Mr. Don E. Erb (1)
Acting Assistant Director for
Reactor Projects
Division of Reactor Research and
Technology
U. S. Department of Energy
Washington, DC 20545

Mr. Harold H. Hoffman (1)
Site Representative
U. S. Department of Energy
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, PA 15663

Mr. J. E. Nolan (2)
Project Manager, CRBRP
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box W
Oak Ridge, TN 37830

Mr. W. W. Dewald, Project Manager (2)
CRBRP Reactor Plant
Westinghouse Electric Corporation
Advanced Reactors Division
P. O. Box 158
Madison, PA 15663

Mr. T. R. Stevens (1)
Resident Manager, CRBRP
Burns and Roe, Inc.
P. O. Box T
Oak Ridge, TN 37830

Mr. George G. Glenn, Manager (2)
Clinch River Project
General Electric Company
P. O. Box 5020
Sunnyvale, CA 94086

Mr. Hardy B. Adams, Jr. (2)
Projects Manager, LMFBR Programs
Tennessee Valley Authority
1300 Commerce Union Bank Building
Chattanooga, TN 37401

LICENSING DISTRIBUTION

Mr. N. B. Hughes
Manager of Power
Tennessee Valley Authority
830 Power Building
Chattanooga, TN 37401

Mr. R. M. Little
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

Dr. Jeffrey H. Broido, Manager
Analysis and Safety Department
Gas Cooled Fast Reactor Program
P. O. Box 81608
San Diego, CA 92138

Amendment 47

Clinch River Breeder Reactor Plant

Preliminary Safety Analysis Report

(Docket No. 50-537)

This forty-seventh amendment to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report includes an update to sections describing Industrial Security, the Communication system, the Compressed Gas System, Buckling Stress Criteria and General Plant Transient Data, as well as other updates and revisions. Vertical margin lines on the left hand side of the page are used to identify new design information, while lines on the right hand side identify question/response information.

A page replacement guide appears following the list of responses to NRC questions.

Reference: NRC Letter Dated December 1, 1976

NRC Ques. No.
Q001.589

PAGE REPLACEMENT GUIDE FOR

AMENDMENT 47

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 47 to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket No. 50-537. Amendment 47 consists of new and replacement pages for the PSAR text and question/response supplement pages.

The following attached sheets list Amendment 47 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

Amendment 47

List of Responses to NRC Questions

Reference: NRC Letter Dated December 1, 1976

NRC Ques. No.
001.589

Question 001.589 (F2.2)

Update the PSAR to resolve the inconsistencies regarding the attachment of the reactor vessel to the support ring. Question 001.379 (Amendment 17) and Section 5.2.2.1 (Amendment 17) assert that the vessel flange is welded to the support ring, while Table F.2-2 (Amendment 19) and Question 001.544 (Amendment 23) continue to discuss a bolted attachment.

Response:

Details regarding the reactor vessel attachment to the support ring are provided in PSAR Section 5.2 and Figure 5.2-1. Appendix F (Table F2-2) reflects the design in place at the time the appendix was submitted (Amendment 5). The analysis results reported in Appendix F are for that design configuration.

The information contained in the response to Question 001.379 and PSAR Section 5.2.2.1 (Amendment 17) noted a change in the bolted support design to the current welded configuration and indicated that previous analytical models are appropriate for the new design configuration. Concurrent with this response, the response to Question 001.544 is being changed to reflect the present design configuration.

Question 001.544 (F6.3.2.3)

The closure head moves independently of the reactor vessel in the REXCO computer model. Section F6.3.2.3 states that the "mass of the head has been increase to better reflect the restraint afforded by the suspended vessel and holdown bolts". Provide the mass of the head and head-mounted components. Discuss what amount the mass was increased and indicate how the amount of the increase in mass was determined.

Response:

In the current reactor system design (See PSAR Section 5.2) and the REXCO model, the reactor vessel and its contents are suspended from a ledge. The closure head is mounted on the vessel flange. In calculating the sodium slug upward force, it was thought necessary to increase the mass of the head, so that the head upward acceleration and its effects on the sodium slug force calculation would be properly modeled. 47

In a static sense, the total mass of the reactor vessel system would be added. However, during a core disruptive accident, the whole reactor vessel system behaves as a vertical spring, able to vibrate up or down. There is no clear definition as to the amount of additional mass that should be added to the head. To resolve this difficulty, a sensitivity study of the sodium slug upward force with different head masses was performed.

The cases studied ranged from one case in which the head was modeled with the head plus head-mounted components mass (1.26×10^6) to the other extreme case in which the head was modeled with the mass of the complete reactor system (3.97×10^6 lb). The sodium slug impact energy for the larger mass (3.97×10^6 lb) was less than 1% greater than the impact energy for the mass of the head plus head mounted components (1.26×10^6 lb.). The smaller mass (head plus head mounted components) was used in the REXCO model for Structural Margin Beyond the Design Basis (SMBDB) 47
analysis because of the insensitivity of the results to the mass, and because of the non-conservatism of the larger mass model with regard to the load on the holdown bolt system. 47

Amend. 47
Nov. 1978

Question 020.18 (9.9.2)

Provide design criteria and single-failure analysis for the Emergency Chilled Water System (ECHWS). Provide, in table form, individual cooling requirements of various coolers served by the emergency chilled water system.

Response:

Since the original response was submitted to this question, the Emergency Chilled Water System (ECW) has become part of the Chilled Water System and is described in Section 9.7.2 of the PSAR. Section 9.7.2.1 contains the design basis for the Emergency Chilled Water System. Single failure analysis of the system is presented in Section 9.7.2.3 and Table 9.7-3 is revised to include individual cooling load requirements in addition to the existing component locations. 47

Amendment 54

List of Responses to NRC Questions

There are no new or updated Responses to Questions in this Amendment.