

Clinch River Breeder Reactor Plant

Final Technical Report

Volume 2 of 2

March, 1985

Prepared For

Clinch River Breeder Reactor Plant
Project Office
United States Department of Energy
DE-AC15-76CL50003

~~APPLIED TECHNOLOGY~~

Any further distribution by any holder of this document or of the data therein to third parties representing foreign interests, foreign governments, foreign companies and foreign subsidiaries or foreign divisions of U.S. companies should be coordinated with the Director, Office of Breeder Technology Projects, Department of Energy.

Distribution of this report is Unlimited David Hamrin OSTI 8/6/2019



**Rockwell
International**

Rocketdyne Division
6633 Canoga Avenue
Canoga Park, California 91304

GENERAL  ELECTRIC

NUCLEAR SYSTEMS TECHNOLOGY OPERATION
P.O. BOX 3508 • SUNNYVALE, CALIFORNIA 94088



Westinghouse

Advanced Energy Systems Division
Waltz Mill, P.O. Box 158, Madison, PA 15663

MASTER

~~Released for announcement
in ATF. Distribution limited to
participants in the LMFBR
program. Others request from
BRP DOE~~

9516120

DISCLAIMER

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

DISCLAIMER

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

Clinch River Breeder Reactor Plant

Final Technical Report

Volume 2 of 2

March, 1985

Prepared For

**Clinch River Breeder Reactor Plant
Project Office
United States Department of Energy
DE-AC15-76CL50003**

~~APPLIED TECHNOLOGY~~

~~Any further distribution by any holder of this document or of the data therein to third parties representing foreign interests, foreign governments, foreign companies and foreign subsidiaries or foreign divisions of U.S. companies should be coordinated with the Director, Office of Breeder Technology Projects, Department of Energy.~~



**Rockwell
International**

Rocketdyne Division
6633 Canoga Avenue
Canoga Park, California 91304

GENERAL  ELECTRIC

NUCLEAR SYSTEMS TECHNOLOGY OPERATION
P.O. BOX 3508 • SUNNYVALE, CALIFORNIA 94088



Westinghouse

Advanced Energy Systems Division
Waltz Mill, P.O. Box 158, Madison, PA 15663

MASTER

~~Released for announcement
in ATF Distribution limited to
participants in the LMFBR
program. Others request from
BAP DOE~~

9516120

INFORMATION CONCERNING USE OF THIS DOCUMENT

WARD-D--0340-Vol.2

TI85 028078

~~PRELIMINARY DOCUMENT~~

~~This document contains information of a preliminary nature prepared in the course of work for the U.S. Department of Energy. This information is subject to correction or modification upon the collection and evaluation of additional data.~~

DISCLAIMER

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

WESTINGHOUSE ELECTRIC CORPORATION
ADVANCED REACTORS DIVISION
BOX 158
MADISON, PENNSYLVANIA 15663

DISCLAIMER

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

WARD-D-0340

CLINCH RIVER BREEDER REACTOR PLANT

FINAL TECHNICAL REPORT

VOLUME 2 OF 2

NOTICE

This report contains information of a preliminary nature and was prepared primarily for internal use at the originating installation. It is subject to revision or correction and therefore does not represent a final report. It is passed to the recipient in confidence and should not be abstracted or further disclosed without the approval of the originating installation or USDOE Office of Scientific and Technical Information, Oak Ridge, TN 37830.

March 1985

~~APPLIED TECHNOLOGY~~

~~Any Further Distribution by any Holder of this Document or of the Data Therein to Third Parties Representing Foreign Interests, Foreign Governments, Foreign Companies and Foreign Subsidiaries or Foreign Divisions of U. S. Companies Should Be Coordinated with the Deputy Assistant Secretary for Breeder Reactor Programs, Department of Energy.~~

Report Integration: F. S. Frantz

Approved by:

Vijay K. Sazawal

V. K. Sazawal, Manager

DOE Advanced Reactor Programs

DOE Contract DE-AC15-76CL50003
Advanced Energy Systems Division
Westinghouse Electric Corporation
Madison, Pennsylvania 15663

MASTER

~~Released for announcement
in AEP Distribution limited to
participants in the LMFBR
program. Others request from
BRP DOE~~

Jaw

INDEX

	<u>Page</u>
Abstract	iii
Table of Contents	iv
List of Tables	ix
List of Figures	xv
Abbreviations and Acronyms	xxx1

ABSTRACT

This technical report summarizes the work done on the major systems of the Nuclear Steam Supply System (NSSS) of the Clinch River Breeder Reactor Plant (CRBRP). Emphasis is placed on describing the basis for design features and approaches rather than details of design; key problems are identified and the resolutions described and the final status of the systems/components is given together with outstanding items remaining at Project termination. Extensive references to the CRBRP data base are given at the end of each section or major sub-section. These references are available/accessible through the Department of Energy Technical Information Center, unless another source is identified.

TABLE OF CONTENTS

	<u>Page</u>
1.0 CLINCH RIVER BREEDER REACTOR PLANT	1
1.1 General Report Integration	1
1.2 Beyond-the-Design-Basis Analyses for the ACRS	4
1.3 Decay Heat Removal Using Natural Circulation	31
1.4 Resolution of Metal-Joining Problems	50
1.4.1 Fixturing & Fabrication Distortions	50
1.4.2 Bi-Metallic Welds and Weld Problem Resolutions	108
2.0 SODIUM FIRE PROTECTION	122
2.1 Functions and Description	122
2.2 Evaluation of System 26A Design	135
2.3 Key Problems and Resolutions	139
2.4 Final System Status	203
2.5 References	205
3.0 REACTOR	207
3.1 Fuel Rod and Assembly Design	207
3.2 Blanket Assembly Design	258
3.3 Removable Radial Shield Design	295
3.4 Heterogeneous/Homogeneous Core Comparison	312
3.5 Assembly Arrangement in Heterogeneous Core	329
3.6 Flux Tilt Considerations	336
3.7 Control Rod Systems	356
3.7.1 Primary Control Rod System	356
3.7.2 Secondary Control Rod System	388
3.8 Core Transient Design Considerations	400
3.9 Upper Internals Structure Design	426
3.10 Core Restraint Design	499
3.11 Thermal Striping Mitigations	520
3.12 Thermal Striping Design Limits	557
3.13 In-Vessel Flow Management	610
3.14 Reactor Internals Flow Induced Vibration	629
4.0 REACTOR CLOSURE HEAD STRUCTURAL MARGIN BEYOND THE DESIGN BASIS (SMBDB) DESIGN	644
4.1 Summary of Functions and Description	644
4.2 Evolution of Design	662
4.3 Key Problems and Resolutions	686
4.4 Final System Status	708
4.5 Conclusions	711
4.6 References	713

TABLE OF CONTENTS (Continued)

	<u>Page</u>
5.0 REACTOR REFUELING	715
5.1 Overall System Scope	715
5.2 Functions and Description	717
5.3 Evolution of System 41 Design	743
5.4 Key Problems and Resolutions	751
5.5 Final Design Status	770
5.6 References	780
6.0 PRIMARY HEAT TRANSPORT SYSTEM	787
6.1 Piping Arrangement Guidelines	787
6.1.1 Introduction	787
6.1.2 System Description	788
6.1.3 Design Requirements, Guidelines, and Methods	791
6.1.4 Snubber Minimization	800
6.1.5 Seismic Margin Assessments	803
6.1.6 Inelastic Analysis Utilization	806
6.1.7 References	808
6.2 Pipe Clamp Design	810
6.2.1 Introduction	810
6.2.2 Analysis Methods and Modeling Techniques	816
6.2.3 Clamp/Pipe Stiffness and Performance Data	827
6.2.4 Test Programs	831
6.2.5 Primary/Secondary Stress Classification	837
6.2.6 Final Design Status	840
6.2.7 References	840
7.0 INTERMEDIATE HEAT TRANSPORT SYSTEM	843
7.1 Intermediate Heat Transport System and Piping Design	843
7.1.1 Summary of Functions and Design Description	843
7.1.2 Evolution of Design	847
7.1.3 Key Problems and Resolutions	850
7.1.4 Final System Status	863
7.1.5 References	865
7.1.6 Major ECPs Affecting IHTS Design	869
7.2 Leak Jacket Design	871
7.2.1 Introduction	871
7.2.2 Background	872
7.2.3 Design Basis Leak Rationale	874
7.2.4 Leak Jacket Design Requirements and Features	879

TABLE OF CONTENTS (Continued)

	<u>Page</u>
7.2.5 Leak Jacket Design Evaluation	819
7.2.6 Conclusions	895
7.2.7 References	895
7.3 Sodium Pumps	898
7.3.1 Summary of Functions and Description	898
7.3.2 Evolution of Design	898
7.3.3 Key Problems and Resolutions	903
7.3.4 Final Status of the Sodium Pump	907
7.3.5 References	912
7.3.6 List of Major ECPs	913
8.0 STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM	916
8.1 Summary of Functions and Design Description	916
8.2 Design Evolution of the Steam Generator Auxiliary Heat Removal System	920
8.3 Key Problems and Resolution	927
8.4 Final System Status	937
8.5 References	944
8.6 Major Engineering Change Proposals (ECPs) Affecting Design	947
9.0 STEAM GENERATOR SYSTEM	949
9.1 Summary of Functions and Design Description	949
9.2 Steam Generator System Design Evolution	955
9.3 Key Problems and Resolutions	968
9.4 Final System Status	985
9.5 References	989
9.6 Major ECPs Affecting SGS Design	993
10.0 REACTOR HEAT TRANSPORT INSTRUMENTATION AND ELECTRICAL HEATING AND CONTROL	995
10.1 Reactor Heat Transport Instrumentation	995
10.1.1 Summary of Functions and Description	995
10.1.2 Evolution of Design	997
10.1.3 Key Problems and Resolutions	998
10.1.4 Final System Status	1000
10.1.5 References	1002
10.1.6 Major ECPs Affecting System 56 Design	1004
10.2 Piping and Equipment Electrical Heating	1006
10.2.1 Summary of Functions and Description	1006
10.2.2 Evolution of Design	1008

TABLE OF CONTENTS (Continued)

	<u>Page</u>
10.2.3 Key Problems and Resolutions	1009
10.2.4 Final System Status	1010
10.2.5 References	1013
10.2.6 Major ECPs Affecting System 68 Design	1014
 11.0 AUXILIARY LIQUID METAL SYSTEM	 1015
11.1 Overall System Scope	1015
11.2 Functions and Description	1015
11.3 Evolution of System 81 Design	1030
11.4 Key Problems and Resolutions	1045
11.5 Final System Status	1073
11.6 References	1085
 12.0 INERT GAS RECEIVING AND PROCESSING	 1087
12.1 Functions and Description	1087
12.2 Evolution of System 82 design	1100
12.3 Key Problems and Resolutions	1113
12.4 Final System Status	1135
12.5 References	1144
 13.0 IMPURITY MONITORING AND PROCESSING	 1147
13.1 Functions and Description	1147
13.2 Evolution of System 85 Design	1158
13.3 Key Problems and Resolution	1167
13.4 Final System Status	1169
13.5 References	1176
 14.0 PLANT CONTROL SYSTEM DESIGN	 1179
14.1 Summary of Functions and Description	1179
14.2 Evolution of Design	1183
14.3 Key Problems and Resolution	1189
14.4 Final System Status	1196
14.5 References	1197
 15.0 PLANT PROTECTION SYSTEM	 1198
15.1 Summary of Functions and Description	1198
15.2 Evolution of Design	1208
15.3 Key Problems and Resolution	1211
15.4 Final System Status	1226
15.5 References	1227
15.6 Major ECPs Affecting PPS Design	1227

TABLE OF CONTENTS (Continued)

	<u>Page</u>
APPENDIX A DETAILED DESCRIPTION OF THE THERMAL STRIPING METHODOLOGY	1228
APPENDIX B THERMAL STRIPING DAMAGE CHARACTERIZATION	1254
APPENDIX C GENERAL HCDA PROGRESSION PATHS	1258

LIST OF TABLES

<u>Table No.</u>	<u>Title</u>	<u>Page</u>
1-1	Report Organization and Contributors	2
1-2	Maximum Core Outlet Sodium Temperature for Different Pipe Rupture Locations	10
1-3	Reactivity Balance and Estimated Core Temperatures for Hypothetical Core Support Failure Conditions	18
1-4	Operational Assumptions	33
1-5	T/TS Weld Problems and Their Resolution Through Row 5	73
1-6	Defect Correction	74
2-1	Essential Liquid Metal Fire Protection Features Provided by Other Systems	124
2-2	Cell Designation and Sodium Fire Protection Features	127
2-3	Comparison of Test Parameters and SGB Design Basis Leak Parameters	150
2-4	CRBRP Fire Test Results	154
2-5	Sodium Fire Parameter Survey	158
2-6	Large-Scale Fire Suppression Test Conditions	189
2-7	Summary of Fire Suppression Test Results	189
2-8	Oxygen Content of Cores from Frozen Sodium in the Catch Pan Area	193
2-9	Material and Equipment Recommended for Sodium Cleanup	197
3-1	CRBRP Fuel Assembly Functional Requirements	208
3-2	CRBRP Fuel Assembly Comparison with FFTF	210
3-3	CRBRP Fuel Rod Comparison with FFTF	213
3-4	CRBRP Fuel Rod Design Details	214
3-5	CRBRP Fuel Assembly Design Details	217
3-6	Requirement-Damage Severity Limits	223

LIST OF TABLES (Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page</u>
3-7	CRBRP Fuel Rod Duty Cycle Normal Operation, Fuel Damage Severity Limit--No Significant Loss of Effective Lifetime	224
3-8	CRBRP Fuel Rod Duty Cycle Anticipated (Upset) Events	225
3-9	CRBRP Fuel Rod Duty Cycle Unlikely (Emergency) Events	226
3-10	CRBRP Design Basis (Enveloping) Transients	227
3-11	CRBRP Fuel and Blanket Rod Mechanical Design Limits	229
3-12	Design Procedure for Steady State Creep Analysis	232
3-13	Cumulative Damage Function (CDF)	234
3-14	Fuel Assembly Structural Loads	239
3-15	Core Assembly Structural Criteria	240
3-16	Bases for Current Mechanical Design Limits	241
3-17	Guidelines for Structural Analyses	242
3-18	CRBR Core Assembly Structural Inelastic Criteria and Limits	243
3-19	Design Evaluations - Key Fuel Results	245
3-20	Summary of Thermal Transient Data for Fuel Assembly Rods	248
3-21	Core Assembly Structural Analysis	252
3-22	CRBRP Fuel Assembly Structural Analysis Summary	253
3-23	CRBRP Blanket Assembly Functional Requirements	259
3-24	CRBRP Blanket Assembly Design Details	261
3-25	Blanket Assembly Design Description	264
3-26	CRBRP Blanket Rod Design Details	265
3-27	Design Evaluations - Key Blanket Results	269
3-28	Steady State CDF Value for the Hot Rod of Blanket Assembly 201 of the First Core	270

LIST OF TABLES (Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page</u>
3-29	Summary of Thermal Transient Data for Blanket Rod Region	272
3-30	Core Assembly Structural Analysis	276
3-31	CRBRP Blanket Assembly - Structural Analysis Summary	277
3-32	CRBRP Removable Radial Shield Assembly Functional Requirements	296
3-33	CRBRP Removable Radial Shield Assembly Structural Analysis Summary	301
3-34	Reactor System Design Requirements	303
3-35	CRBRP Breeding Ratio - Homogeneous Core	315
3-36	Comparison of the Major Neutronic Characteristics of the CRBRP Homogeneous and Heterogeneous Cores	322
3-37	Comparison of Major CRBRP Nuclear Design Parameters in AFMS and AFMS-M Configurations	334
3-38	Hot Rod Parameters for Uncompensated Single Rod Withdrawal at 15% Overpower with Long Trip Line	345
3-39	Event Classification and Cladding Damage Severity Limits	401
3-40	Summary of Preliminary Design Limits	402
3-41	Assembly Umbrella Design, Duty Cycle Events and Frequency	405
3-42	Names of Umbrella Transients Described in OPDD-10	408
3-43	Specific Design Requirements for the Reactor Shutdown Systems	410
3-44	Core Map, Rev. 4, Primary Control Rod System Scram Insertion Requirements	413
3-45	Core Map, Rev. 4, Secondary Control Rod System Scram Insertion Requirements	414
3-46	CRBR Fuel Assemblies Rod Temperature Nuclear Uncertainty Factors with and Without Control Assembly Influence	418

LIST OF TABLES (Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page</u>
3-47	CRBR Inner/Radial Blanket Assemblies Rod Temperature Nuclear Uncertainty Factors	419
3-48	CRBR Fuel/Blanket Assemblies Rod Temperatures Engineering Uncertainty Factors	420
3-49	Evolution of UIS Design	444
3-50	Major Changes to UIS Design	445
3-51	Minimum Positive Margins for UIS Components	471
3-52	UIS Mixing Chamber Water Table Striping Results	527
3-53	Maximum Striping from Local Interstitial Flow Test	534
3-54	Summary of Interstitial Flow Striping Tests	537
3-55	Summary of Test Results - Y-Junction Striping Tests	540
3-56	Summary of Observed Core Components Thermal Striping Data - PEOV	552
3-57	Thermal Striping-Results	604
3-58	Primary Control Assembly Combination 31/75	605
3-59	Fuel and Blanket Assemblies Orificing Zones Average Flows	621
3-60	Flow Allocation to Non-Fueled Components/Flow Paths	625
3-61	Flow Vibration of CRBRP Internals - General Survey	632
4-1	Partition of Energy for Structural Margin Beyond the Design Base	651
4-2	Phase II Simple Model Experimental Tests (1/20 scale)	667
4-3	Status of Development Program for HCDA Load Simulation	709
5-1	Component Safety and Code Classification--System 41	720
5-2	Reactor Refueling System Component Status	773
5-3	Reactor Refueling System Equipment Ready for Procurement but Delayed by National Policy Debate on Funding Restrictions	778

LIST OF TABLES (Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page</u>
6-1	HTS Piping within the Reactor Containment Building	792
6-2	Design Data for HTS Piping Within The Reactor Containment Building	793
6-3	CRBRP HTS Large Diameter Clamp Drawings	818
6-4	Structural Analyses for CRBRP HTS Clamps	820
6-5	CRBRP Clamp/Pipe Stiffnesses for 24-inch and 36-inch HTS Clamps	829
6-6	Load Ratings for CRBRP In-Containment HTS Pipe Clamps	830
7-1	Plant Pump Steady State Functional Requirements	898
7-2	Pending Changes to Plant Sodium Pump Documentation	909
7-3	Major ECPs for CRBRP Sodium Pumps	914
8-1	AFW Turbine Drive Steam Supply Line Heating; An Evaluation of Alternative Heating Methods	931
11-1	Component Safety and ASME Classification - System 81	1020
11-2	Elements Which Could Cause Valve Failure	1065
11-3	Component Status	1077
11-4	Approved Engineering Change Proposals (ECPs) Not Completely Incorporated at Termination	1079
11-5	Unapproved ECP's Outstanding at Termination	1081
12-1	Safety and Code Classifications of System 82 Components	1090
12-2	Summary Scenario for Post-Accident Cleanup of a Typical Cell Containing High Levels of Radioactivity	1134
13-1	Component Safety and ASME Classification, System 85	1149
15-1	Reactor Shutdown System Protective Subsystems	1200
15-2	Protective Subsystem Summary	1201
15-3	Reactor Shutdown System Design Basis Events	1203

LIST OF TABLES (Continued)

<u>Table No.</u>	<u>Title</u>	<u>Page</u>
15-4	Specific Design Requirements for the Reactor Shutdown Systems	1207
15-5	Event Classification and Cladding Damage Severity Limits	1209
15-6	Instrumentation Common to the Control and Protection Systems	1217
15-7	Protection/Control Interaction	1218
A-1	RRS Evaluation Using Westinghouse Equation	1249
A-2	RRS Evaluation Using EG&G Equation	1250

LIST OF FIGURES

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
1-1	Failure Location for Guillotine Rupture Cases Evaluated	6
1-2	Variation in Power and Flow for Case I	11
1-3	Reactor Response for Case I	12
1-4	Reactor Vessel Na Level Above Outlet Nozzle for Case I	14
1-5	Geometry of CRBRP System Before and After a Hypothetical Core Support Cone Failure	16
1-6	Natural Circulation	21
1-7	CRBRP Loss of Heat Removal	22
1-8	Reactor Scram from Full Power to Natural Sodium Circulation	23
1-9	Bulk Na Temperature, Reactor Scram from Full Power to Natural Sodium Circulation	26
1-10	Cell Temperature as a Function of Time After Loss of Cooling (Station Blackout)	28
1-11	The Elevation Differences in Major Components Provide a Natural Circulation Capability for CRBRP	36
1-12	CRBRP Natural Circulation Verification Program	40
1-13	Cycling Controller - Temperature Response	42
1-14	Cycling Controller - IHTS Flow Response	43
1-15	Cycling Controller - PHTS Flow Response	44
1-16	Controller - Core Temperature Response	46
1-17	Controller - IHTS Flow Response	47
1-18	Controller - Steam Drum Pressure Response	48
1-19	A Cut-Away Isometric View of the Core Support Plate Assembly	52
1-20	A One-Fourth Scale Sketch of the Proposed and the Production Joint Geometries	55
1-21	Core Support Plate Sub-Assemblies	57

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
1-22	Core Support Plate Assembly	58
1-23	Isometric View of Procedure Qualification Test Assembly	59
1-24	Isometric View of Core Support Structure	62
1-25	Topside Welding Setup - Core Support Plate to Reactor Vessel Core Support Cone	63
1-26	Topside Fit-Up and Welding Layout	65
1-27	Backside Welding Setup	66
1-28	Final Machining	68
1-29	Tube/Tubesheet Weld Technique	69
1-30	Tube Installation, Welding and Inspection Process	71
1-31	Typical Riser Assembly Pressure Boundary (Axisymmetric Cross-Section)	109
1-32	Bi-Metallic Weld	111
2-1	CRBRP Inerted Cell Liner System	126
2-2	Typical Catch Pan Fire Suppression Deck Arrangement	129
2-3	Catch Pan System Arrangement, SG Loops 1 and 3	133
2-4	Catch Pan System Arrangement, SG Loop 2	134
2-5	"Q" Deck Arrangement Detail	143
2-6	Vent Pipe	144
2-7	Test Facility Configuration	151
2-8	Test Article	152
2-9	Air Benchmark Test No. 2	155
2-10	Air Benchmark Test No. 2	157
2-11	Modified Piping Insulation Design	166
2-12	LSSS Test Article	167

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
2-13	Large-Scale Sodium Fire Test, Sodium Flow Rates Versus Time	168
2-14	Sodium Fire Test, Cell Gas Temperature	170
2-15	Sodium Fire Test, Cell Gas Average Temperature	171
2-16	Sodium Fire Test, Building Concrete Wall Surface Temperature	172
2-17	Sodium Fire Test, Concrete Ceiling Surface Temperature	173
2-18	Sodium Fire Test, Cell Oxygen Concentration	174
2-19	Sodium Fire Test, Concrete Floor Surface Temperature Under Catch Pan	175
2-20	Sodium Fire Test, Sodium Pool Temperature	176
2-21	Effect of Pool Burning on Sodium Temperature	177
2-22	Structural Temperature Predictions	179
2-23	Sodium Spill Area on Q-Deck Surface	180
2-24	Fire Suppression Deck Temperatures	181
2-25	I-Beam Temperatures	182
2-26	Sodium Fire Test, I-Beam Temperatures	183
2-27	Sodium Fire Suppression Test Article	187
2-28	Schematic of Fire Suppression Test Article	188
2-29	Configuration of Sodium Spill	190
2-30	Initial Condition of Catch Pan Floor	196
2-31	Initial Condition of Catch Pan Floor Near Support Pan	196
2-32	Addition of Dry Ice to Catch Pan Residues	198
2-33	Inerting Drum and Sealing Drum Liner	199
2-34	Opened Q-Deck Section for Cleaning Catch Pan	200
2-35	Q-Deck Drain Pipe Removal	201

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
3-1	CRBRP Fuel Assembly	209
3-2	CRBRP Fuel Rod	212
3-3	Core Assembly Design Process	220
3-4	Cumulative Mechanical Damage Function Due to Steady State Operation, Anticipated Events and Unlikely Event Example	235
3-5	Fuel Assembly Loads	237
3-6	Maximum Fuel Rod Bundle - Duct Diametral Clearance and Interference Versus Time - F/A 45, First Core	249
3-7	Maximum Duct Radial Dilation Across Flats Versus Axial Location at End-of-Life for F/A 45	250
3-8	CRBRP Blanket Assembly	260
3-9	CRBRP Blanket Rod	263
3-10	Maximum CRBR Blanket Assembly Bundle - Duct Interference and Clearance Versus Time	273
3-11	Duct Dilation Versus Axial Height for IB/A 67 and RB/A 201	274
3-12	CRBRP Duty Cycle - Load Following Characteristics for Typical Day of Power Operation	286
3-13	Typical EBR-II Transient Cycles During Run 112 after Startup and Hold Time	288
3-14	Removable Radial Shield Assembly	297
3-15	Removable and Fixed Radial Shielding Configuration; Elevation View	304
3-16	Location of Removable Radial Shield Assemblies	306
3-17	RRSA Material Configuration	307
3-18	Duct Surveillance Tensile Specimen	308
3-19	Nesting of Tensile and Charpy V Notch Specimens in Tube	309

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
3-20	Location of Duct Surveillance Specimens	310
3-21	Clinch River Breeder Reactor Homogeneous Core	313
3-22	Clinch River Breeder Reactor Heterogeneous Core	319
3-23	Evolution of the CRBRP Heterogeneous Core Arrangement	332
3-24	Percent Increase in 36-Inch Integrated Power Shape Due to 15¢ Uncompensated R7C Control Rod Runout (~1.1 Inches from Half Inserted). CRBRP (4), Beginning of Cycle Four.	339
3-25	Percent Increase in 36-Inch Integrated Power Shape Due to 15¢ Uncompensated R7C Control Rod Runout (~2.1 Inches from 8 Inches Inserted). CRBRP (4), End of Cycle Four	340
3-26	Relative Power Variation for Fuel Hot Rod Cases with and without Flux Tilt Effect.	342
3-27	Effect on Fuel Hot Rod Maximum Cladding Temperature of Power Rise Rate with and without Flux Tilt Effect	343
3-28	Effect on Fuel Hot Rod Maximum Fuel Temperature of Power Rise Rate with and without Flux Tilt Effect.	344
3-29	CDF Histogram for a Single Control Assembly Withdrawal Event	347
3-30	CDF Cumulative Distribution for a Single Control Assembly Withdrawal Event	348
3-31	Percent Molten Fuel Versus Power Level to Typify Molten Fuel Increase with Increasing Power	349
3-32	Rod Block System	352
3-33	Reactor Cross Section	358
3-34	Primary Control Rod Drive Mechanism (PCRDM)--Upper Assembly	360
3-35	Primary Control Rod Drive Mechanism (PCRDM)--Lower Assembly	362
3-36	Primary Control Rod Driveline (PCRD)	364

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
3-37	Primary Control Assembly (PCA)	366
3-38	Shield and Seismic Support	368
3-39	Disconnect Actuating Tool for the Primary Control Rod System	370
3-40	Secondary Control Rod Schematic	390
3-41	Secondary Control Assembly	392
3-42	Comparison of System 31 SCRS Absorber Pin Designs	395
3-43	Comparison of System 31 - SCRS Control Assembly Filler Rod to Absorber Pin Spacing	396
3-44	Longest and Quickest Flow Coastdown Rates for CRBR Primary System	415
3-45	Relative Axial Locations for Hot Rod Temperatures	417
3-46	Graphical Illustrations of Semi-Statistical Method	421
3-47	Upper Internals Structure	427
3-48	Chimney Effluent Temperature Time History	436
3-49	Upper Internals Structure - Elevation	439
3-50	Upper Internals Structure - Plan View	440
3-51	Upper Internals Structure - Outlet Flow Module and Chimney	441
3-52	Elevation Schematic of the Upper Internals Structure	447
3-53	Instrumentation Post Installation	451
3-54	Instrumentation Post (Bench Subassembly)	452
3-55	Method of Positioning Control Rod Shroud Tube	455
3-56	Instrumentation and Conduit Installation	456
3-57	Upper Internals Structure	457
3-58	Photograph of Full Size Upper Internals Structure Mockup	461

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
3-59	Prototypic Fuel and Blanket Exit Temperatures and Reactor Flow Rate Following a Reactor Trip	480
3-60	Schematic of Integral Reactor Flow Model	483
3-61	Evaluation of Thermal Striping Fatigue is a 5-Step Process	521
3-62	Selected CRBRP Temperature Distributions (°F)	524
3-63	Lower Shroud Tube Water Table Model	528
3-64	Typical Temperature Trace from Lower Shroud Tube Water Table Results	530
3-65	Test Set-up on Water Table of Local Interstitial Flow Test Rig	532
3-66	Plan View of Interstitial Flow Test Rig	533
3-67	Plexiglass Model for Interstitial Flow Testing	536
3-68	Y-Junction Model Test Setup	539
3-69	ANL 1/10 Scale Outlet Plenum Thermocouple Map	542
3-70	ANL 1/10 Scale Outlet Plenum Thermocouple Map	543
3-71	ANL 1/10 Scale Outlet Plenum Thermocouple Map	544
3-72	CRBRP Integral Reactor Flow Model - Phase II	546
3-73	Instrumentation Post and Chimney Model	549
3-74	Lower Shroud Tube Model	550
3-75	Structure Internal Striping Temperature Amplitudes	553
3-76	Fuel/Blanket Assembly Outlet Nozzle Design	559
3-77	Thermal Striping Mechanism	560
3-78	Fuel & Blanket Assembly Outlet Nozzle	562
3-79	Section of F/A Outlet Nozzle (Drawing 1182E45)	563
3-80	Schematic of Na Flow Pattern from Reactor Core	565

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
3-81	Core Assembly Striping Potential	572
3-82	Striping Temperature History of Fuel and Radial Blanket Assembly Interface	573
3-83	Seven Assembly Striping Test - FA-08; TC Location FN-7N	575
3-84	Seven Assembly Striping Test - FA-08; TC Location FN-7JJ	576
3-85	7 Assembly Test Results	577
3-86	Saturation of Maximum Striping with Time	578
3-87	Probability of Measuring Larger Striping in Longer Test Is Small	579
3-88	Mixing of Hot and Cold Fluid in a Channel Indicates $\geq 50\%$ Decrease of Striping Near Wall	582
3-89	Calculated Surface Striping	584
3-90	Calculated Striping Film Coefficient-H	586
3-91	Qualitative Comparison of Thermal Film Coefficients	587
3-92	Failure of SA 316SS In Air at 538 to 566°C	588
3-93	Fatigue of SA 316SS at 538 to 566°C, In-Sodium Data Versus In-Air Correlation	589
3-94	Fatigue of SA 316SS at 538 to 566°C, In-Sodium Data Versus In-Sodium Correlation with In-Air Correlation Asymptote	591
3-95	Comparison of In-Air Fatigue Limits at 10^8 Cycles	592
3-96	Summary of 316SS TSTF Thermal Striping Results	593
3-97	Construction of Elastic-Plastic Strain Fatigue Design Curve	595
3-98	Striping Fatigue Analysis Method	596
3-99	Record of Minimum and Maximum Temperature Striping	597
3-100	Rain-Flow Cycle Counting Method	599

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
3-101	Use of a Constant Amplitude S-N Curve with the Miner Rule to Predict Fatigue Life for Variable Amplitude Loading	600
3-102	Allowable Gradient (Striping) Correlations	602
3-103	CRBR In-Vessel Flow Paths: Flow Allocation and Its Basis	611
3-104	CRBRP Core Orificing Map	620
3-105	Complete Schematic of CRBRP In-Vessel Flow Paths	626
4-1	Thermal Source for Structural Margin Beyond the Design Base	648
4-2	Pressure Volume Relationship for SMBDB Loading	649
4-3	Slug Force on Under Head Shielding of Reactor Vessel Closure Head	652
4-4	Reactor Closure Head Assembly	654
4-5	Closure Head Assembly Plan View	655
4-6	Reactor Schematic	656
4-7	Reactor Closure Head Elevation	658
4-8	CRBRP Riser Layout	660
4-9	Margin Ring Sections	661
4-10	Schematic of Static Vessel Head Apparatus, Test SM1	668
4-11	1/20 - Scale Models of the CRBR	670
4-12	Schematic of Test Fixture for SM7	673
4-13	Differences in SM7 and SMB Shield Plate Designs	674
4-14	Shield Plate Designs for SM7 and SM8	675
4-15	Profiles of Head at Six Pressures; Static Test SM1	680
4-16	Profiles of Head at Seven Pressures for SM7	684
4-17	Profiles of Head at Seven Pressures for SM8	688

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
4-18	SM-8 Model Plug Configuration of Failure (2010 psi)	689
4-19	Design Modification Logic CRBRP Closure Head SMB08 Capacity Enhancement	672
4-20	CRBR RV and Rotating Plugs Absolute Displacements Due to HCDA	673
4-21	CRBR RV and Rotating Plugs Absolute Displacements Due to HCDA	675
4-22	Relative Displacements Across Point of Disengagement for Tests SM7 and SM8	676
4-23	Projected Closure Head Deflections and Interferences at 2700 psi	677
4-24	Projected Closure Head Deflections and Interferences at 3000 psi	678
4-25	Machining Modifications Proposed to Eliminate Interferences at the Node 5-6 Interface	679
4-26	Machining Modifications Proposed to Eliminate Interferences at the Node 5-6 Interface	680
4-27	Projected Closure Head Deflections at 2700 psi with Prototype Radial Restraint at the LRP-Reactor Vessel Interface	682
4-28	Projected Closure Head Deflections at 3000 psi with Prototype Radial Restraint at the LRP-Reactor Vessel Interface	683
4-29	Acceptance Criteria	684
4-30	Joint Shear Strength	686
4-31	Flange Stud Capacity	687
4-32	Potential Gap Reduction Factors	688
4-33	Seismic Model: From Calculation LRA-SA-78-77	689
4-34	Acceleration Versus Frequency for Linear and Non-Linear Analyses	690

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
4-35	Head Seismic Response with Reduced Radial Gaps Between Plugs	691
4-36	Plug Strain Summary	693
4-37	Head Energy Input	695
4-38	Propagation of Pressure Waves Caused by Instantaneous Complete Gate Closure	696
4-39	Slug Bulk Compression Energy	697
4-40	Pressure Volume Change for Original and Modified Head Designs	699
4-41	Head System Energy Absorption Capability	701
4-42	SMBDB Analysis Model	702
4-43	Force Time History Head Up/Force CSS Down/Force	703
4-44	Vessel Flange SMBDB Response	704
4-45	Reactor Vessel Support and Ledge Dynamic Force-Deflection Relationship (75 MJ Slug Load)	705
4-46	Closure Head System Energy Absorption Summary	706
4-47	Summary Logic Diagram for CRBR Closure Head Modification SMBDB Enhancement	710
5-1	Reactor Refueling System (Perspective)	718
5-2	In-vessel Transfer Machine (IVTM)	722
5-3	Auxiliary Handling Machine (AHM)	724
5-4	AHM Floor Valve	725
5-5	Ex-vessel transfer Machine (EVTM)	727
5-6	EVTM Gantry-Trolley	729
5-7	Core Component Pot (CCP)	731
5-8	Ex-vessel Storage Tank (EVST)	733
5-9	Fuel Handling Cell (FHC)	735

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
5-10	New Fuel Unloading Station	737
5-11	New Core Assembly Inspection Equipment	739
5-12	Reactor Refueling System (RRS) Control Concept	740
5-13	Refueling Communications Center (RCC) Block Diagram	742
5-14	Reactor Refueling System Drawing Tree	771
6-1	General Arrangement of CRBRP Heat Transport System	789
6-2	Loop 1 of CRBRP Heat Transport System Piping	790
6-3	Initial HTS Flexible Clamp Design	812
6-4	Typical HTS Horizontal Pipe Clamp Assembly (Type B)	814
6-5	Typical Horizontal Restraint Assembly for CRBRP HTS Piping	815
6-6	Typical HTS Vertical Pipe Clamp Assembly	817
6-7	2-D Clamp/Pipe Interaction Model	822
6-8	3-D Clamp/Pipe Interaction Model	826
6-9	Typical Clamp and Pipe Super Elements	828
7-1	CRBRP Intermediate Heat Transport System Schematic	845
7-2	IHTS Leak Jacket Design Concept	882
7-3	IHTS Leak Jacket Details and Interfaces	883
7-4	IHTS Inner Jacket Attachment Detail	884
7-5	IHTS Z-Ring Details at Pipe Clamp	885
7-6	IHTS Insulation Leak Test Specimen Assembly	888
7-7	IHTS Modified Z-Ring Seal Over Marinite Ring Halves	889
7-8	CRBRP Prototype Pump - Cutaway View	901
7-9	CRBRP Prototype Pump	902

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
8-1	Steam Generator Auxiliary Heat Removal System Schematic	917
8-2	Process Streams in Relationship to Major Components Schematic	921
8-3	Steam Supply Line Schematic	929
8-4	Control Logic for Turbine Driven AFWP	938
8-5	Control Logic for Turbine Driven AFWP	939
9-1	CRBRP Steam Generator System Schematic	952
9-2	CRBRP Steam Drum	959
11-1	System 81 Schematic - Final Configuration	1017
11-2	Flow During EVST Cooling Operation	1027
11-3	Flow During EVST-Plus-DHRS	1028
11-4	System 81 Schematic - Initial Conceptual Design	1031
11-5	System 81 Schematic - Final Conceptual Design	1035
11-6	Small Insulated Pipe Clamp Assembly	1050
11-7	EM Pump Header Configurations	1057
11-8	EM Pump Throat Configurations	1058
11-9	EM Pump Assembly	1059
11-10	Flow Mapping at 276 V and 800°F	1060
11-11	Flow Mapping at 368 V and 800°F	1060
11-12	EM Pump Design Limits at 600°F	1061
11-13	Typical Thermal Transient Valve	1062
11-14	Typical Thermal Transient Operator Assembly	1063
11-15	Valve Disc-to-Stem Design	1069
11-16	Bonnet Modifications	1071
11-17	System 81 Documentation Diagram	1075

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
11-18	System 81 Electrical, Instrumentation, and Control (EI&C) Documentation Diagram	1076
12-1	Simplified Flow Schematic RSB Argon Distribution	1092
12-2	Simplified Flow Schematic RCB Argon Distribution	1093
12-3	Simplified Flow Schematic SGB Argon Distribution	1094
12-4	Simplified Flow Schematic RSB/RCB Nitrogen Distribution	1096
12-5	Simplified Flow Schematic	1097
12-6	Radioactive Argon Processing Subsystem	1099
12-7	Cell Atmosphere Processing Subsystem	1101
12-8	Use of CAPS for Post-Accident Cleanup	1123
12-9	Dose in Cell 307 After Postulated Rupture in Cell 379	1124
12-10	Site Boundary Doses After a Postulated Rupture in Cell 379	1125
12-11	Calculated Kr and Xe Inventory as a Function of Time Following a Large Postulated Release in Cell 379 Followed by a Purge to CAPS at 15 cfm	1131
12-12	System 82 Documentation Diagram	1137
12-13	System 82 I&C Documentation Diagram	1138
13-1	Primary Plugging Temperature Indicator Package	1150
13-2	Plugging Temperature Indicator	1151
13-3	Primary Sodium Sampling Package	1153
13-4	Multipurpose Sampler	1154
13-5	Ex-vessel Sodium Sampling Package	1155
13-6	Intermediate Sodium Characterization Package	1157
13-7	System 85 Drawing Tree	1171
13-8	System 85 I&C Document Diagram	1172

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
14-1	CRBRP Plant Control Schematic	1180
14-2	CRBRP Part Load Profile	1181
15-1	Control/Protection Interface	1213
15-2	Control/Protection Interface	1214
15-3	Effect of PM Flowmeter Time Constant on Signal Noise	1223
A-1	Fatigue of SA 316SS In Air at 538 to 566°C	1230
A-2	Fatigue of SA 316SS at 538 to 566°C, In-Sodium Data Versus In-Sodium Correlation with In-Air Correlation Asymptote	1231
A-3	Illustration of Fluid Temperature History as Input to "TAP A" (as Boundary Temperatures)	1232
A-4	Illustration of Instantaneous Structural Temperature Amplitude (ΔT^*) History as Input to "STRIPE"	1233
A-5	Illustration of Total Thermal Striping Amplitude (ΔT^*) History as Input to "RAINFAT"	1234
A-6	Element Representation of Stress-Strain Curves	1236
A-7	Use of Elements and Availabilities to Count Cycles and Simulate Stress-Strain Behavior	1237
A-8	Fatigue Design Curve Sodium, -2σ	1238
A-9	Study of the Effect of Film Coefficient on Damage	1240
A-10	Typical Fuel Assembly Outlet Temperature Striping Potentials	1241
A-11	Typical Fuel Assembly Outlet Temperature Striping Potentials	1242
A-12	Study of How Damage Saturates with Time	1244
A-13	Fluid (Surface) Percent Striping	1245
A-14	How the Maximum Strain Range and Low Cycle Strain Ranges Effect Total Damage	1247

LIST OF FIGURES (Continued)

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
A-15	Constant Striping Potential Versus Fatigue Damage for Measured Fluid Striping Signals	1251
B-1	Hardness Profiles Through the Thickness Follow Temperature Profile	1255
B-2	TSTF Specimen Hardness Change Versus Depth Below Surface Due to Striping	1256
C-1	CRBRP HCDA Progression Paths	1259

ABBREVIATIONS AND ACRONYMS

Following is the alphabetical list of abbreviations and acronyms as they appear within this document:

ABHX	-	Air Blast Heat Exchanger
ACLP	-	Above Core Load Plane
ACRS	-	Advisory Committee on Reactor Safeguards
A-E	-	Architect-Engineer (Burns and Roe, Inc.)
AEC	-	U.S. Atomic Energy Commission (now split to DOE and NRC)
AFMS	-	Alternate Fuel Management Scheme
AFW	-	Auxiliary Feedwater Subsystem of SGAHRS
AHM	-	Auxiliary Handling Machine
AI	-	Atomics International Division of Rockwell International
AISI	-	American Iron and Steel Institute
ALARA	-	As Low As Reasonably Achievable
ANL	-	Argonne National Laboratory
ANSI	-	American National Standards Institute
APU	-	Atmosphere Purification Unit
ARPI	-	Absolute Rod Position Indicator
ASB	-	Auxiliary Steam Boiler
ASME	-	American Society of Mechanical Engineers
ASTM	-	American Society for Testing Materials
BA	-	Blanket Assembly
BDI	-	Bundle-Duct Interaction
BOC	-	Beginning of Cycle
BOL	-	Beginning of Life
BOP	-	Balance of Plant
BPFM	-	Bypass Flow Module

ABBREVIATIONS AND ACRONYMS (Continued)

B&PV	-	Boiler and Pressure Vessel
BTP	-	Branch Technical Position (NRC)
BWR	-	Boiling Water Reactor
CA	-	Control Assemblies
CAPS	-	Cell Atmosphere Processing System
CB	-	Core Barrel
CBI	-	Chicago Bridge and Iron Company
CCP	-	Core Component Pot
CCTV	-	Closed Circuit Television
CDF	-	Cumulative Damage Function
CFR	-	Code of Federal Regulations
CFS	-	Core Former Structure
CLCV	-	Cold Leg Check Valve
CLEM	-	Closed Loop Ex-vessel Machine (at FFTF)
CRBR	-	Clinch River Breeder Reactor
CRBRP	-	Clinch River Breeder Reactor Plant
CRD	-	Control Rod Driveline
CRDM	-	Control Rod Drive Mechanism
CSP	-	Core Support Plate
CSS	-	Core Support Structure (31)
CSS	-	Conditioning and Service Subsystem (41)
CTM	-	Cable Transfer Machine
CW	-	Cold Worked
DBL	-	Design Basis Leak
DDC	-	Direct Digital Control
DELT	-	Cumulative Damage Function Equivalent Limiting Temperature
D-G	-	Diesel Generator
DEG	-	Double Ended Guillotine
DHRS	-	Direct Heat Removal Service
DLS	-	Ductility Limited Strain
DNB	-	Departure from Nucleate Boiling
DOE	-	U.S. Department of Energy

ABBREVIATIONS AND ACRONYMS (Continued)

DRS	- Development Requirements Specification
EBR-II	- Experimental Breeder Reactor II (located in Idaho)
ECP	- Engineering Change Proposal
EDEG	- Equivalent Double Ended Guillotine
EFPD	- Effective Full Power Days
EM	- Electro-Magnetic
EOC	- End of Cycle
EOL	- End of Life
ERDA	- U.S. Energy Research and Development Administration (now DOE)
ESF	- Engineered Safety Feature
ESG	- Energy Systems Group of Rockwell International
ETEC	- Energy Technology Engineering Center (located in Santa Susanna, CA)
EVHS	- Ex-Vessel Handling Subsystem
EVS	- Ex-Vessel Storage
EVSS	- Ex-Vessel Storage Subsystem
EVST	- Ex-Vessel Storage Tank
EVTM	- Ex-Vessel Transfer Machine
FA	- Fuel Assembly
FAG	- FAG Bearing Corporation
FCMI	- Fuel Cladding Mechanical Interaction
FFTF	- Fast Flux Test Facility (located in Hanford, Washington)
FHC	- Fuel Handling Cell
FIV	- Flow Induced Vibration
FMEA	- Failure Modes and Effects Analysis
FRS	- Fixed Radial Shield
FTR	- Fast Test Reactor (FFTF)
FTSA	- Fuel Transfer and Storage Assembly
FV	- Floor Valve
FWSDR	- Final Working System Design Review
GE	- General Electric Company
HAA	- Head Access Area

ABBREVIATIONS AND ACRONYMS (Continued)

HB	-	Horizontal Baffle
HBM	-	Horizontal Boring Mill
HBRR	-	Horizontal Baffle Retaining Ring
HBSO	-	Hot Bare Strap-On
HCDA	-	Hypothetical Core Disruptive Accident
HEOL	-	Hanford Engineering Development Laboratory
HTS	-	Heat Transport System
HUD	-	U.S. Department of Housing and Urban Development
H&V	-	Heating and Ventilating
I&C	-	Instrumentation and Control
IBA	-	Inner Blanket Assembly
ICD	-	Interface Control Document/Drawing
IEEE	-	Institute of Electrical and Electronics Engineers, Inc.
IFT	-	Instantaneous Fault Trips
IGRP	-	Inert Gas Receiving and Processing System
IHTS	-	Intermediate Heat Transport System
IHX	-	Intermediate Heat Exchanger
IMAS	-	Impurity Monitoring and Analysis System
IPFM	-	Inlet Plenum Feature Model
IRFM	-	Integral Reactor Flow Model
IRP	-	Intermediate Rotating Plug
ISI	-	In-service Inspection
IVHS	-	In-Vessel Handling Subsystem
IVTM	-	In-Vessel Transfer Machine
KE	-	Kinetic Energy
KSTF	-	Key System Task Force
LCC	-	Local Control Center
LD	-	Leak Detector
LDM	-	Leak Detection Module
LEFM	-	Linear Elastic Fracture Mechanics
LHGR	-	Linear Heat Generation Rate
LIM	-	Lower Inlet Module

ABBREVIATIONS AND ACRONYMS (Continued)

LLM	-	Liquid Level Monitor
LLTR	-	Large Leak Test Rig
LMFBR	-	Liquid Metal Fast Breeder Reactor
LN ₂	-	Liquid Nitrogen
LOD	-	Lines of Defense
LOEP	-	Loss of Electrical Power
LOF	-	Loss-of-Flow
LRM	-	Lead Reactor Manufacturer
LRP	-	Large Rotating Plug
LWR	-	Light Water Reactor
MCC	-	Master Control Center
MCR	-	Main Control Room
MEB	-	Mechanical Engineering Branch of the NRC
MEFS	-	Moderate Energy Fluid System
MI	-	Mineral Insulated
MPS	-	Multipurpose Sampler
NCVP	-	Natural Circulation Verification Program
NDHX	-	Natural Draft Heat Exchanger
NDT	-	Non-Destructive Test
NI	-	Nuclear Island
NLD	-	Non-Linear Disengagement
NRC	-	U.S. Nuclear Regulatory Commission
NSSS	-	Nuclear Steam Supply System. Synonymous with Nuclear Island.
OBE	-	Operating Basis Earthquake
OCC	-	Operator Control Centers
OHX	-	Overflow Heat Exchanger
O&M	-	Operations and Maintenance
OPDD	-	Overall Plant Design Description
ORT	-	Operational Reliability Testing
P/F	-	Power to Flow
PACC	-	Protected Air Cooled Condenser
PAS	-	Plant Annunciator System

ABBREVIATIONS AND ACRONYMS (Continued)

PCA	-	Primary Control Assembly
PCRD	-	Primary Control Rod Driveline
PCRDM	-	Primary Control Rod Drive Mechanism
PCRS	-	Primary Control Rod System
PCS	-	Plant Control System
PDH&DS	-	Plant Data Handling and Display System
PEEHCS	-	Piping and Equipment Electrical Heating and Control System
PEL	-	Proportional Elastic Limit
PEOC	-	Plant Expected Operating Conditioning
PHENIX	-	Prototype Fast Reactor (250 MWe) located at Marcoule, France
PHTS	-	Primary Heat Transport System
P&ID	-	Piping and Instrumentation Drawing
PIT	-	PCA Irradiation Test
PMC	-	Project Management Corporation
PNC	-	Power Reactor and Nuclear Fuel Development Corporation (Japan)
PO	-	Project Office, CRBRP
PPS	-	Plant Protection System
PSAR	-	Preliminary Safety Analysis Report
PSB	-	Plant Service Building
PSP	-	Primary Sodium Pump
PSS	-	Primary Shutdown System
PTI	-	Plugging Temperature Indicator
PWHT	-	Post Weld Heat Treatment
PWR	-	Pressurized Water Reactor
PWST	-	Protected Water Storage Tank
RAPS	-	Radioactive Argon Processing System
RBA	-	Radial Blanket Assembly
RCB	-	Reactor Containment Building
RCC	-	Refueling Communications Center
RDSD	-	Reliability Design Support Document
RDT	-	Division of Reactor Development and Technology, U.S. AEC
RFP	-	Request for Proposal

ABBREVIATIONS AND ACRONYMS (Continued)

RFTP	-	Reactor Fuel Transfer Port
RHTIS	-	Reactor Heat Transport Instrumentation System
RM	-	Reactor Manufacturer
RPST	-	Reaction Products Separator Tank
RRD	-	Division of Reactor Research and Development, ERDA
RRPI	-	Relative Rod Position Indicator
RRS	-	Reactor Refueling System
RRSA	-	Removable Radial Shield Assembly
RSB	-	Reactor Service Building
RSS	-	Reactor Shutdown System
RT	-	Room Temperature
RV	-	Reactor Vessel
SAW	-	Submerged Arc Welding
SCA	-	Secondary Control Assembly
SCC	-	Stress Corrosion Cracking
SCRD	-	Secondary Control Rod Driveline
SCRDM	-	Secondary Control Rod Drive Mechanism
SCTI	-	Sodium Component Test Installation (located at Santa Susanna, CA)
SDD	-	System Design Description
SDOF	-	Single Degree of Freedom, Dynamic Response Analysis
SELT	-	Inelastic Cladding Strain Equivalent Limiting Temperature
SEM	-	Scanning Electron Microscopy
SFSC	-	Spent Fuel Shipping Cask
SFPS	-	Sodium Fire Protection System
SGAHRs	-	Steam Generator Auxiliary Heat Removal System
SGB	-	Steam Generator Building
SGM	-	Steam Generator Module
SGS	-	Steam Generator System
SMBDB	-	Structural Margin Beyond the Design Basis
SPTF	-	Sodium Pump Test Facility (located at ETEC)
SRI	-	Southwest Research Institute

ABBREVIATIONS AND ACRONYMS (Continued)

SRP	-	Standard Review Plan (NRC)
SRP	-	Small Rotating Plug (32)
SS	-	Stainless Steel
SSE	-	Safe Shutdown Earthquake
SSP	-	Sodium Sampling Package
SSS	-	Secondary Shutdown System (99)
SSS	-	Shield and Seismic Support (31)
SSST	-	Site Suitability Source Term
SUPER PHENIX	-	Commercial Fast Reactor (1,000 MWe), located at Crey's-Malville, France
SWR	-	Sodium Water Reaction
SWRPRS	-	Sodium Water Reaction Pressure Relief Subsystem
T/C	-	Thermocouple
TDAFWP	-	Turbine Drive Auxiliary Feedwater Pump
TELT	-	Worst-time-in-life Transient Equivalent Limiting Temperature
T/H	-	Thermal-Hydraulic
THDV	-	Thermal Hydraulic Design Value
TLP	-	Top Load Plane
TMBDB	-	Thermal Margin Beyond the Design Basis
TMI 2	-	Three Mile Island, Unit 2 Nuclear Power Plant (located near Harrisburg, PA)
TOP	-	Transient Overpower
TREAT	-	Transient Reactor Test Facility (located in Idaho)
TSTF	-	Thermal Striping Test Facility (located at WAESD)
T/TS	-	Tube-to-Tubesheet
TWJ	-	Transition Weld Joint
UCFS	-	Upper Core Former Structure
UIS	-	Upper Internal Structure
UISJM	-	UIS Jacking Mechanism
UPS	-	Uninterruptible Power System
UT	-	Ultrasonic Testing
VBM	-	Vertical Boring Mill

ABBREVIATIONS AND ACRONYMS (Continued)

WAESD	-	Westinghouse Advanced Energy Systems Division
WARD	-	Westinghouse Advanced Reactors Division. Renamed WAESD
WSS	-	Water/Steam Subsystem
ZPPR	-	Zero Power Plutonium Reactor (located in Idaho)

4.0 REACTOR CLOSURE HEAD STRUCTURAL MARGIN BEYOND DESIGN BASIS (SMBDB) DESIGN

4.1 Summary of Functions and Description

4.1.1 Functions

The Clinch River Breeder Reactor Plant (CRBRP) closure head assembly functions as a pressure boundary structure, an equipment support, a refueling machine element, a radiation shield, a thermal insulator and a gas entrainment control device. [1] Specific functional requirements are summarized as follows:

- A. Form an upper pressure boundary to the reactor enclosure in conformance with the ASME rules for "NPT" components.
- B. Provide access for the passage of replaceable core components, radial shield assemblies, control rod systems components, maintenance tooling, instrumentation and replaceable reactor internals components.
- C. Make provision for the in-vessel refueling machine to be positioned directly over any core lattice position so as to effect a direct push/pull insertion/extraction of any core component or removable radial shield assembly.
- D. In conjunction with the in-vessel refueling machine, make provision for the transport of core components and removable radial shield assemblies between their assigned operating locations and a transfer station located outside the core barrel.
- E. In conjunction with the ex-vessel refueling machine, make provisions for the transfer of components through the pressure boundary to and from the transfer station.
- F. Attenuate nuclear radiation emanating from within the vessel.

- G. Absorb, in conjunction with the head riser and reactor vessel support system, the impact energy from any accident occurrence without fragmentation of the basic head structure.
- H. Provide thermal insulation of the structural and process regions of the head from the core coolant.
- I. Support the control rod drives and upper internals structure with their actuators, and other necessary equipment mounted thereon.
- J. Provide structure(s) for prevention of gas entrainment within the reactor coolant.
- K. Provide the dip seals between rotating plugs to prevent sodium vapor and cover gas circulation in the annulus between each plug.
- L. Provide for retention of a minimum gas volume to prevent gas pressures upon head mounted equipment seals from exceeding allowable design values under imposed accident conditions.
- M. Provide penetrations, nozzles, and attachment surfaces for nozzle extensions and control rod drive mechanisms (CRDM).
- N. Provide keys and keyways for the CRDM lateral seismic shock supports.
- O. Absorb without damage, loads equal to the weight of the reactor core and upper internals, with the requisite loading factors in seismic or loss of core holddown events.
- P. Support refueling equipment.
- Q. Provide threaded holes whereby external reactor equipment can be securely attached to the head.
- R. Provide penetrations and sealing surfaces for interior vessel instrumentation and surveillance specimen insertion and retrieval.

4.1.2 Closure Head SMBDB Loads

SMBDB functional requirements for the CRBRP closure head are defined in a qualitative manner in item (G) above. The postulated event governing the SMBDB design of the closure head is the hypothetical core disruptive accident (HCDA). Derivation of the closure head SMBDB loads produced by the HCDA event was done in a two phase analysis program. In the first phase the SAS-VENUS computer program was used to calculate the core work potential following a core HCDA. The work potential was expressed in terms of a vapor bubble pressure-volume relationship. In the second phase the REXCO-HEP hydrodynamic/structural computer program was used to calculate the kinetic energy in the slug propelled towards the closure head and the pressure-time history on the underside of the closure head produced by the slug impact.

A detailed report of the analyses of Phases 1 and 2 is provided in References 2 and 3 respectively. A brief summary of the principal assumptions and the resulting closure head SMBDB loads is provided below.

A. Phase 1: Establishing Pressure-Volume Relationship for SMBDB Evaluations

The process for establishing the SMBDB pressure-volume relationship considered both the energetic potential of the core and the capability of the reactor coolant boundary, the objective being to achieve a large structural margin in a practical design of the plant.

The SAS-VENUS calculations for the homogeneous core indicated that when pessimistic assumptions are used, the HCDA could produce average core fuel temperatures in the range of 4300°K. Average core fuel temperatures in the range of 5000°K could be generated only if arbitrary, very low probability (conservative) assumptions were used for the VENUS-II calculations for the homogeneous core.

Several VENUS-II cases were run with nonuniform temperature distributions to produce average core fuel temperatures in the range of 4300°K to 5000°K (based on homogeneous core considerations).

These cases were not related specifically to either the initiating phase energetics or disruption phase recriticality energetics, since the objective was to establish loads that would provide margin for a range of scenarios.

Different temperature distributions were also developed for the same average temperature, and pressure-volume curves were developed by isentropic expansion of the two-phase fuel mixture for all the temperature distributions. The temperature distribution can be as significant to the damage potential as the average temperature, since the high temperature (and corresponding high pressure) regions contribute disproportionately to the damage potential. All of the pressure-volume curves considered provided margin relative to the work potential of best-estimate cases* and were generally characteristic of the very pessimistic cases*.

The pressure-volume relationships were used in the REXCO-HEP code to develop reactor coolant boundary loads. These loads were then used to assess, in a preliminary way, the capability of the design to withstand the loads. Consideration was given to the options available for improving the structural capability and the possibility of more detailed evaluations identifying additional limiting areas.

The temperature distribution chosen has a 4800°K average temperature, and a peak temperature and pressure of 6030°K and 273 bars, respectively. Figure 4-1 shows the temperature distribution on a percent fuel mass basis and Figure 4-2 shows the P-V relationship that is developed by isentropic expansion of fuel vapor from the initial state. Based on an initial two-phase fuel volume of 2.56 m³, the integral of the P-V curve to the approximate point of sodium slug impact with the reactor head is 101 MJ. This pressure-volume curve is judged to provide substantial margin relative to the potential of the core to apply mechanical loads to

* See Appendix C.

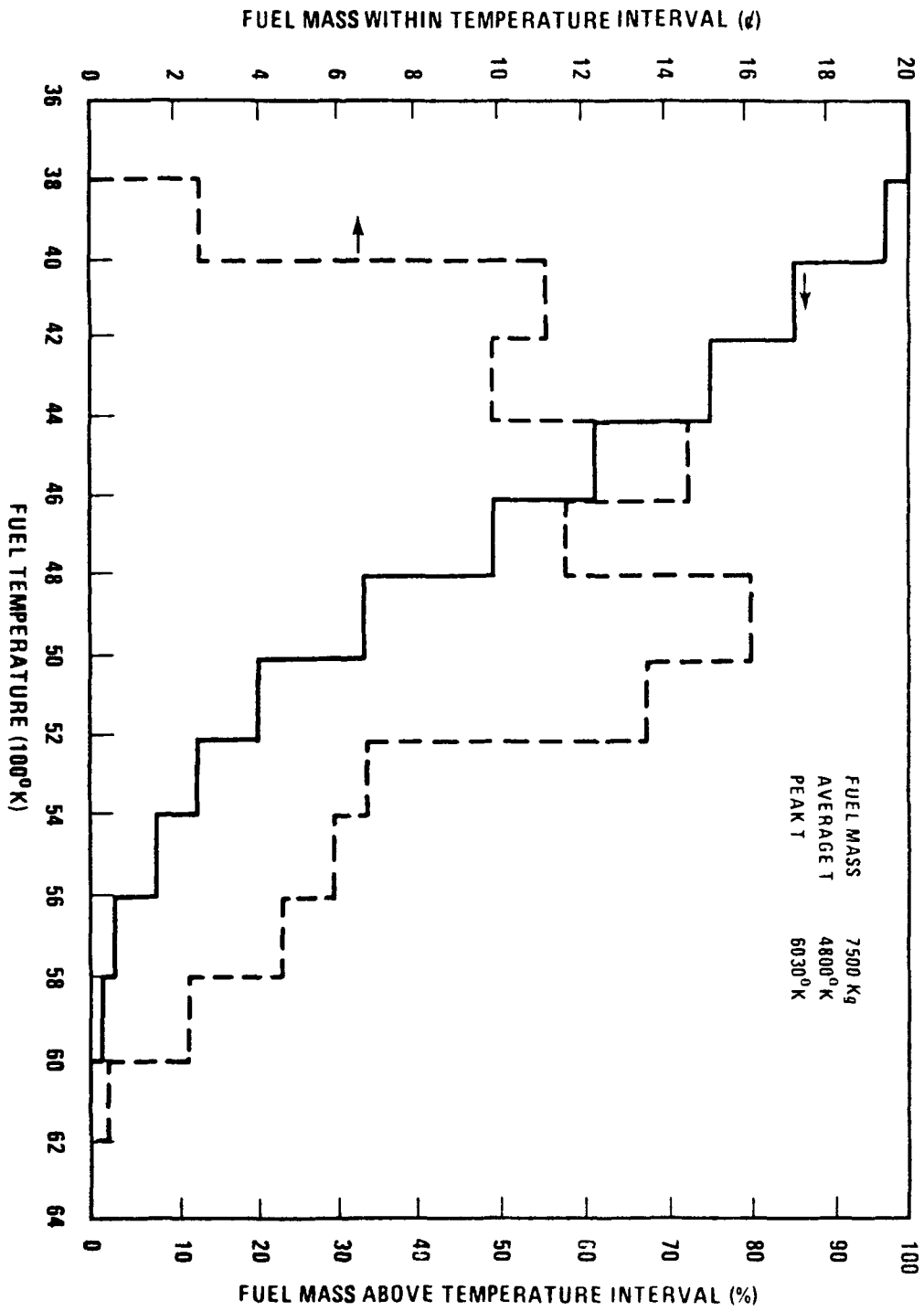


Figure 4-1 Thermal Source for Structural Margin Beyond the Design Base

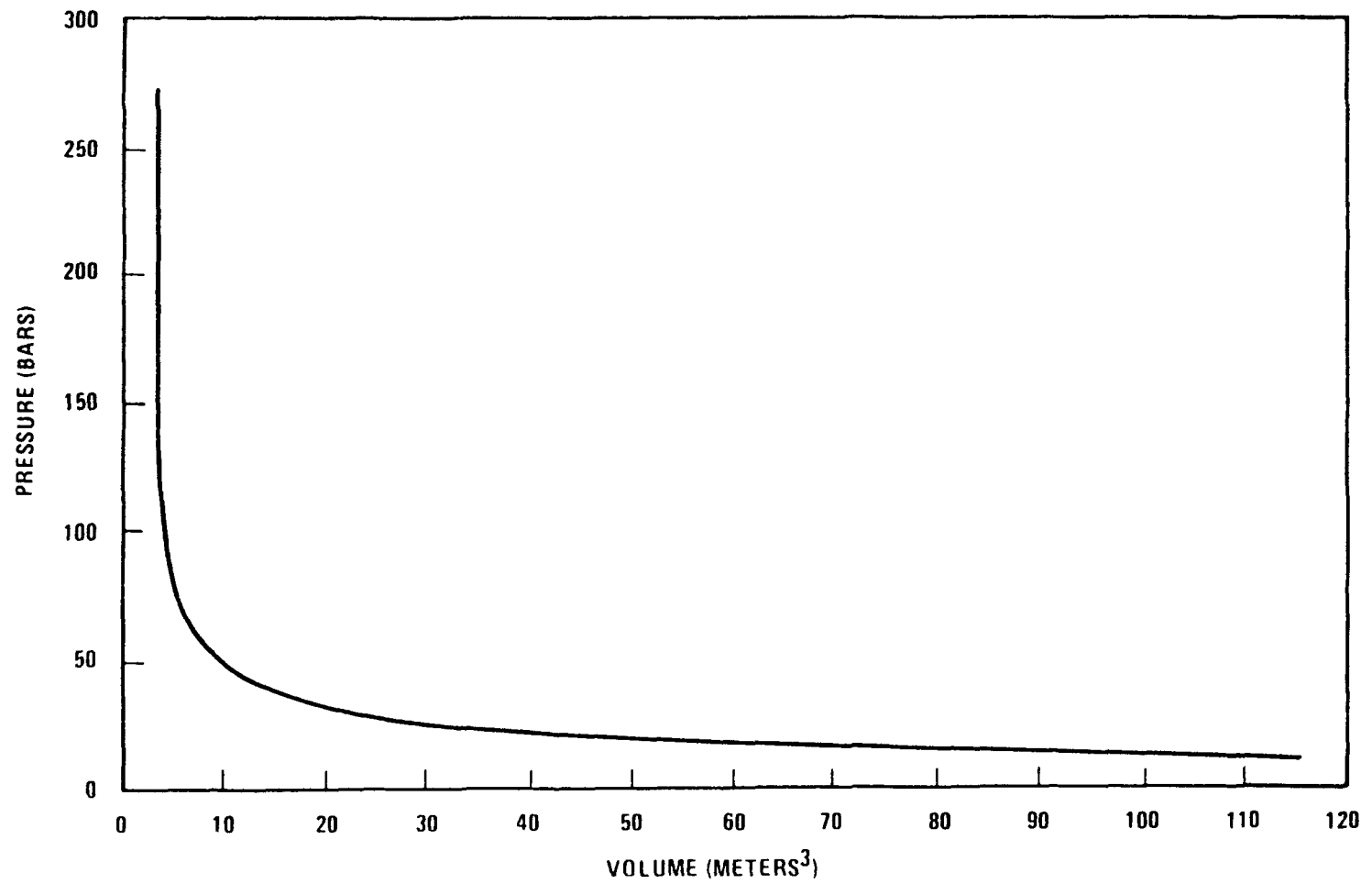


Figure 4-2 Pressure Volume Relationship for SMBDB Loadings

the reactor coolant boundary as the result of an HCDA. Expanding the P-V curve up to a volume corresponding to a pressure of 1 atmosphere yields a total energy release of 661 MJ. This latter value is not used directly in the SMBDB analysis of the closure head, but it does provide a convenient quantitative definition of the HCDA event.

B. Phase 2: Establishing the Closure Head SMBDB Loads

SMBDB loadings are based on two phenomena connected with the HCDA, namely (1) short-term (up to a few hundred milliseconds) loadings resulting from core hydrodynamic disassembly and (2) the longer term (up to 1000 seconds) loadings due to the dynamics of the released vapors and gases (i.e., fuel vapor, sodium vapor, fission product gases).

The sequence of events occurring during the postulated hydrodynamic disassembly is as follows: As the vapor expands, it would first load the structures surrounding the core (blankets, shields, upper pin structures, etc.) As it expands further, it would displace the core support structure downward, load the core barrel and vessel wall radially, and accelerate upward the slug of sodium coolant above the core. The sodium slug would impact the reactor vessel head and upper vessel wall. Some of the available energy would be dissipated by plastically straining portions of the vessel wall, core barrel, and other structures. These conditions are simulated using the structural/hydrodynamic code REXCO-HEP.

Results obtained using the REXCO-HEP code in conjunction with the pressure volume curve of Figure 4-2 are summarized in Table 4-1 and Figure 4-3. Table 4-1 shows the partitioning of energy at the instant of slug impact. Note that the slug energy content at the instant of impact with the closure head is 75.1 MW-sec. Results from the REXCO-HEP analysis of the force produced on the underside of the head by the slug impact are shown as a function of time in Figure 4-3. Both the slug energy content and the calculated force-time history were used in the development of the closure head design.

TABLE 4-1
PARTITION OF ENERGY FOR STRUCTURAL MARGIN
BEYOND THE DESIGN BASE

	<u>At Slug Impact</u>
Time, msec	69.2
Upward Kinetic Energy, MW-sec	75.1
Radial Kinetic Energy, MW-sec	2.9
Core Barrel Strain Energy, MW-sec	16.9
Vessel Strain Energy, MW-sec	4.9
Coolant Internal Energy, MW-sec	1.4
Vessel Head Energy, MW-sec	0
Total Work Energy, MW-sec	101.4

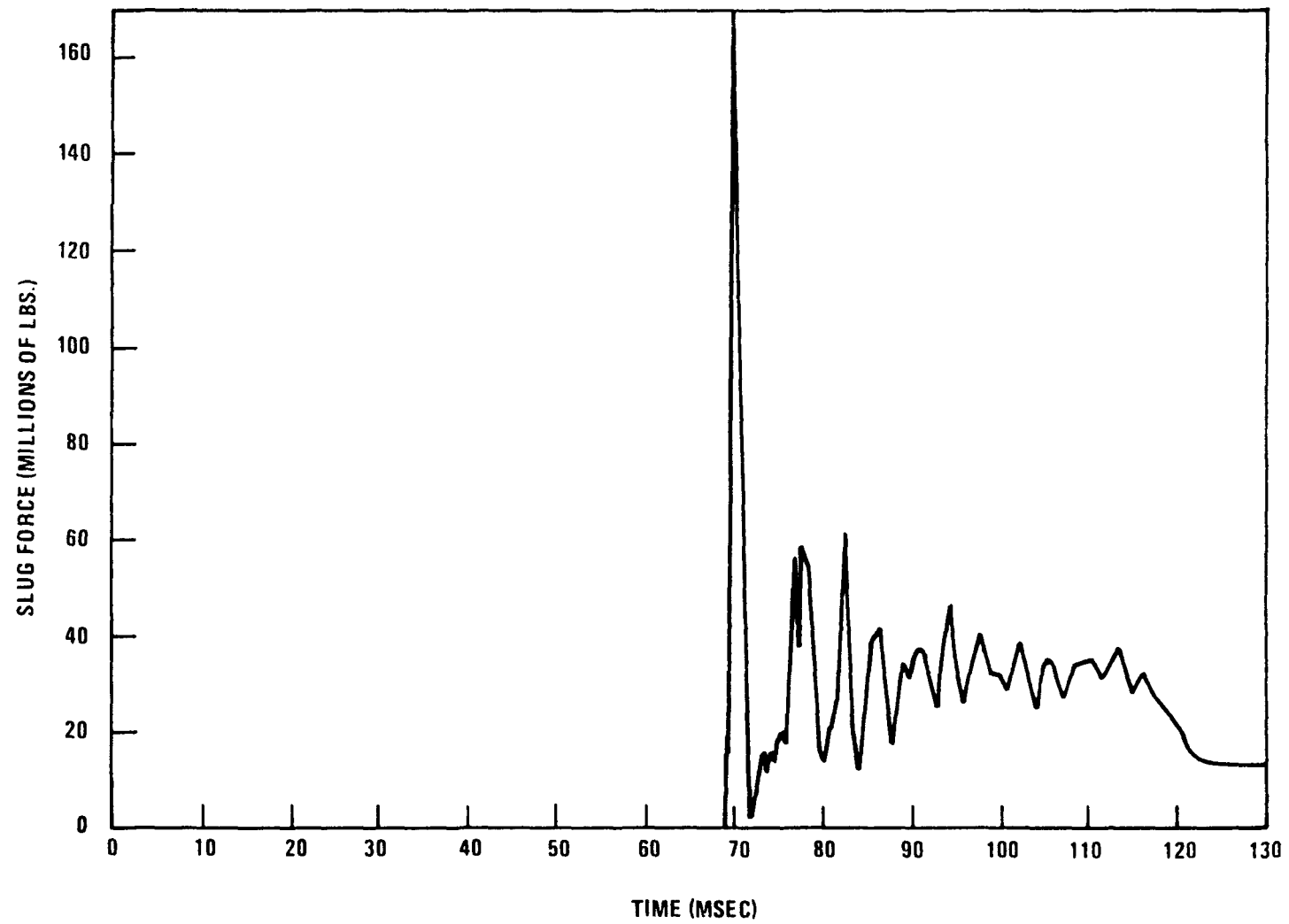


Figure 4-3 Slug Force On Under Head Shielding of Reactor Vessel Closure Head

Since the SMBDB loading represented a condition beyond the limits considered by existing design codes of practice, special structural criteria had to be developed for the SMBDB loading case. These criteria are summarized in Reference 3. They limit both stresses and strains so as to protect the structure of both the closure head and head mounted components from both tensile plastic instability and local ductile rupture failure modes.

4.1.3 Design Description

The closure head assembly forms the upper closure of the reactor vessel, supports the associated reactor equipment, and contains various penetrations for control rods, in-vessel instrumentation, upper internals, refueling equipment, and dip-seal maintenance and filling equipment (Figures 4-4 and 4-5). The reactor vessel and closure head are illustrated in Figure 4-6. It includes three independently rotatable plugs interconnected by plug risers. A bull gear and bearing atop each riser permit plug rotation by a plug drive and control system. One pair of plug risers supports the closure head assembly from the reactor vessel flange.

The structural portion of the closure head consists of three rotating plugs, made from SA-508 Class-2 low-alloy steel forgings, called the Large Rotating Plug (LRP), Intermediate Rotating Plug (IRP) and Small Rotating Plug (SRP). The LRP fits into and is supported by the Reactor Vessel Flange; the IRP fits into an eccentric hole in the LRP; and the SRP fits into an eccentric hole in the IRP. The overall dimensions of the three plugs are as follows:

- A. The LRP has a nominal outside diameter of 257.38 inches and a nominal thickness of 22.00 inches. Its centerline coincides with the reactor centerline.
- B. The IRP has a nominal outside diameter of 175.50 inches and a nominal thickness of 22.00 inches. Its centerline is offset 27.20 inches from that of the LRP.

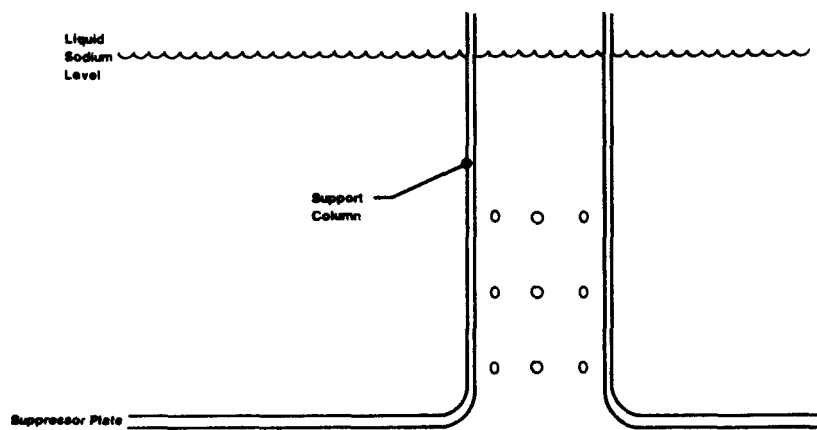
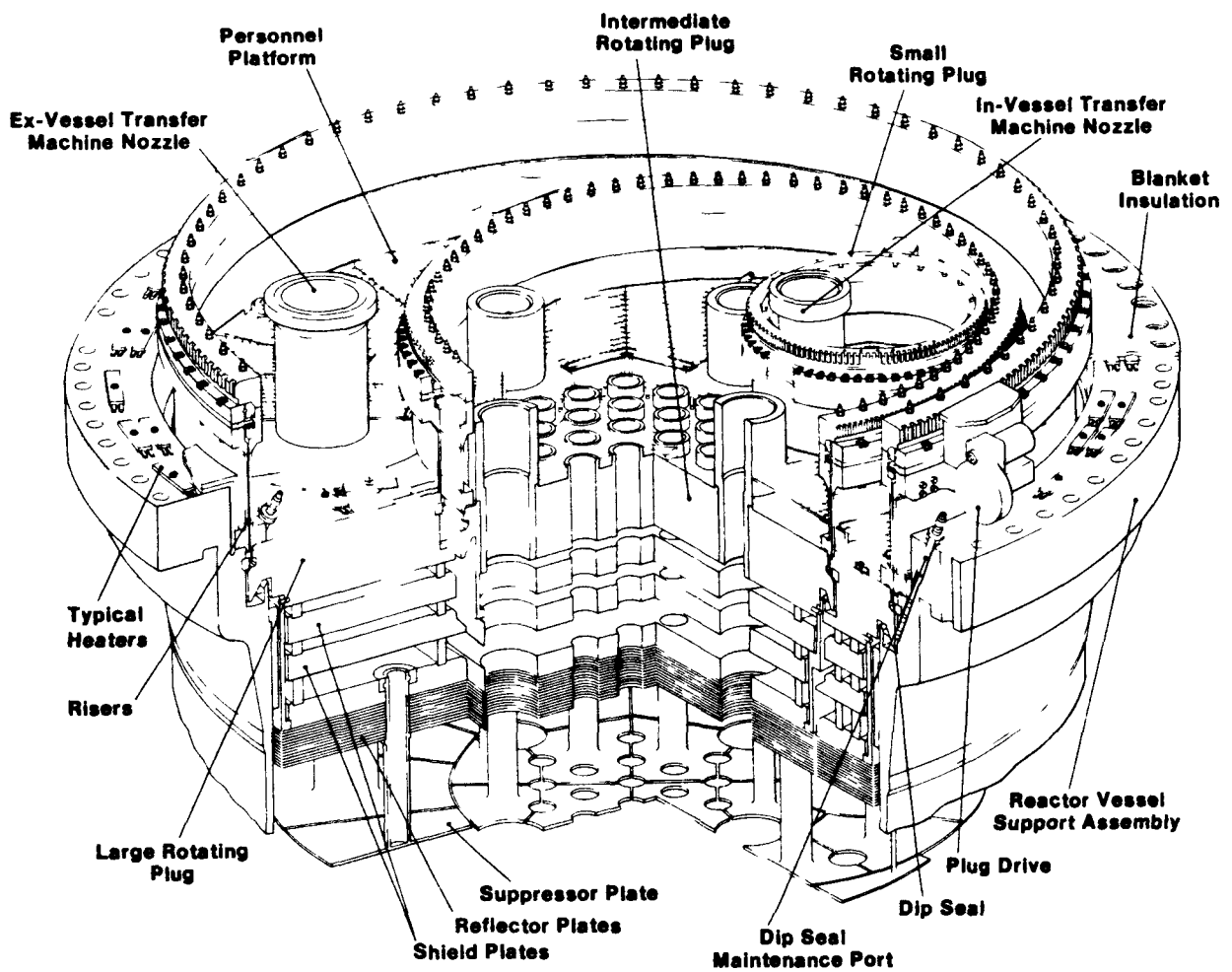


Figure 4-4 Reactor Closure Head Assembly

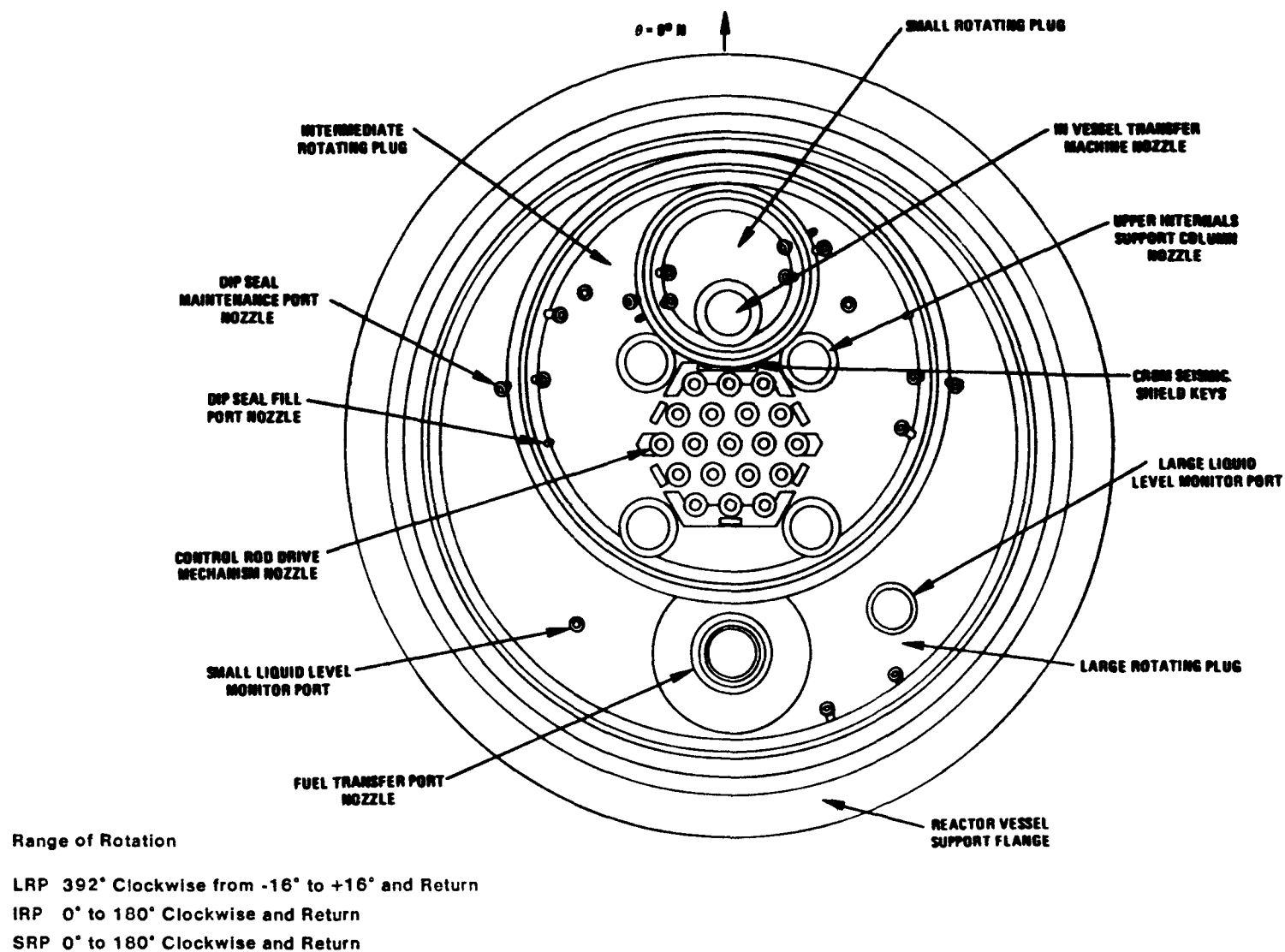


Figure 4-5 Closure Head Assembly Plan View

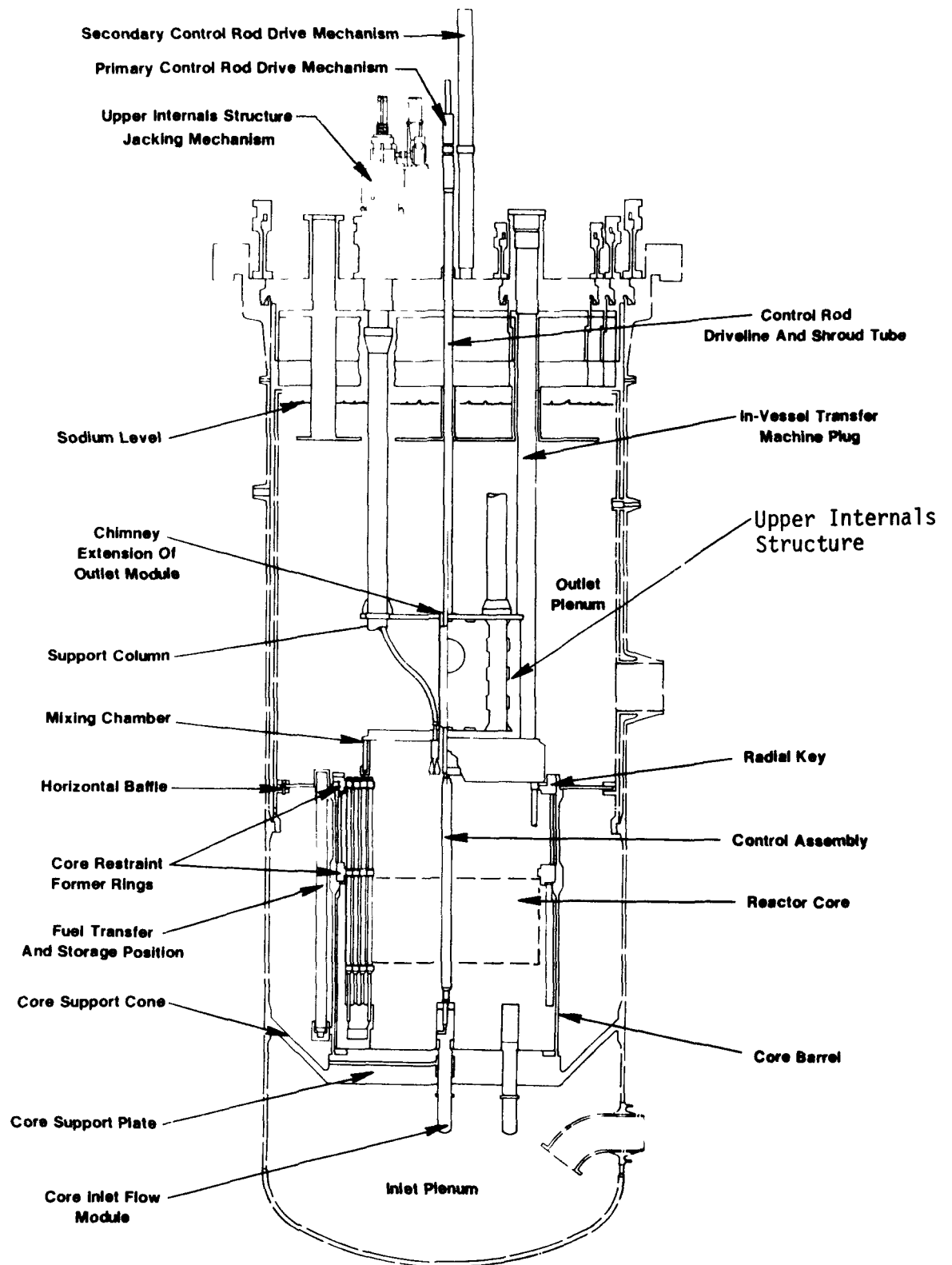


Figure 4-6

Reactor Schematic

- C. The SRP has a nominal outside diameter of 67.94 inches and a nominal thickness of 22.00 inches. Its centerline is offset 42.85 inches from that of the IRP.

The arrangement of the three plugs in their operating configuration is illustrated in Figure 4-5.

Attached to and supported by the three plugs are the underhead shield plates, reflector plates and suppressor plates. The lower shield plate, 10 inches thick, rests on an outer cylindrical support skirt which is in turn pinned to the underside of the respective rotating plug. The intermediate shield plates, 10 inches thick, and upper shield plates, 9 inches thick, rest on spacer rings located near the outer and inner edges of the plates. The reflector and suppressor plates are suspended from columns which are bolted to the lower shield plates. This arrangement is illustrated in Figure 4-7.

Major head mounted components are as follows:

- A. The In-Vessel Transfer Machine (IVTM) Nozzle is attached to the SRP at the location indicated in Figure 4-7.
- B. The Ex-Vessel Transfer Machine (EVTM) Nozzle is attached to the LRP at the location indicated in Figure 4-7.
- C. The 9 Primary and 6 Secondary Control Rod Drive Mechanisms (CRDM) and Control Rod Drivelines (CRD) are attached to the IRP at the locations shown in Figure 4-7.
- D. The Shield and Seismic Support is attached to the IRP and encloses the Primary and Secondary CRDMs.
- E. The Upper Internals Structure (UIS) and UIS Jacking Mechanisms are supported by the IRP at the locations indicated in Figure 4-5. The UIS is suspended below the Closure Head as shown in Figure 4-6.

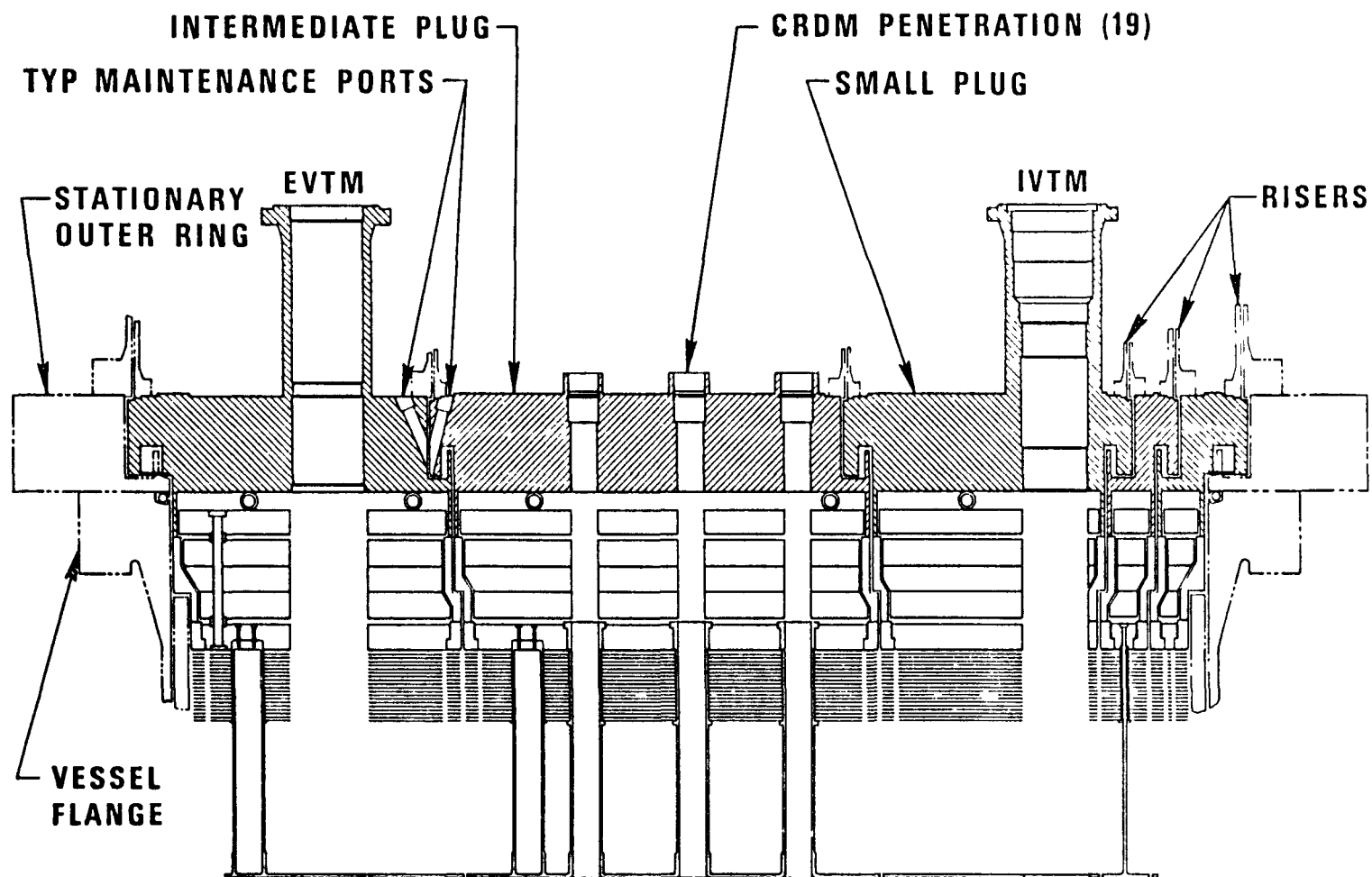


Figure 4-7 CRBRP Closure Head Elevation

The three plugs are suspended from their bearings by a system of inner and outer risers. This plug-riser arrangement is shown in Figure 4-8. Typically, the inner riser is a thin-walled cylindrical shell bolted to the periphery of the plug. The inner riser engages the riser bearing, which is supported by the outer riser. The outer riser is bolted to the periphery of the corresponding cutout. In this manner, the SRP is supported by the IRP, the IRP by the LRP, and the LRP by the Reactor Vessel Flange. The three plugs are independently rotatable, being driven through gears attached to the inner risers. Rotation of the three plugs allows handling machinery attached to the SRP to be positioned over any element in the core for removal and replacement during refueling.

Under normal conditions of operation and refueling, the plugs are held in position by the risers and by their own weight. In the event that abnormally high upward loads are applied to the plugs (as during an HCDA), plug retention is accomplished by the LRP, IRP and SRP Margin Rings. These are segmented rings seated in grooves machined into the peripheries of the Reactor Vessel Flange, LRP and IRP cutouts and held in place by continuous filler rings. The locations of the margin rings and filler rings for the LRP, IRP, SRP are illustrated in Figure 4-8. The cross-sections and dimensions of the margin rings and filler rings are shown in Figure 4-9.

During 1/20 scale model testing of the Closure Head, it was discovered that the Margin Shear Rings disengaged due to an unanticipated kinematic interaction between the deflected head plugs. The kinematic failure mode can be eliminated by local machining of portions of the head plugs and shield plates. With this modification the head system was shown to exceed the design requirements.

4.1.3.1 Margin Loading

The closure head assembly was analyzed for SMBDB in accordance with structural criteria delineated in WARD-D-0156 [3] (and WARD-D-0178, CRBRP-ARD-0218 [4,5]), for loadings beyond the design base. Loading conditions analyzed were specified in the equipment specification and include the WARD-D-0156 curves applicable to the closure head. Loads were postulated to occur during normal

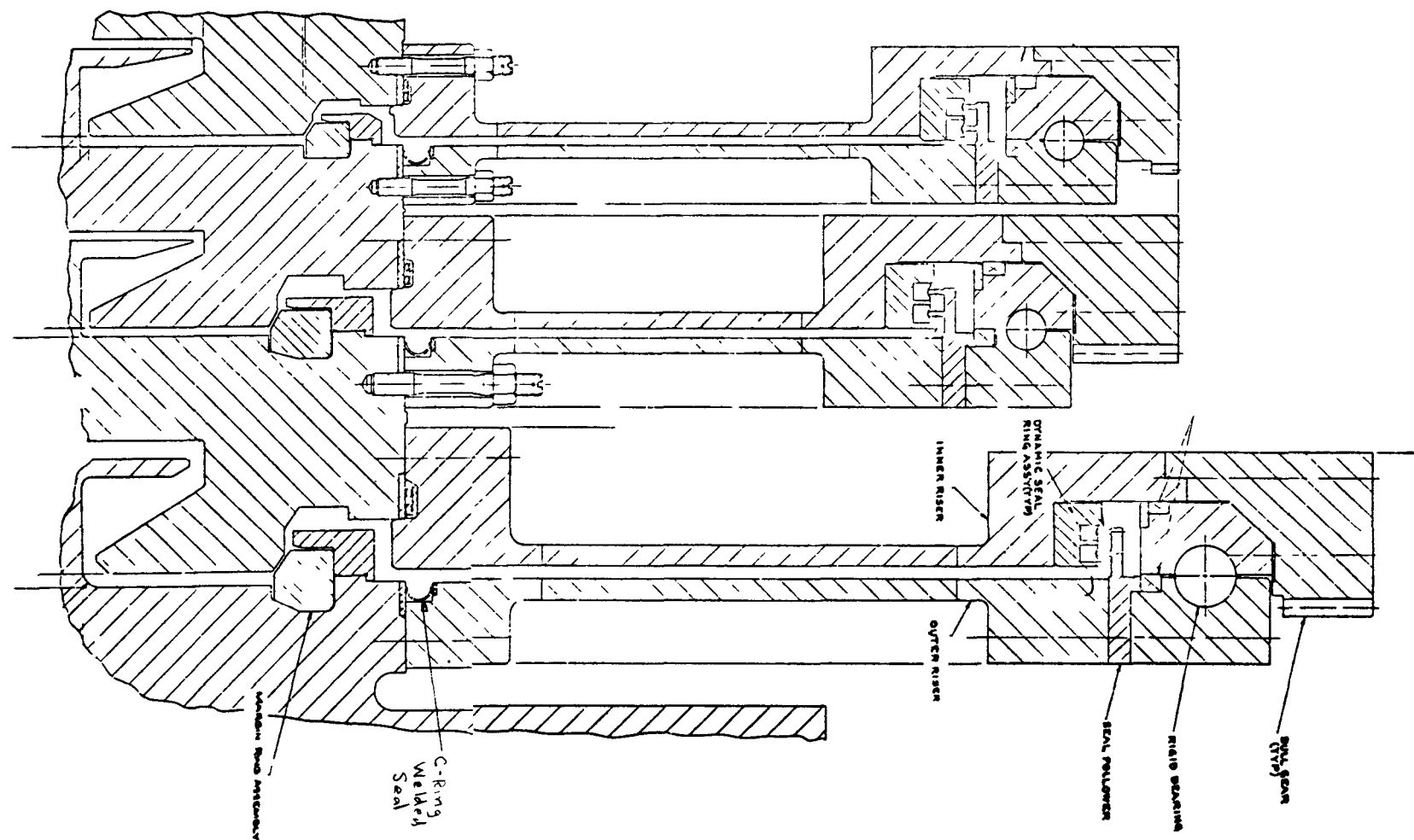
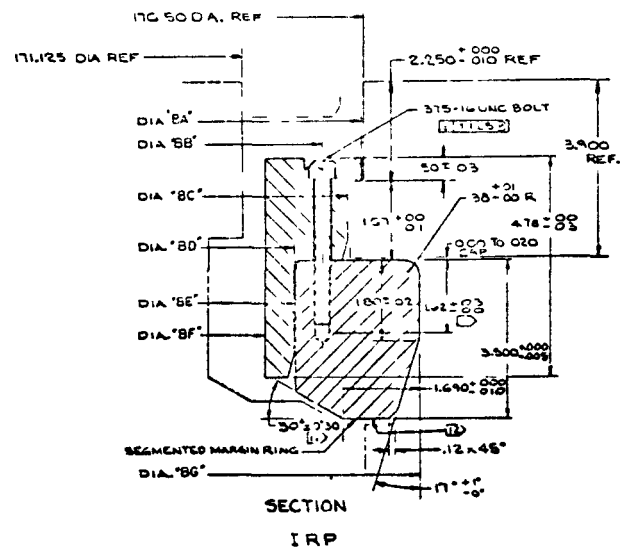
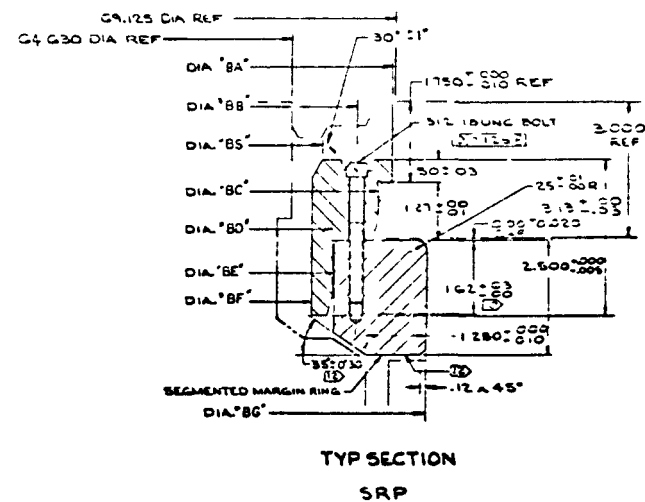


Figure 4-8
CRBRP Riser Layout

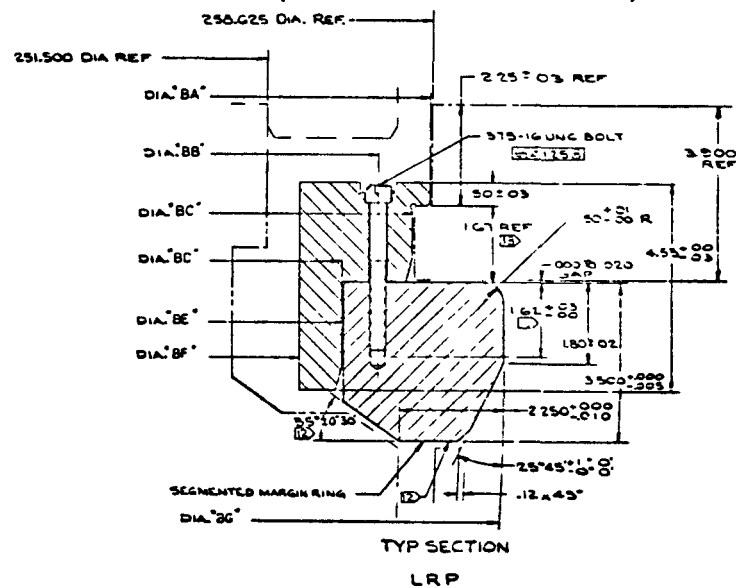


(Machined Groove in LRP)

	MARGIN RING & FILLER RING DIMENSIONS		
	SRP	IRP	LRP
BA	69.035 ^{+0.000} _{-0.030}	176.410 ^{+0.000} _{-0.010}	250.535 ^{+0.000} _{-0.030}
BB	67.250 B5C	174.500 B5C	256.125 B5C
BC	68.331 ^{+0.000} _{-0.030}	175.756 ^{+0.000} _{-0.010}	257.551 ^{+0.000} _{-0.030}
BD	64.470 ^{+0.000} _{-0.030}	173.345 ^{+0.000} _{-0.010}	254.545 ^{+0.000} _{-0.030}
BE	66.500 ^{+0.000} _{-0.030}	173.375 ^{+0.000} _{-0.010}	254.575 ^{+0.000} _{-0.030}
BF	65.025 ^{+0.000} _{-0.030}	174.175 ^{+0.000} _{-0.010}	252.600 ^{+0.000} _{-0.030}
BG	70.500 ^{+0.000} _{-0.010}	178.875 ^{+0.000} _{-0.010}	261.875 ^{+0.000} _{-0.030}
BH	25.148 ^{+0.000} _{-0.010}	24.354 ^{+0.000} _{-0.010}	32.910 ^{+0.000} _{-0.010}
BI	40 ^{+0.00} _{-0.00} B5C	160 ^{+0.00} _{-0.00} B5C	45 ^{+0.00} _{-0.00} B5C
BK	16 ^{+0.00} _{-0.00} B5C	7 ^{+0.00} _{-0.00} B5C	5 ^{+0.00} _{-0.00} B5C
BL	59 ^{+0.00} _{-0.00} B5C	26 ^{+0.00} _{-0.00} B5C	20 ^{+0.00} _{-0.00} B5C
BM	—	46 ^{+0.00} _{-0.00} B5C	34 ^{+0.00} _{-0.00} B5C
BN	14 ^{+0.00} _{-0.00} B5C	6 ^{+0.00} _{-0.00} B5C	4 ^{+0.00} _{-0.00} B5C
BP	7.50 MIN	2.24 MIN	4.40 MIN
BH	31.700 ^{+0.000} _{-0.010}	100.063 ^{+0.000} _{-0.010}	121.111 ^{+0.000} _{-0.010}
BS	65.412 ^{+0.000} _{-0.010}	—	—



(Machined Groove in LRP)



BEST AVAILABLE COPY

Figure 4-9

Margin Ring Sections

(Machine Groove in Reactor Vessel Flange)

full power operation independent of any other event. (See Section 4.1.2). To demonstrate the capability of the closure head to accommodate the SMBDB loading, scale model testing was used in addition to analysis.

4.2 Evolution of Design

4.2.1 Conceptual Design

The conceptual design review of the closure head was held in late November 1975. Prior to the conceptual design review, the closure head radiological shielding was redesigned. This shielding now consists of three SA-516 Grade-60 low-alloy steel shield plates instead of the previous five plates composed of SA-533 Grade B low-alloy steel, Type 304 stainless steel, and Alloy 600. The material for the reflector plates was fixed as 13-gauge Type 304 stainless steel sheet. A design employing mechanical attachment of the suppressor plates to their support columns was studied to reduce the severe thermal stresses in that area. All thermal, seismic, and margin loads applied to the closure head by attached equipment were also determined.

The request for proposal (RFP) for the design and fabrication of the closure head was issued in late October 1975. At the same time, another RFP was issued for a study contract that would culminate in a responsive proposal to the former RFP. Two study contracts were awarded, and two responsive proposals for the design and fabrication of the closure head were received. The study contracts resulted in two independent reviews and analyses of the current closure head design. Details of the shielding support system and nozzle attachments were sized.

4.2.2 Preliminary Design

The preliminary design review of the closure head was held on June 30, 1976. Interim fabrication release documentation was prepared to permit initial fabrication of the plugs and shield plates. Chicago Bridge and Iron Company (CBI) was authorized to begin fabrication on September 15, 1976 and fabrication was started on September 22, 1976.

A significant achievement during the preliminary design phase was the baselining of the suppressor plate design. The design utilizes a rounded plate-to-column transition formed integrally with the plate by a "flueing" operation. (Figure 4-4). The feasibility of the design was demonstrated by production of a prototype plate by Taylor Forge. The plate was formed and solution annealed by Taylor Forge. Final flattening was performed by CBI at Birmingham, Ala.

4.2.3 Final Design

Design and analysis for accommodation of the structural margin beyond the design base (SMBDB) load event resulted in the design of a system that will limit head motion and prohibit sodium expulsion from the reactor vessel during the event. A segmented shear ring assembly is located in the annulus between each rotating plug and between the large plug and the reactor vessel. This margin shear ring assembly functions only during the margin event, such as an HCDA, and limits upward motion of the closure head. The capability of the system to accommodate HCDA loads would be verified by model testing in addition to analysis.

Margin seals mounted between the bull gear and bearing assemblies and the tops of the risers trap any sodium that may escape past the shear ring and inflatable seals. The margin seals are made of Viton and have a minimum life expectancy of 10 years.

The final design review for the closure head was completed in July 1977, and a fabrication release was made in October 1977. Fabrication of all major components of the closure head was completed by CBI. Assembly operations began in March 1980. Each closure head plug assembly was completed in its individual assembly stand. The bolted risers, shear ring assemblies, the seals, and the bull gear and bearings were previously delivered to CBI from other suppliers. These components were assembled to the closure head, and seal leak tests were performed.

The final design report consisted of a general dynamic analysis of the closure head assembly, the ASME Boiler and Pressure Vessel Code stress report for the pressure boundary, a distortion and alignment study, and a margin load analysis. Analyses were also done on non-pressure boundary components such as skirt and shield plate assemblies, suppressor plate assemblies, and reflector plate assemblies. The analyses considered operating, seismic, and margin loads where applicable.

4.2.4 Description of Development Program for HCDA Load Simulation

To demonstrate the capability of the closure head to accommodate the SMBDB loading, an analytical and a model testing approach were used. The complexity of the reactor vessel and internals design made the analytical hydrodynamic and structural modeling very difficult. Therefore, it was prudent to perform experiments to verify the structural integrity of the reactor vessel and the closure head. [6]

An appropriate model against which to assess the structural response of the CRBR reactor was a 1/20th scale model of CRBR, because the loading pressures, stress, strain, and slug velocity are the same in the 1/20th scale model as in the prototype. Scale factors, however, had to be applied to strain rate and accelerations [7]. The tests, based on such a configuration, were divided into five phases. Phase 0 was a preliminary phase to assess the potentially non-prototypic response of scale model materials under the predicted dynamic loads. In Phase I, the core pressure-volume relationship of the model was calibrated to correspond to the appropriate energy releases required in Phase II. In Phase II, seven tests to systematically evaluate the overall reactor structural response were performed. Tests SM-2 through SM-5 simulated the energetics associated with the 661MJ HCDA. SM-6 was not tested. Tests SM-1, SM-7 and SM-8 were static tests to assess the head failure mode and the influence of under head shielding representations.

Phase III employed static tests SM-9 and SM-10 to verify the capability of the head to accommodate a conservative estimate of the energy deposited by a 75 MJ sodium slug and provide the data required to refine the configuration of any required head modification.

Phase IV was planned to dynamically test a scale model with a simulated 75 MJ sodium slug. Test SM-11 was to be conducted similarly to SM-5, except for deletion of the upper internals structure, and was to confirm that the closure head structure would remain integral following the HCDA loading. Test SM-11 requirements were to be defined in detail only after Phase III testing had been completed and evaluated.

A. Phase 0: Materials Properties Studies

In testing scale models of Liquid Metal Fast Breeder Reactors (LMFBRs) which include rotating head plugs, special attention must be paid to the response of the head material located near the point of plug-to-plug impact. At this location, deviations from similitude in material response may exist as a result of strain rate and must be assessed. In addition, the strain rate properties of the Ni 200, from which the test vessel and internals were made, were evaluated. The tests used to evaluate these effects are described below.

In the test program the strain rates were 20 times those predicted in CRBRP. Accordingly, the strain rates in the 1/20 scale small, intermediate and large rotating plug shear ring notches were predicted (from ANSYS extensions of REXCO loads) to be 50, 39 and 24 in/in/sec respectively. However, because of effects not accounted for in the above analysis, e.g., tolerancing and asymmetric effects) it was felt that strain rates significantly higher than those could be experienced. Strain rate data on the scale model head material (SA-533B) was available up to 100 in/in/sec and it is in this range and beyond that the tests were performed. Assessment of the strain rate properties of the Ni 200 was also made. Since the components made from this material do not experience such high strain rates as the head, strain rates only up to 100 in/in/sec were assessed.

The SA-533B strain rate tensile tests was conducted at four strain rates near 1, 100, 400 and 700 in/in/sec. Three or four tests were performed at each strain rate to establish reproducibility. The Ni 200 strain rate tests included base material and welded joints. Both

the base and welded materials were tested at 1 and 100 in/in/sec, three or four tests being performed at each strain rate and material configuration.

B. Phase I. Source Calibration

To calibrate the pressure-volume source for the Phase II tests, a simple rigid vessel filled with water and containing the explosive source in a rigid core barrel was employed. The change in bubble pressure was determined directly from a pressure sensor in the core barrel. The bubble volume was determined by observing the vertical rise of a float resting on the surface of the water.

In tests SM-2 through SM-5 of the Phase II tests, a core pressure volume relationship corresponding to the 661MJ HCDA was used. A PETN explosive-glass microballoon combination was used as the energy source.

C. Phase II. Structural Response of CRBR Models

Phase II, consisting of seven tests (SM-6 was untested), provided additional information to assist in understanding the loading phenomena. The models used in these tests are described in Table 4-2 while the requirements used to plan tests SM-1 through SM-8 are recorded below to maintain historical continuity. Summary of test results and status are shown in Sections 4.2.5, 4.2.6, 4.2.7 and 4.2.8.

o SM-1 (Figure 4-10)

Test SM-1 was performed to determine the static force-deflection characteristics of the plug shear-ring combination. This test, performed by statically pressurizing the underside of a model of the head, was aimed at: (1) obtaining an understanding of the mode of shear ring and plug deformation to assist in defining failure modes and criteria and (2) obtaining an estimate of the structural capability of the head under static pressure loading.

TABLE 4-2
PHASE II SIMPLE MODEL EXPERIMENTAL TESTS (1/20th SCALE)

<u>Experiment Number</u>	<u>Head</u>	<u>Vessel</u>	<u>Core Barrel</u>	<u>Upper Internals</u>	<u>Energetics</u>
SM-1	S1	R1	None	None	Static Pressure
SM-2	P1	F2	F3	None	SMBDB
SM-3	F1	F2	F3	S3	SMBDB
SM-4	S2	F2	F3	S3	SMBDB
SM-5	S2	F2	F3	S3	SMBDB
SM-6	S2	F2	F3	S3	Not Completed
SM-7	S2	R1	None	None	Static Pressure
SM-8	S4	R1	None	None	Static Pressure

Where:

- S1 = Simplified head including plugs and shear rings.
- S2 = Simplified head including plugs, shear rings, under-head shielding.
- S3 = Simplified upper internal structures including lower plate and columns.
- S4 = Simplified head including plugs and shear rings, with prototypic under-head shielding.
- R1 = Rigid vessel (no plastic deformation).
- F1 = Flexible head (plus idealized representation of plug to plug gaps).
- F2 = Flexible vessel.
- F3 = Flexible core barrel.
- SMBDB = Structural Margin Beyond Design Base conditions considered for CRBR (661 MJ work energy on expansion to one atmosphere)

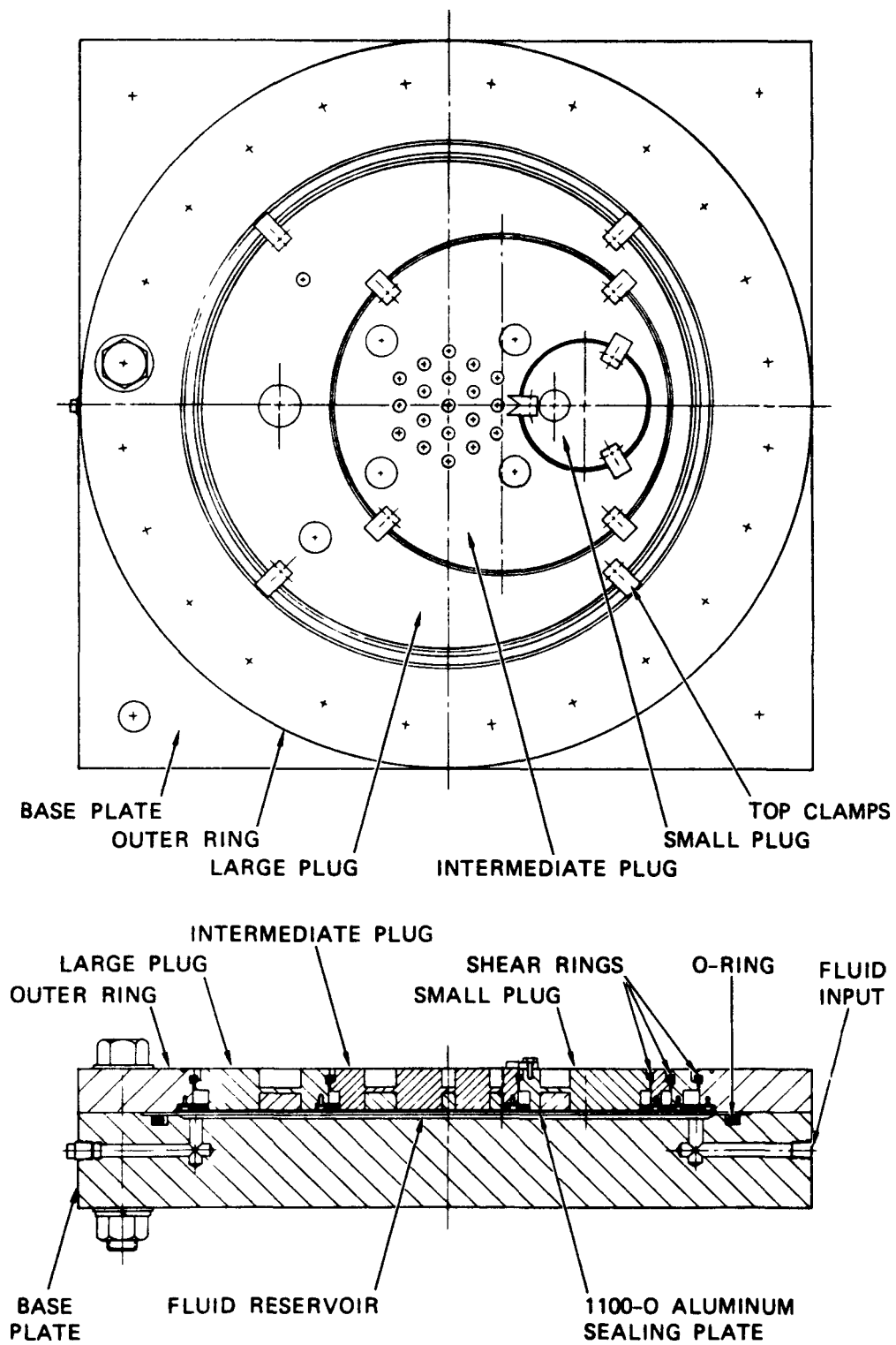


Figure 4-10 SCHEMATIC OF STATIC VESSEL HEAD APPARATUS, TEST SM 1

o SM-2 (Figure 4-11)

To assess the effect of such features as upper internals and head plugs in the overall reactor response, an idealized base case whose only features were a simple vessel, core barrel and head was tested. This case had a flexible vessel and core barrel for energy absorption. The head consisted of a uniform plate whose thickness corresponds to that of the CRBR head. To transmit the slug load from the head to the vessel flange the model included a scaled shear ring and associated head to vessel flange gap. This produced head impact loadings approximating those in the large rotating plug. The associated acceleration spike was used to calibrate head mounted accelerometers. In addition, the test was also useful for checking out analytical treatments of head systems which include gaps. Since this test most closely simulates the REXCO calculation used to determine the dynamic loads used as a basis for structural margins, it was used for comparison with this analysis. Finally, if the head experienced non-prototypic material response (e.g., local cracking) as a result of the geometric scaling effect on strain rates, this effect could be assessed before continuing with subsequent tests.

o SM-3 (Figure 4-11)

The effect of the upper internal structures was assessed in this test. The REXCO code is limited in its ability to model the potentially complex dynamic response of the upper internals structures and, in addition, is unable to simulate the orifice flow of the sodium between the core barrel and upper internals structures. Since the addition of upper internals structures represented the only change from test SM-2, the effects solely due to the upper internals structures could be assessed.

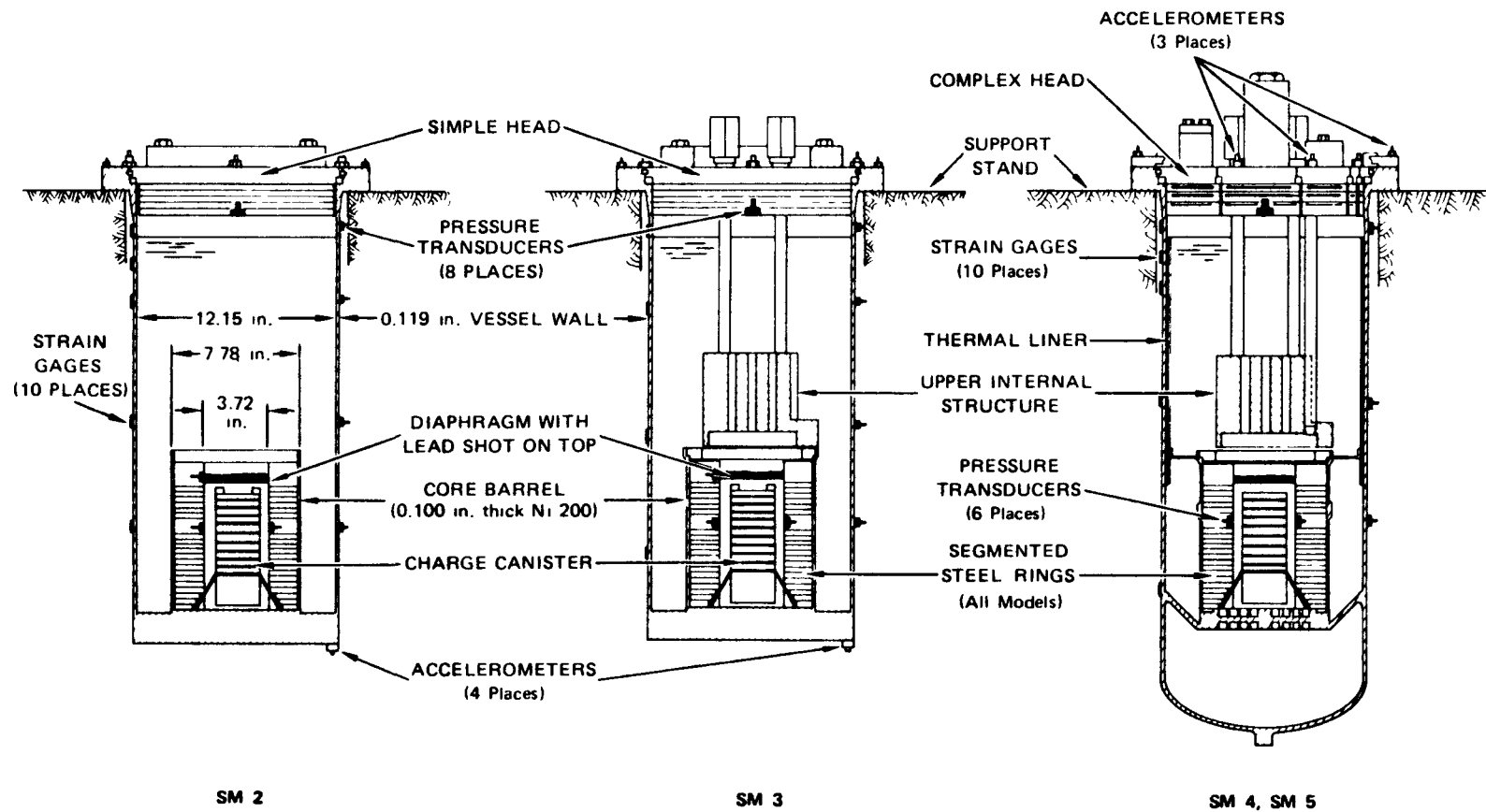


Figure 4-11 1/20-SCALE MODELS OF THE CRBR

- o SM-4 (Figure 4-11)

Test SM-4 incorporated the essential features of the reactor which were necessary to assess the gross system structural response. It included the upper internals structures, flexible vessel wall including thermal liner, core support structure with inlet plenum and a head which incorporates the three rotating plugs. The head shear-rings were accurately represented at 1/20th scale. The shield plates below the head were modeled to the extent that their impact on the head dynamic response were included.

This model was designed to represent the reactor features which have a significant bearing on the overall system response. Thus an accurate assessment of the primary loads (e.g., pressures, accelerations) was made with this test. Additional features (such as head mounted components) which only have a minor bearing on the system response were not modeled. Where possible results of SM-4 were used to support the design and analysis efforts to help validate the modeling used in CRBR assessments of HCDA loading conditions.

- o SM-5 (Figure 4-11)

In order to check the reproducibility of the system response, test SM-5, mechanically identical to SM-4 was run. SM-5 differed from SM-4 in that more extensive measurements were recorded.

- o SM-6

SM-6 was not tested.

o SM-7 (Figures 4-12 and 4-14)

Test SM-7 was performed to determine the static force deflection characteristics of the plug, shear ring and simplified under-head shielding combination. This test, performed by statically pressurizing the underside of a model of the head, had three objectives. The first was to obtain an understanding of the mode of the plug deformation when the stiffening provided by the simplified shielding is accounted for. This understanding is used when defining failure modes and criteria. The second was to obtain an estimate of the structural capability of this head/shielding configuration under pressure loading. The third was to assess the differences in stiffness between heads having and not having the simplified under-head shielding.

o SM-8 (Figures 4-13 and 4-14)

Test SM-8 was performed to determine the static force deflection characteristics of the plug, shear ring and a prototypic under-head shielding configuration. This test, performed by statically pressurizing the underside of a model of the head, had three objectives. The first was to obtain an understanding of the mode of plug deformation when the stiffening provided by the prototypic shielding is accounted for. This understanding is used when defining failure modes and criteria. The second was to obtain an estimate of the structural capability of this head/shielding configuration under pressure loading. The third was to assess the differences in stiffness between heads having the prototypic shielding, simplified shielding and no shielding.

D. Phase III. Static Tests of CRBR Head Models

Phase III was in progress when the CRBRP project was terminated in November, 1983. The SM-10 static test was completed but neither an interpretive report or a final report on SM-10 was completed. The SM-9 static test was not completed.

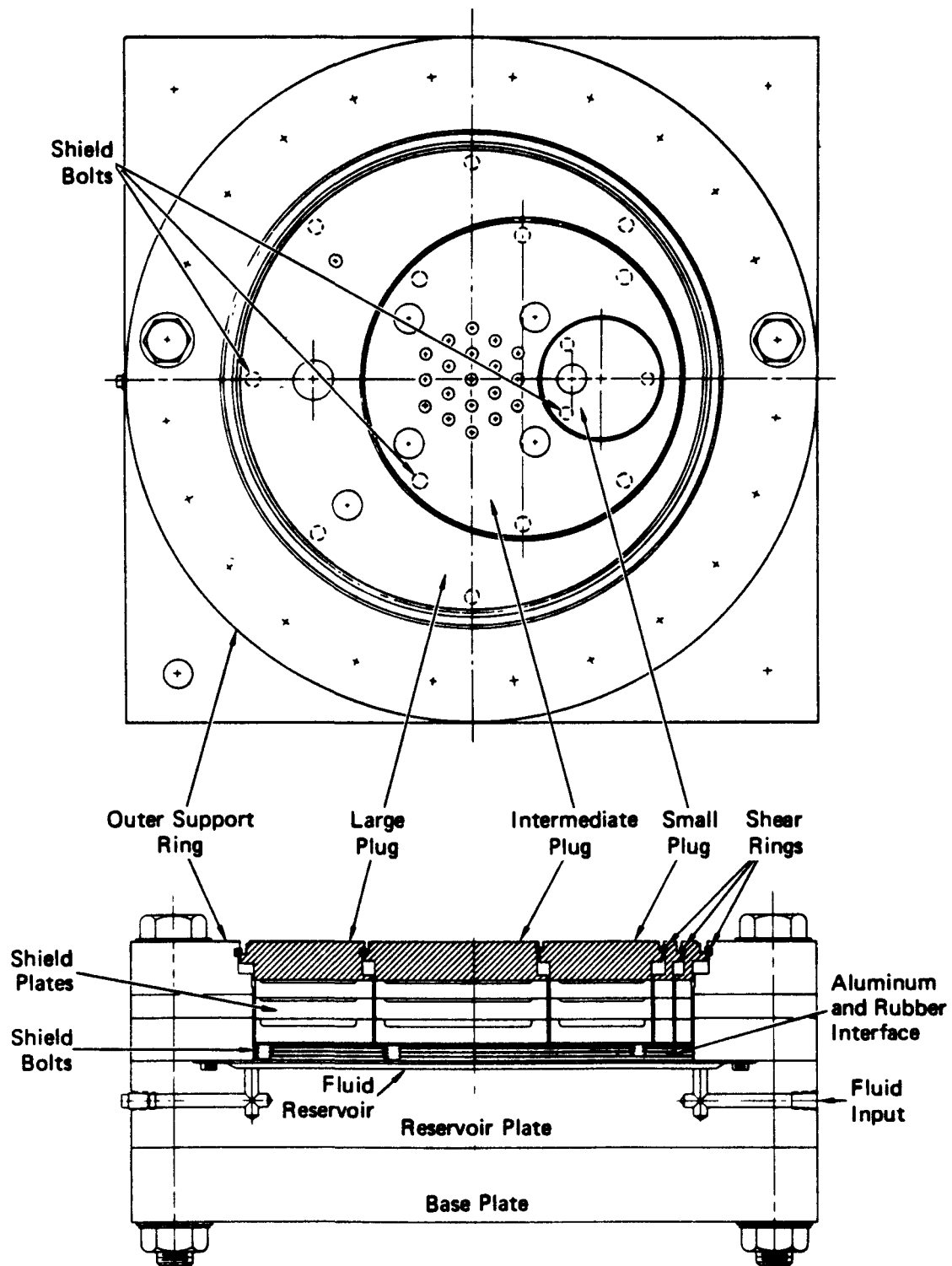


Figure 4-12 SCHEMATIC OF TEST FIXTURE FOR SM 7

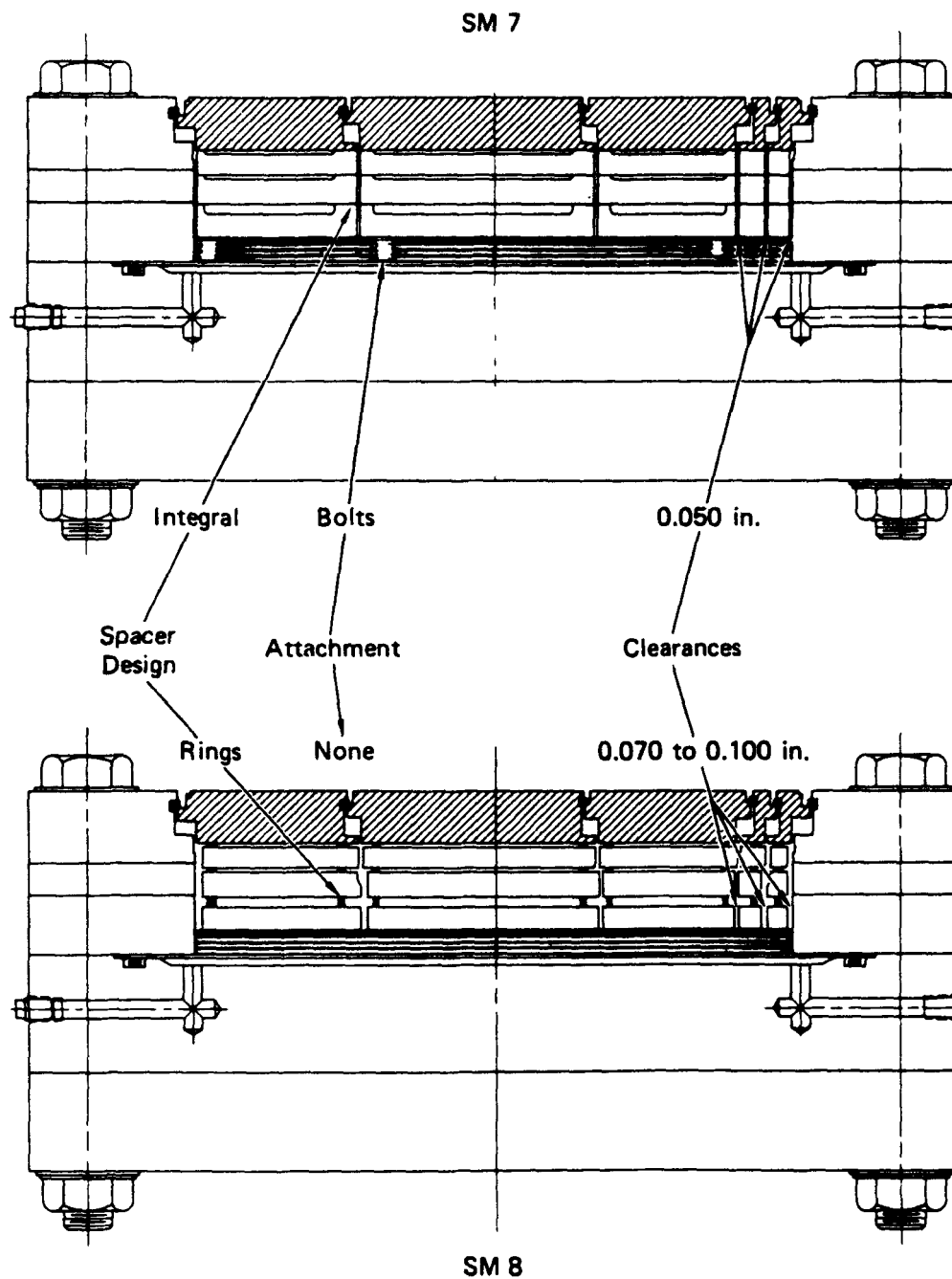
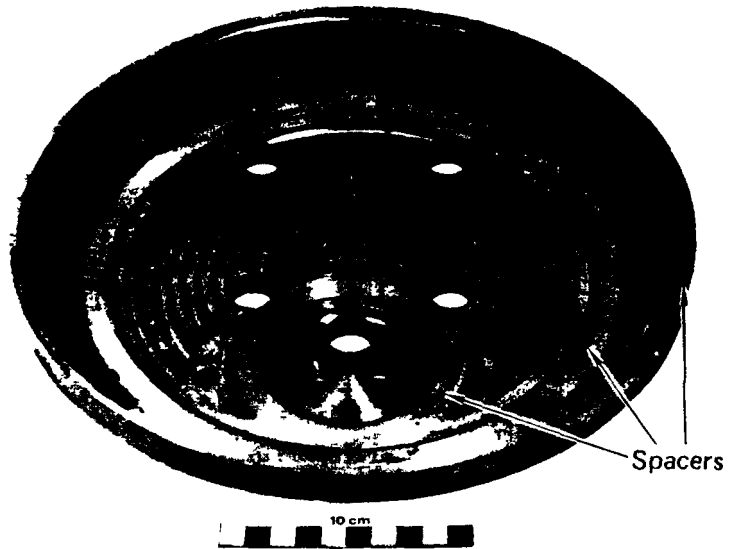
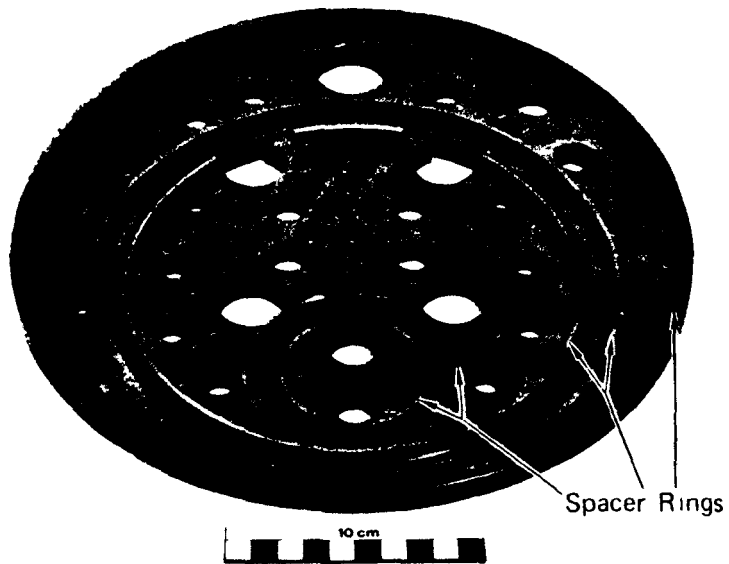


Figure 4-13 DIFFERENCES IN SM 7 AND SM 8 SHIELD PLATE DESIGNS



(a) Bottom layer of shield plates for SM 7



(b) Bottom layer of shield plates for SM 8

Figure 4-14 SHIELD PLATE DESIGNS FOR SM 7 AND SM 8

The upper and lower nozzle halves must be finished as matched sets in order to assure the proper fit and alignment of the pin grooves at final installation.

Special care should be exercised with the support block because it has already been custom fitted to the CSS.

F. References

Horizontal Baffle Assembly E-Spec, E-953031
Installation Requirements Document for WWM Components, WARD-D-0200
LRD-81-178, S. F. Bauer to R. L. Oelrich, 2/13/81
LMF-81-043, R. G. Van Voorhis to S. F. Bauer, 2/19/81.

Interface Control Drawing CW 50023, Horizontal Baffle

Drawings:

1184E56 Design Layout
235C138 Pin and Stop
5935D99 FT and SA Surveillance and Archive Material
9631D54 FT and SA Pin and Stop
9632D56 Key Latch
9632D82 Piston Ring Requirements
9632D86 Archives and Surveillance Material
1440E27 Lower Support Block Final Machining
1498E27 Lower Support Block Profile Machining
1499E01 Lower Support Block Installation
1537E14 Plug Support
1537E15 Plug Support Nut
1537E16 Top Half Plug
1537E17 Bottom Half Plug
1537E18 Locking Piece
1537E19 Plug Assembly
1537E25 Base Plate Assembly
1537E30 Base Plate

- o SM-10

The SM-10 model was similar to the SM-9 model with material removed from certain head plug and shield plug corners in a manner prototypic of the changes planned for the CRBR head and vessel components to provide increased energy absorbing capability.

These changes were intended to increase the energy absorbing capability of the head by preventing disengagement due to kinematic interaction between the deflected head plugs.

Information from SM-10, together with SM-9, was to provide the basis for the selection of the final design modification of the head and the basis for Phase IV confirmatory dynamic testing.

E. Phase IV. Confirmatory Dynamic Test

Following the evaluation of interim data from Phase III testing, a decision was to be made regarding a confirmatory dynamic test designated SM-11. This test was to be conducted similarly to SM-5 but not including the upper internals structure. This test was to confirm that the closure head structure would satisfy HCDA loading requirements by accurately simulating the dynamic loads and head structures in a dynamic scale model test.

- o SM-11

The SM-11 model to be used for dynamic testing was to be similar to the SM-5 model with the head components similar to the SM-9 or SM-10 Test. The SM-11 vessel was to have a prototypic flange, mounting ring, and method of attachment to simulate the support ledge. The simulated Upper Internals Structure utilized in the SM-5 model was to be omitted in SM-11.

Phase III was to consist of two static tests, SM-9 and SM-10. Load versus deflection data was to provide the basis for defining the energy absorption capability of the CRBRP full scale closure head in both its existing and modified form. The tests are described below.

o SM-9

Test SM-9 was to be performed to determine experimentally the capability of the current CRBR head design relative to the required HCDA loading. SM-9 was to model the head in a more prototypic manner than did SM-7 and SM-8 and was to model the reactor vessel flange in a manner similar to tests SM-7 and SM-8.

The SM-9 hydrostatic test model was to be a 1/20th scale model of all the CRBR head parts. The model was to include the complete riser assembly, margin ring, keepers and shield plug skirts which were not included in the SM-8 hydrostatic test model. The simulated head plugs in SM-9 were to include prototypic dip seal geometry. Also, the plug-to-vessel and plug-to-plug clearances were to be reproduced in more prototypic detail. The SM-9 model was to be mounted on a reservoir plate similar to that used in SM-8. The materials selected for the SM-9 model were as follows: the head plugs, vessel flange, and riser block material were to be the same material (SA-508, Class 2) as the CRBR head plugs and vessel flange selected from available heats of head plug material with materials properties matched as close as possible with these heats. The remaining model parts were to be made from non-prototypic materials with the mechanical properties (yield strength) closely matching the at-temperature values of the corresponding CRBR head components at normal steady-state operating temperatures.

SM-9 was to provide a baseline for comparison with SM-10 results.

4.2.5 Summary of Results for Test SM-1 [8]

The results of evaluation of the load-deflection data can be summarized as follows:

- o The volume of fluid between the head and base plate increased linearly with pressure up to a pressure of 700 psi. The slope of the pressure-volume change curve was 123 psi/in^3 . At 700 psi, the head began to yield and the volume to increase rapidly with increasing pressure.
- o The largest deflection occurred along the axis of symmetry of the head at the joint between the large plug and the intermediate plug nearest the center of the head (see Figure 4-15). A peak deflection of 0.416 in. was measured at this point at a pressure of 1160 psi.
- o Only the large plug showed large permanent plastic deformation after the test. The other plugs and the outer ring showed little plastic deformation.
- o The mechanical process that led to large deformation of the head was the disengagement of the shear ring between the large and the intermediate plug as the large plug deformed plastically. This shear ring gradually slipped off the bearing surface of the intermediate plug as the intermediate plug was pushed up through the large plug. At no point did a shear ring or any bearing surface fail catastrophically. Upon unloading, the head remained locked in the final deformed shape.

4.2.6 Summary of Results for Tests SM-2, SM-3, SM-4 and SM-5 [7]

The results can be summarized as follows:

- o No plastic deformation of the shear rings or head occurred in any of the four dynamic experiments. The head remained tightly sealed by the O-rings above and below the shear rings in each experiment. The peak

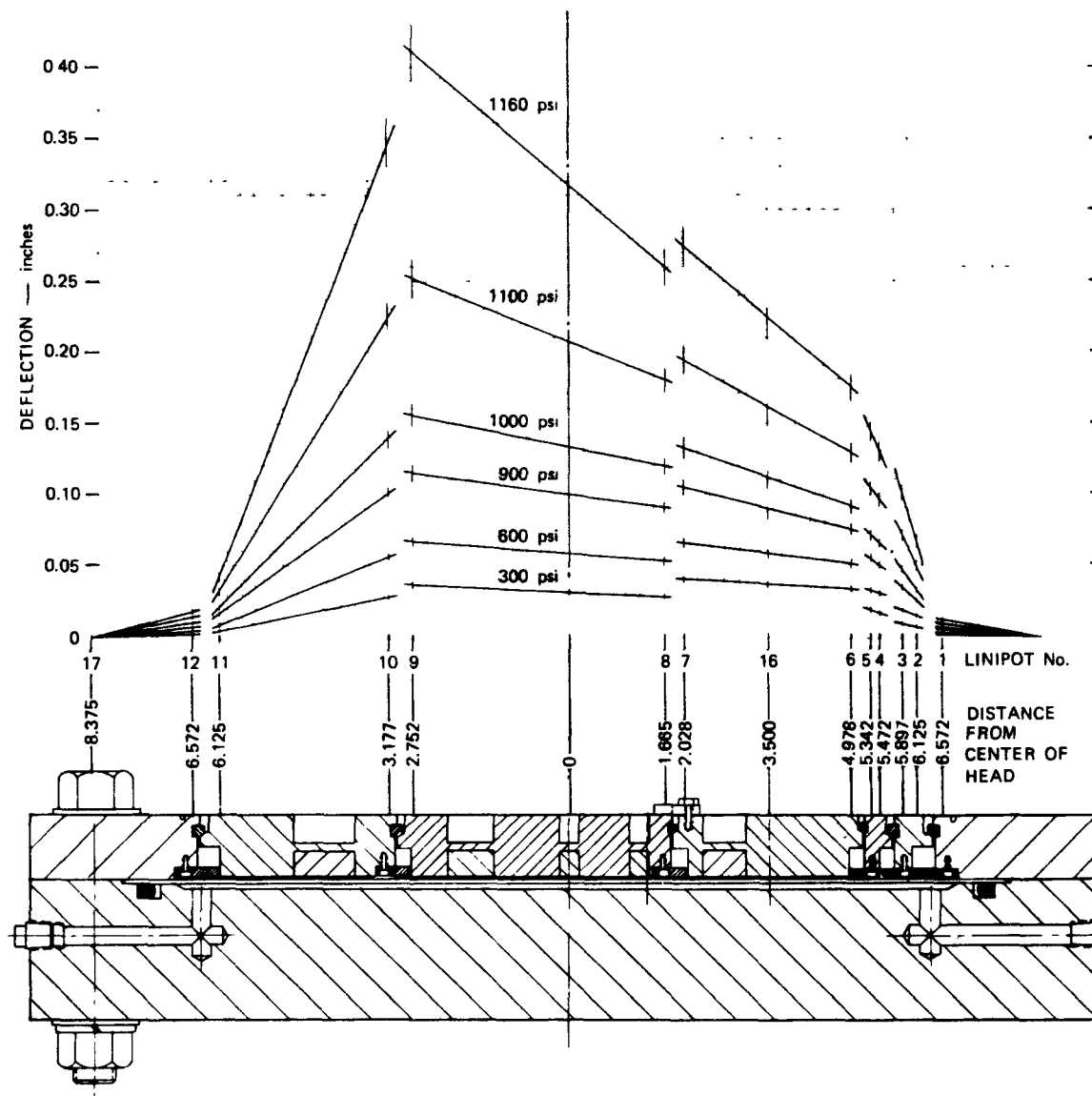


Figure 4-15 PROFILES OF HEAD AT SIX PRESSURES; STATIC TEST SM 1

head pressure reached 5300 psi on SM-2, which resulted in the peak head acceleration of 4800 g.

- o The upper internals structure significantly reduced the slug impact velocity, and hence, impact pressure. The peak impact pressure was reduced by 34% from 5300 psi on SM-2 to 3500 psi on SM-3. Consequently, there were reductions in the deformation of the upper vessel wall where slug impact pressures produced the largest wall strains (4.5% strain on SM-2 compared with 2.8% on SM-3). Along the vessel wall near the UIS, strain was reduced from 3.5% on SM-2 to 1.8% on SM-3.
- o The overall structural response of SM-5, a complex model with extensive instrumentation, was nearly the same as for SM-4. (SM-4 is identical to SM-5 but had less instrumentation and had unannealed UIS columns). The same load was used in both tests. The presence of unannealed columns in SM-4 did not appreciably affect slug kinetic energy, or strains in the model. The slug impact velocity in SM-4 was 62.4 ft/sec and in SM-5, 62.2 ft/sec. The peak strain in SM-5 was 1.9% and in SM-4, 1.4%.
- o The good agreement in structural response between SM-4 and SM-5 demonstrated the reproducibility of the tests and insensitivity of model response to the yield strength variation of UIS columns. Because of this reproducibility and because of the more extensive instrumentation, the results of SM-5 were used in the remaining conclusions on complex model response.
- o The increased stiffness (about 79%) of the vessel wall of the more complex model SM-5, over that of SM-3 (caused by the combined effect of the thicker vessel wall and the thermal liner) greatly reduced the permanent deformation of the vessel. Peak strains in the upper vessel wall were reduced from 2.8% in SM-3 to 1.9% in SM-5. Peak strains in the vessel wall near the UIS were reduced from 1.8% in SM-3 to 0.3% in SM-5.

- o The plastic strain energy absorbed by SM-2 was 6.9 kW-sec, which was 47% of the gas work expended by the explosive up to the final volume change of the vessel (cover gas volume plus volume change of the vessel). For SM-3, only 4.5 kW-sec of strain energy was absorbed, which was only 32% of the gas work expended. Thus, the presence of the UIS resulted in a 35% reduction in strain energy absorption. The strain energy absorbed by SM-5 was only 2.1 kW-sec, which was 15% of the gas work expended. Thus, the stiffer vessel wall on SM-5 resulted in a further reduction in strain energy absorption.

- o General Electric (GE) compared pre- and posttest REXCO calculations with experimental results and found that pretest REXCO strain calculations were conservative: REXCO overpredicts peak strains. This inconsistency was due to the use of low-strain-rate properties of Ni 200 and inadequate modeling of the core and UIS. For example, predicted peak vessel strain for SM-2 was 5.2% compared with a measured peak strain of 4.5%. The predicted slug impact pressure for SM-2 was 3292 psi compared with a measured impact pressure of 5300 psi. Use of high-strain-rate material properties for Ni 200 in the GE version of REXCO gave better agreement between experiment and code, but REXCO strain calculations were still overestimated and impact pressures were still underestimated.

The accuracy of the instrumentation based on manufacturers specifications and repeated calibration was estimated to be $\pm 2\%$ for pressure gages, $\pm 2\%$ for strain gages, $\pm 5\%$ for accelerometers, and $\pm 4\%$ for water surface gages. The signal conditioning and recording system for these gages reduced overall accuracy to $\pm 5\%$ for pressure and strain measurements, $\pm 7\%$ for accelerations and $\pm 6\%$ for water surface measurements. Pressure and accelerometer records were digitally filtered with a low pass filter that had a cutoff frequency of $1/5$ the natural frequency of the gage (50 kHz for pressure gages and 16 kHz for accelerometers) which represents the linear limit of the gages. Selected pressure records and all strain records were filtered to a lower cutoff frequency (10 kHz-25 kHz) without removing important structural response information (the highest structural response frequency for the 1/20-scale CRBR model is about 5 kHz). Based on four calibration experiments, the energy

source was reproducible to within $\pm 5\%$ at high pressures and to within $\pm 20\%$ when the gas had expanded to the cover gas volume. Material property tests were accurate to within $\pm 10\%$.

Measured pre- and posttest radial deflections were accurate to within 1 mil. The out-of-roundness was ± 30 mils for the worst case. This out-of-roundness represents 14% of the maximum deflection, in regions of large deflection.

4.2.7 Summary of Results for SM-7 and SM-8 [9]

The two static tests of models of the CRBR head provided useful qualitative and quantitative data on the effects of shielding on the response of the head to HCDA loading. The primary differences in the two models centered on the design and attachment of the shielding to the three-plug models of the head. On model SM-7, the shielding and method of attachment were the same as that used on dynamic test SM-5; that is, the spacers between layers of shielding were machined into the shielding plates, and the plates were tightly fastened to the head with several strong bolts. The shielding on SM-8 was more prototypic. It included separate, thin spacer rings between shield layers and no bolt attachment of the shield plates to the head.

The following points summarize the important observations and conclusions drawn from the results of these two static tests.

- o The bolts and integrally machined spacer rings on shield plates of model SM-7 strengthened the head. The head with more prototypic shielding (SM-8) had a stiffness that was 33% lower than the head with bolted shielding (SM-7) (See Figure 4-16).
- o The head failed as a result of disengagement of the shear ring between the large and intermediate plugs. (See Section 4.3.2 Key Problems and Resolution). This failure was a gradual process that started at a point along the shear ring interface nearest the center of the head at a displacement of 125 to 130 mils. Disengagement continued until the head could no longer carry a load. The displacement at failure was 250 mils. For SM-7, the failure load was 2587 psi, and for SM-8, the failure load was 2010 psi.

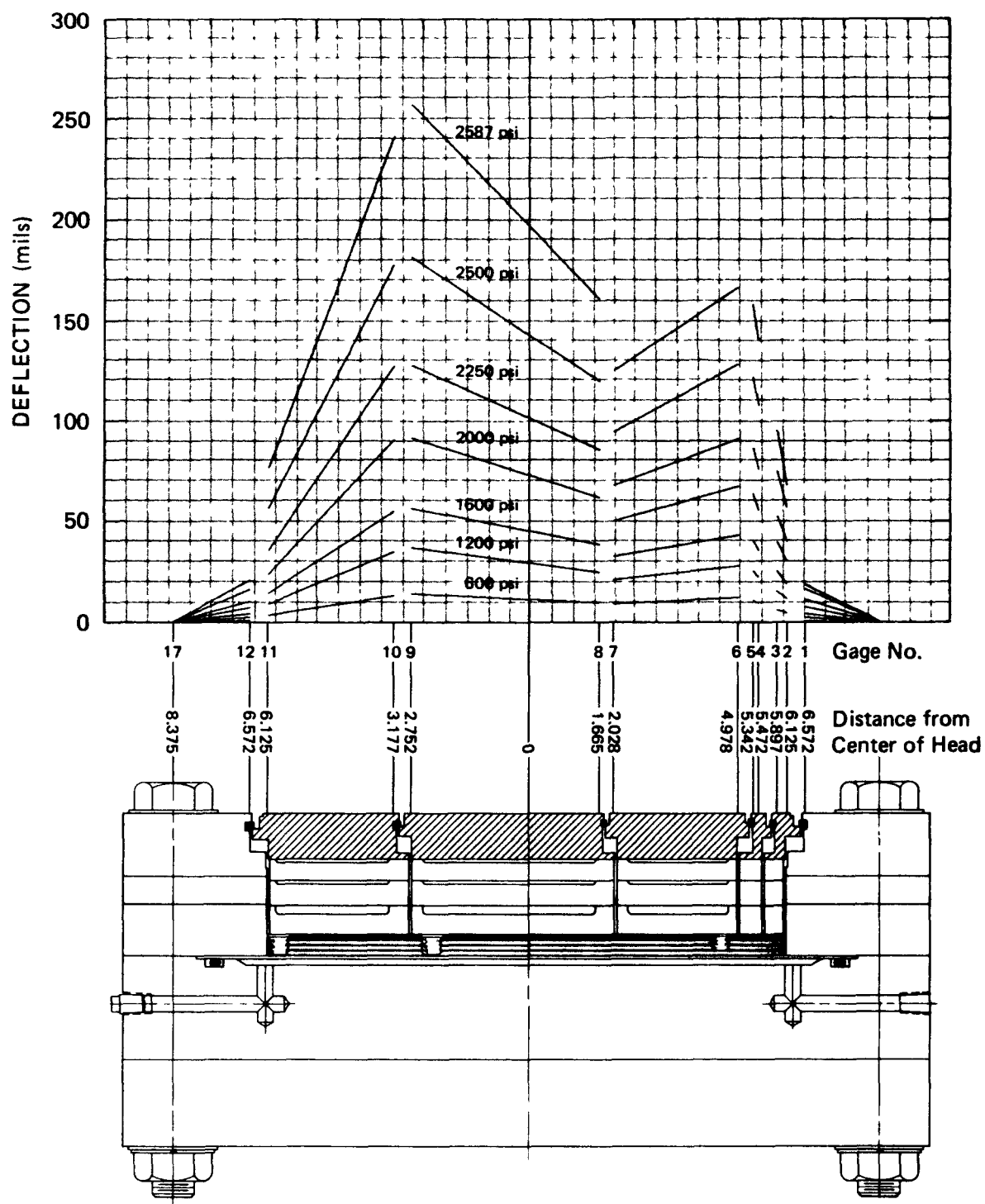


Figure 4-16 PROFILES OF HEAD AT SEVEN PRESSURES FOR SM 7

- o Load-deflection and load-volume change records for both tests were mostly linear up to a displacement of about 50 mils (1500 psi) on SM-7 and of about 65 mils (1100 psi) on SM-8. At these displacements, the head began to deform plastically.
- o Data indicated that shielding "lockup" did not occur because shield plates could slide and realign themselves when contact was made between adjacent plates.
- o An equivalent single degree of freedom (SDOF), dynamic response analysis that was based on an asymmetric model of the CRBR head with bolted shielding (SM-7) and with nonuniform mass distribution and a bi-linear force-deflection relationship predicted a peak dynamic displacement of 89 mils under HCDA slug impact loading. This displacement was about 50% higher than the displacement measured on test SM-5 (~60 mils). A comparison of an SDOF analysis and a linear elastic finite element analysis of a three-piece plate indicated that the SDOF overpredicted peak displacement by about 16%. This degree of conservatism did not explain the difference between the SDOF predictions and the measurements on SM-5.
- o The comparable SDOF analysis for the head with more prototypic shielding design and attachment (SM-8) predicted a peak displacement of 110 mils under HCDA slug impact loading. This displacement was 24% larger than that predicted for the head with bolted shielding.
- o An explanation for the difference between the measured displacement on SM-5 and the predicted displacement using the SDOF technique was the restraining influence of the UIS on the rotation and displacement of the IRP to which the UIS was attached. Because the UIS was keyed into the core barrel and required 225 mils of upward translation for disengagement from the barrel, the UIS could not displace laterally. This constraint forced the UIS columns to apply a moment to the IRP as it tried to rotate during upward displacement.

- o Results of the static tests SM-7 and SM-8, the dynamic test SM-5, and the SDOF analysis pointed out the importance of the UIS in limiting head deflection. As demonstrated by test SM-5, the UIS ensured a safe dynamic response of the SM-7 head and probably would ensure a safe response of the SM-8 head because it would remain engaged in the core barrel.
- o Any future analysis should include the UIS.

4.2.8 Status of Tests SM-9, SM-10, and SM-11

The status of tests SM-9, SM-10 and SM-11 is as follows:

- o SM-9 Static test was not completed.
- o SM-10 Static test was completed.
- o SM-11 Dynamic test was not completed.

4.3 Key Problems and Resolutions

4.3.1 Sodium Slug Energy Absorption Capability of a Modified CRBRP Closure Head [10]

SMBDB design requirements for the CRBRP closure head included a requirement for the head to accommodate the energy deposited by 75MJ sodium slug.

Tests on a 1/20 scale model of the closure head resulted in disengagement of the margin shear rings at pressures and deflections corresponding to a sodium slug energy of 40.8MJ. Investigation of the cause of the premature head failure revealed an unanticipated kinematic interaction between the deflected head plugs, which had the effect of disengaging the margin shear rings as head deflections increased.

Elimination of the kinematic failure mode can be accomplished by local machining of non-load bearing portions of head plugs and shield plates. In its modified configuration the head system had been shown to have the potential capacity to accommodate the energy from a 156MJ slug. The margin

existing between the closure head predicted and required energy absorption capacities was considered more than sufficient to cover any uncertainties in the evaluation procedures. Confirmation of the energy absorption capability of the modified head was to be obtained in additional hydrostatic and dynamic model tests.

Pressure loading produced by an HCDA event causes the sodium within the reactor vessel outlet plenum to be accelerated upward towards the closure head. The kinetic energy content of the slug at the instant of impact with the closure head is defined in Table 5-2 of Reference 2 at 75.1MJ. Static loading tests performed by SRI International on 1/20 scale models of the closure head [9] indicate that the closure head would fail by plug disengagement pressure loading of 2010 psi. Integrating the area under the model closure head load deflection curve up to the collapse load limits defined by Table F-1322.2.1 of Reference 11 produced an allowable slug kinetic energy for the existing closure head design of 40.8MJ. Objectives for the design and analysis tasks summarized in Section 4.3.2 were:

- A. To determine the cause of the premature head failure,
- B. Identify modifications which would increase the head failure load and associated energy absorption capability and,
- C. Determine the slug energy absorption capability of the modified head.

4.3.1.1 Analysis of the SM-8 Model Closure Head Failure Mode

Deflection mode shapes for the SM-8 test are given in Figure 4-17. Model plug layouts were made corresponding to the deflection mode shapes. The layouts were made at 1/5 reactor scale (4 times model scale). The layout showed early closure of the gaps between the IRP and LRP on the lower surface of the head (see point 1 on Figure 4-18).

Premature closure of the gaps between plugs at the lower surface of the head introduced an unanticipated kinematic failure mode at the shear ring location. As the pressure increased beyond that associated with gap closure

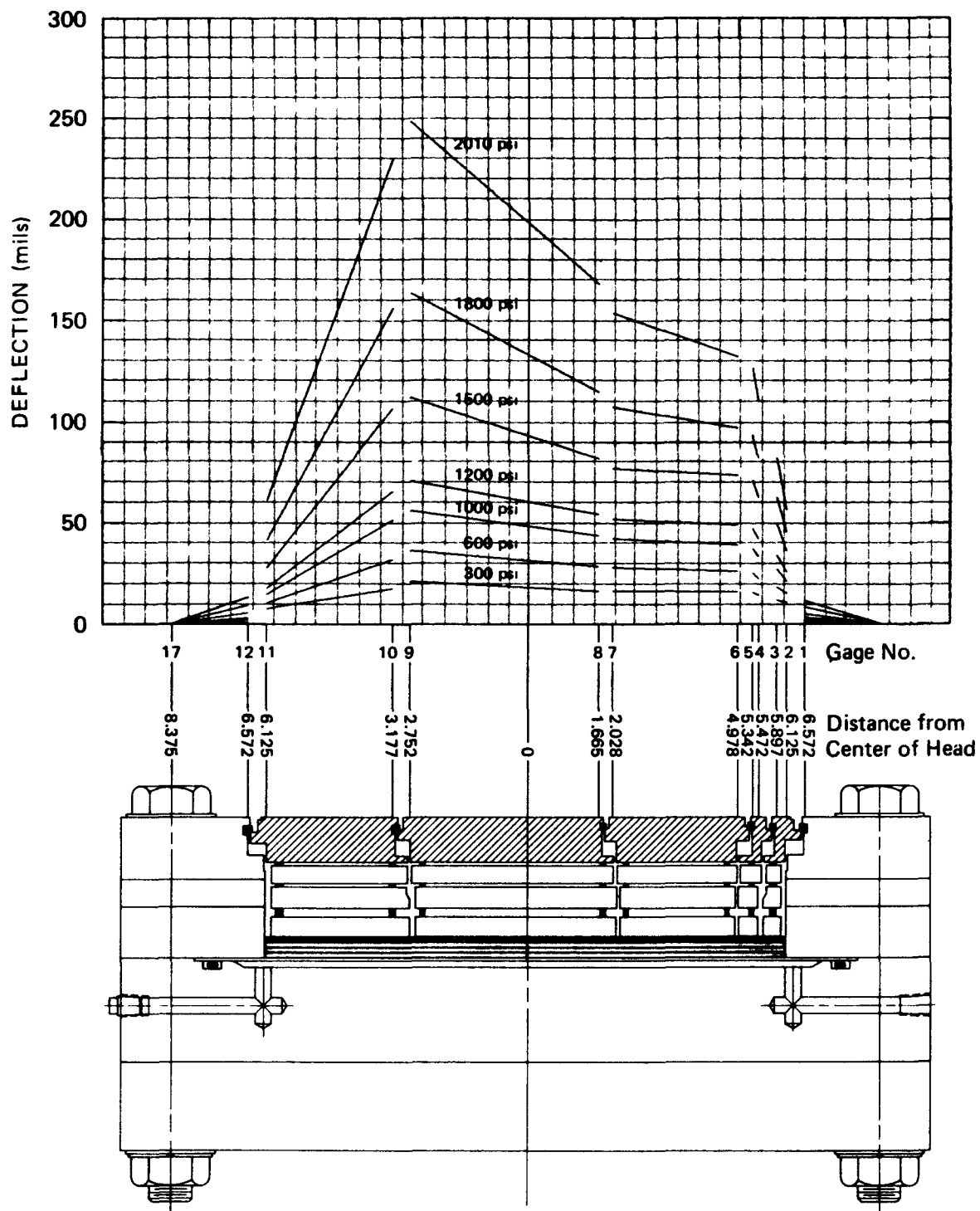


Figure 4-17 PROFILES OF HEAD AT SEVEN PRESSURES FOR SM 8

SM-8 MODEL
PLUG CONFIGURATION
AT FAILURE (2010 PSI)

Figure 4-18

at point 1, further upward deflection of the plugs caused the IRP and LRP plugs to pivot about point 1. The pivoting action produces a lateral motion of the plugs at the elevation of the shear ring which ultimately led to disengagement of the shear ring. Shear ring disengagement was the failure mode observed in the SM-8 tests.

Visual inspection of the component parts following the SM-8 test showed them to be in good condition with no evidence of structural failure. It followed, therefore, that the model had additional load carrying capability, if the premature kinematic disengagement of the shear ring was eliminated. In this connection it was worth noting that the contacting surfaces at point 1 were in the non-load bearing portions of the head associated with the dip seal. It was possible, therefore, to eliminate the cause of the premature disengagement by locally machining the head to eliminate the interaction at point 1, without adversely impacting the structural strength of the individual plugs. This observation provided the basis for the closure head modifications evaluated in subsequent sections of this report.

It is important to note that the prying action associated with premature contact at point 1, resulted in an overall lateral motion of the entire IRP within the clearances provided in the LRP cavity. In the reactor closure head design this lateral motion would be resisted by two mechanisms. These are:

- A. Shear stiffness of the riser assemblies which bridge the gaps between the rotating plugs and
- B. Closure of the gaps between the margin shear ring keeper rings and the plugs once the plug lateral motion had exceeded the radial clearance at these locations. Neither the riser assemblies nor the margin shear ring keeper rings were represented in the SM-8 test. Failure in the SM-8 test would have been delayed to a higher pressure loading had these features been present. A preliminary evaluation of the effect of the missing features with the existing design, however, indicates that energy absorption capacity of the closure head would be less than the required 75 MJ capacity.

It was concluded that modifications in the form of machining relief of the interfering portions of the closure head plugs and shield plates were therefore required. It is important however to recognize the effect of the risers and margin shear ring keeper rings in resisting lateral differential plug motions when assessing the effect of the proposed design modifications.

4.3.1.2 Modification Configuration Definition & Verification

The action plan used to resolve the unanticipated kinematic failure mode was to:

- A. Define configuration of the modified plugs and shield plates
- B. Verify acceptable structural performance of the modified configuration and
- C. Define the slug energy capability of the modified head design as defined in Figure 4-19.

4.3.1.2.1 Configuration Definition

Extrapolation of the deflection mode shapes of Figure 4-17 to pressures beyond the 2010 failure pressure was done using individual node load deflection plots. Typical plots were shown in Figure 4-20. The plots can be divided into 3 sections. An initial section corresponds to largely elastic action of the plug assemblies. This section extends to ~1300 psi in Figure 4-20. Thereafter, the pressure deflection curve takes on a higher slope, indicating yielding of the plug material. The third section in which non-linear deflection of the plug is observed is associated with progressive disengagement of the margin shear rings. In the curves of Figure 4-20 this effect starts to become pronounced at pressures of about 1800 psi and becomes dominant at 2010 psi.

The first two regimes in the curve can be related to bi-linear plug deformation. The plug strains at the instant of plug disengagement are still very much less than the limit of uniform elongation for the plug material.

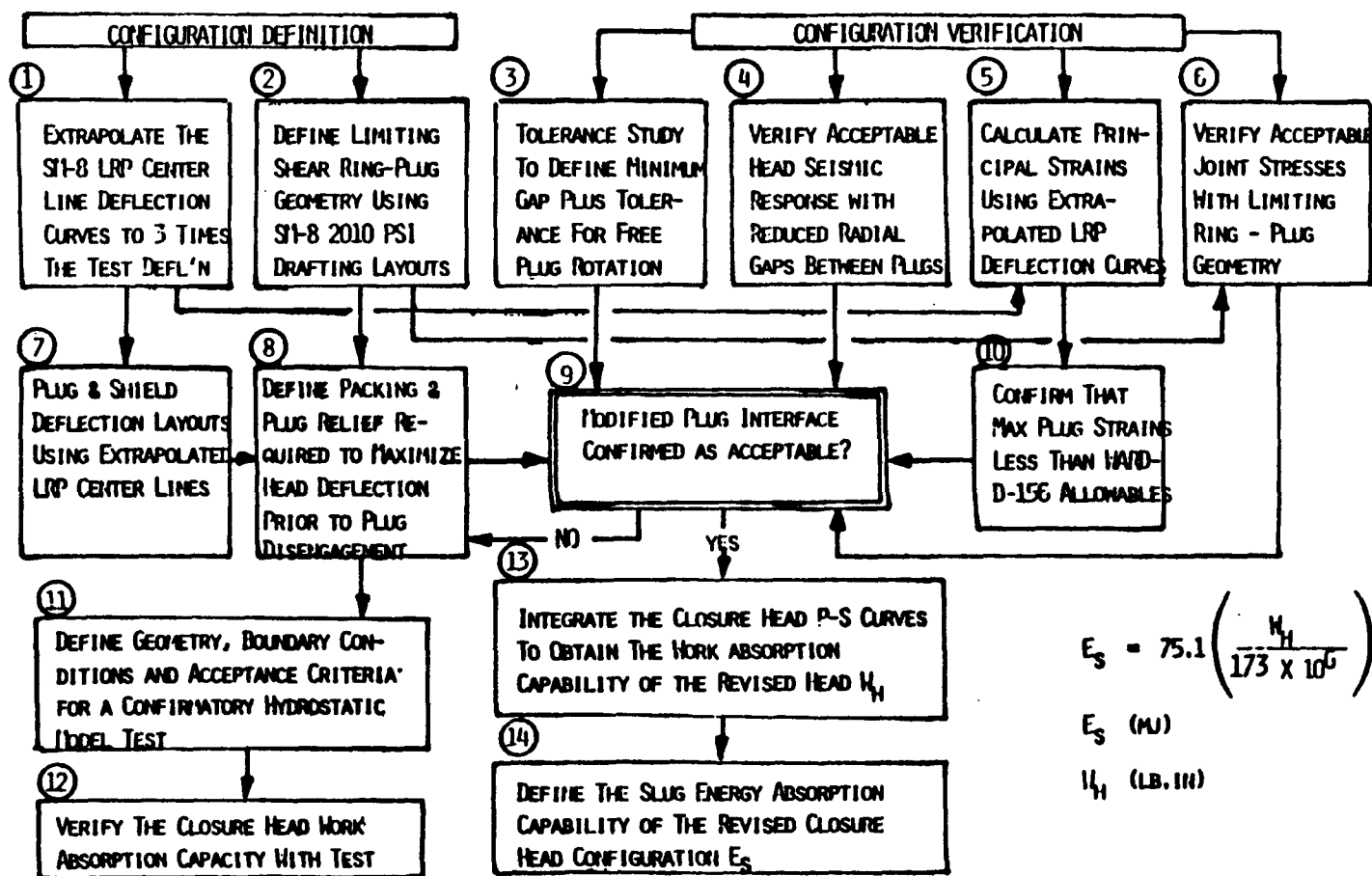


Figure 4-19

Design Modification Logic CRBRP Closure Head SMBDB Capacity Enhancement

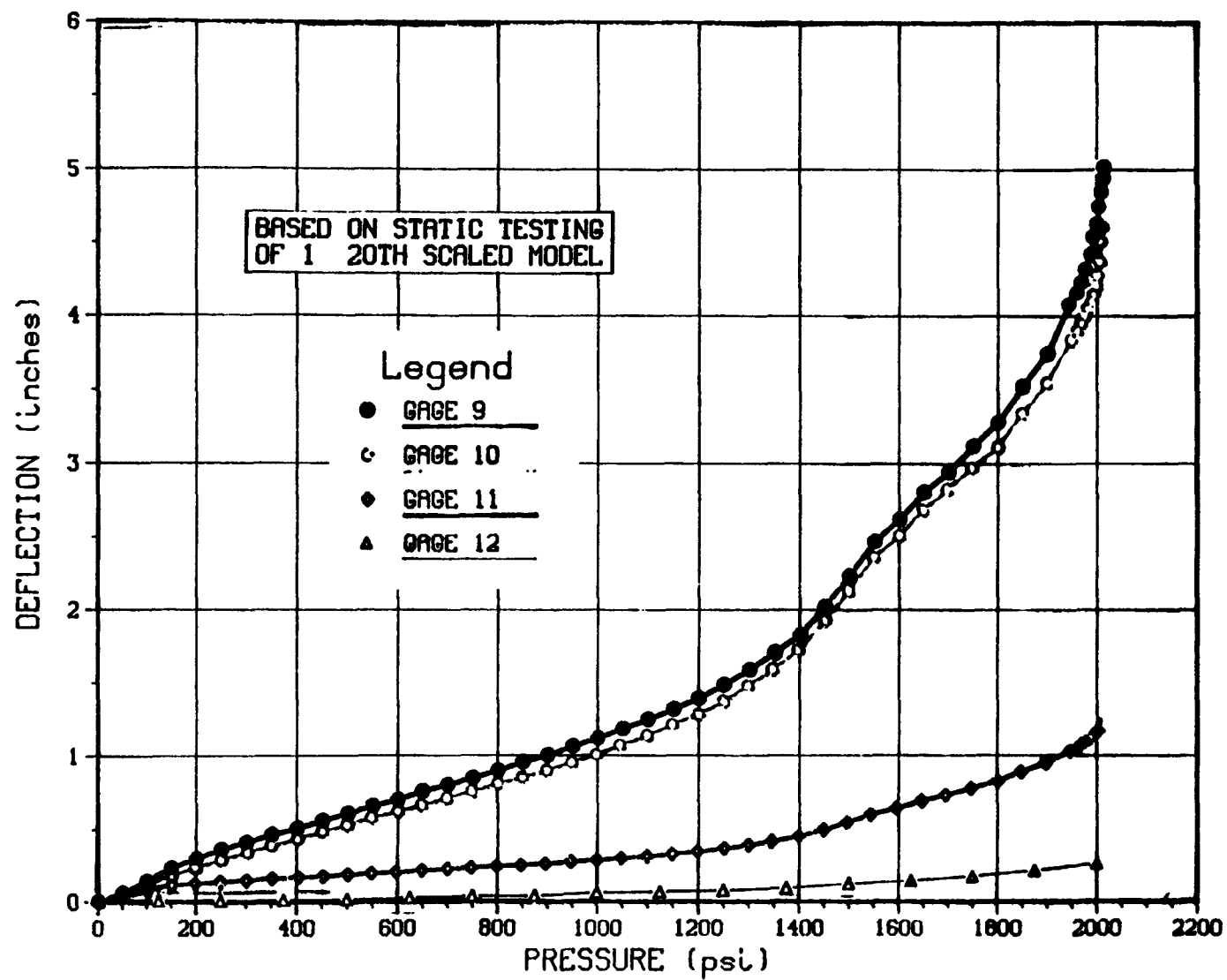


Figure 4-20

CRBR RV and Rotating Plugs Absolute Displacements Due to HCDA

Plug deflection in the absence of premature disengagement can, therefore, be estimated by extrapolating the second portion of the load deflection curves. Samples of extrapolated curves are provided in Figure 4-21. A family of these extrapolated curves was used to define the deflected shape of the plugs at higher pressures.

The effect of disengagement at the shear ring connection is illustrated in Figure 4-22. At pressures in excess of 1600 psi, differential motion between the nodes on either side of the IRP-LRP gap becomes pronounced. The extrapolated curves did not include this element of plug deflection since the design modifications proposed delays plug disengagement such that it ceases to be a significant factor at loadings associated with the 75MJ slug. The non-linear disengagement effects were, however, considered in determining the ultimate energy absorption capability of the modified closure head.

Layouts of the closure head plugs were made at 1/5 scale for a range of pressure loadings. Layouts produced for 2700 psi and 3000 psi head loading are shown in Figures 4-23 and 4-24 respectively. The significance of these pressures is that 3000 psi conditions corresponds to the limiting shear strength of the LRP to reactor vessel joint, and 2700 psi corresponds to the load which is permitted by the collapse load criteria of Reference 11.

Examination of Figures 4-23 and 4-24 shows interference between the IRP and LRP plugs and shield plates at the junction between nodes 9 and 10, interference between the IRP and SRP plugs and shield plates at the junction between nodes 5 and 6 and some interference between the LRP and the shear ring keeper ring at the junction between nodes 1 and 2. Machining modifications to the plugs and shield plates to eliminate the interference between the plugs and shield plates are shown in Figures 4-25 and 4-26. It is estimated from these figures that the machining modifications proposed do not influence the load carrying capability of the individual plug elements. The interference between the LRP and the existing shear ring keeper ring will inhibit rotation of the LRP ring and thereby increase engagement of the LRP-IRP shear rings. For this reason, the existing shear ring keeper rings will remain in their present configuration in the modified head. Head deflection profiles reflecting the effect of interaction between the LRP and the shear ring keeper

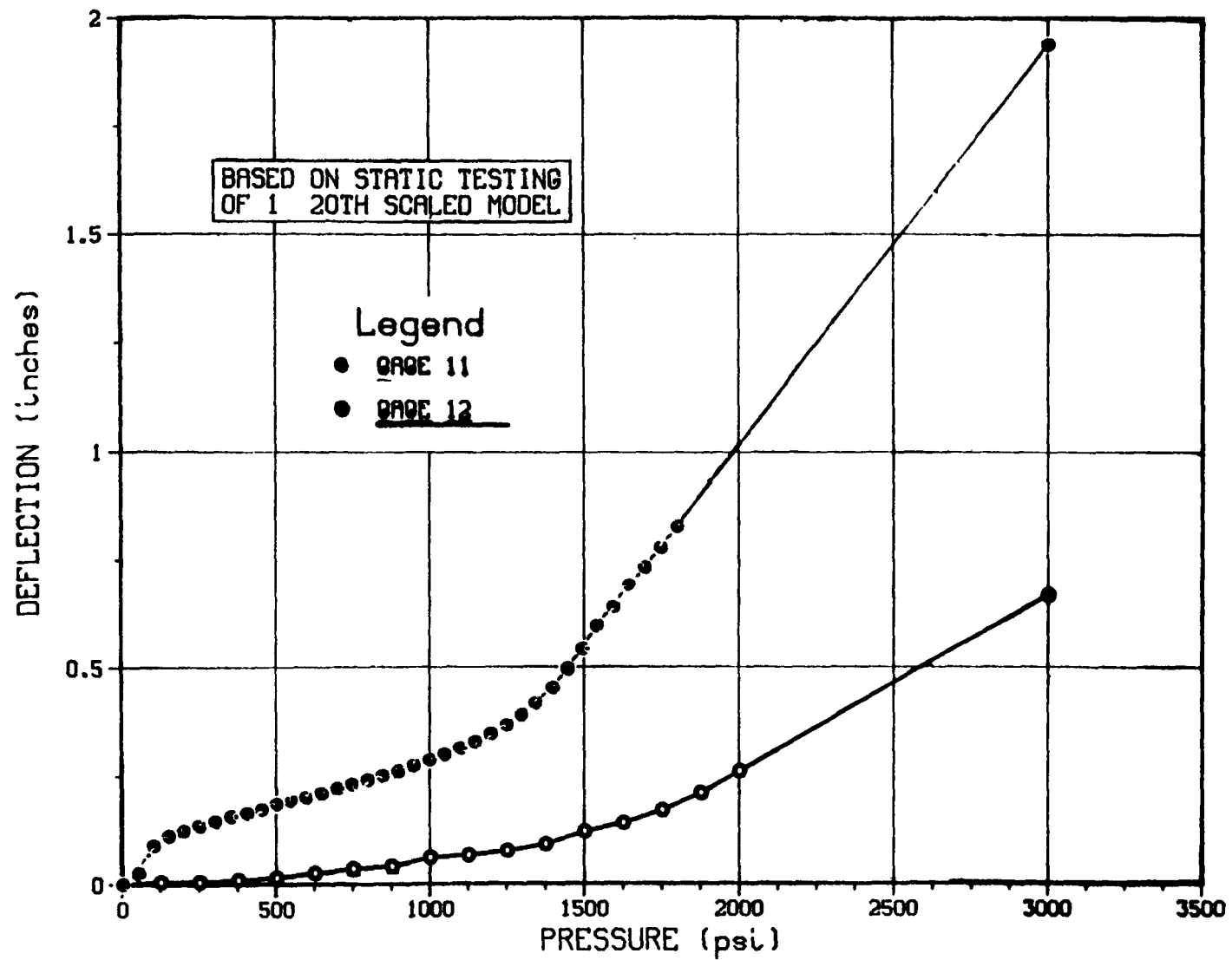


Figure 4-21

CRBR RV and Rotating Plugs Absolute Displacements Due to HCDA

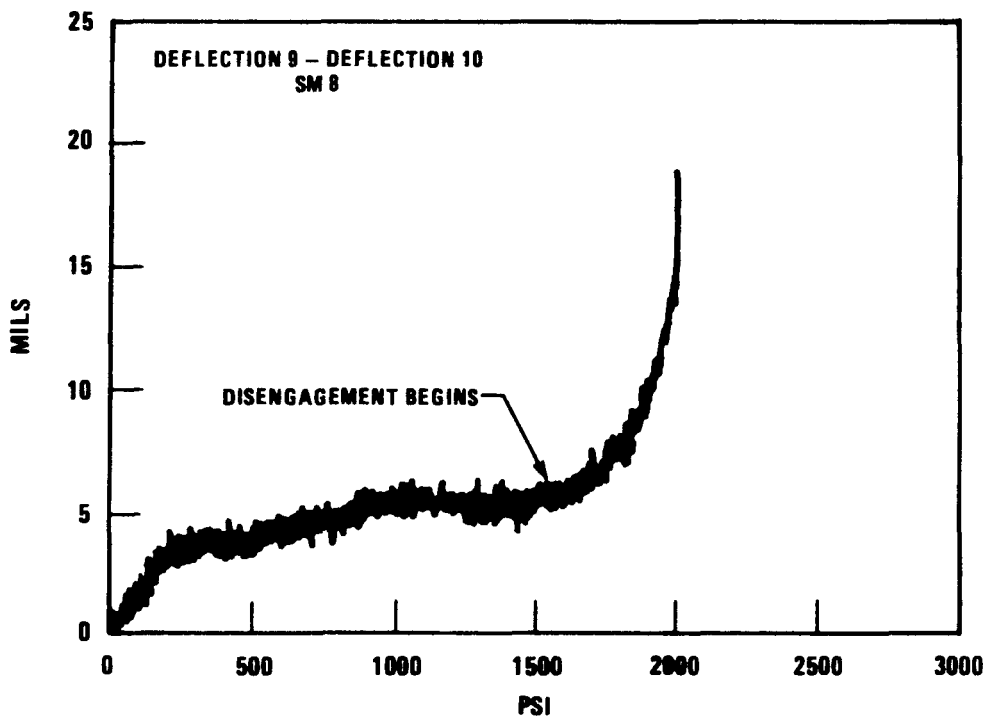
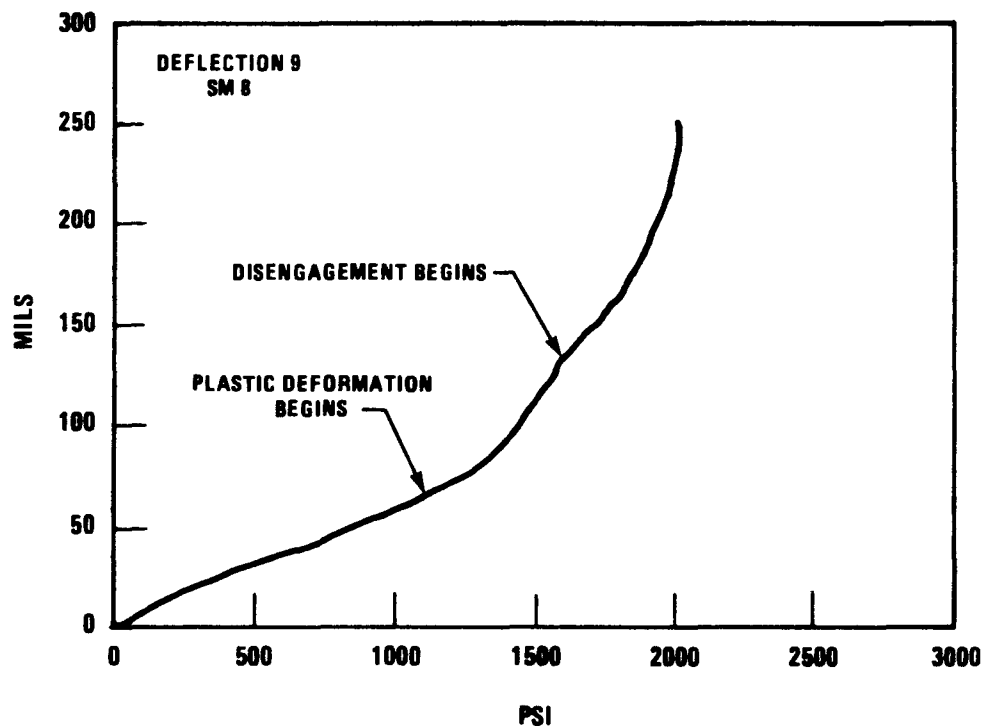
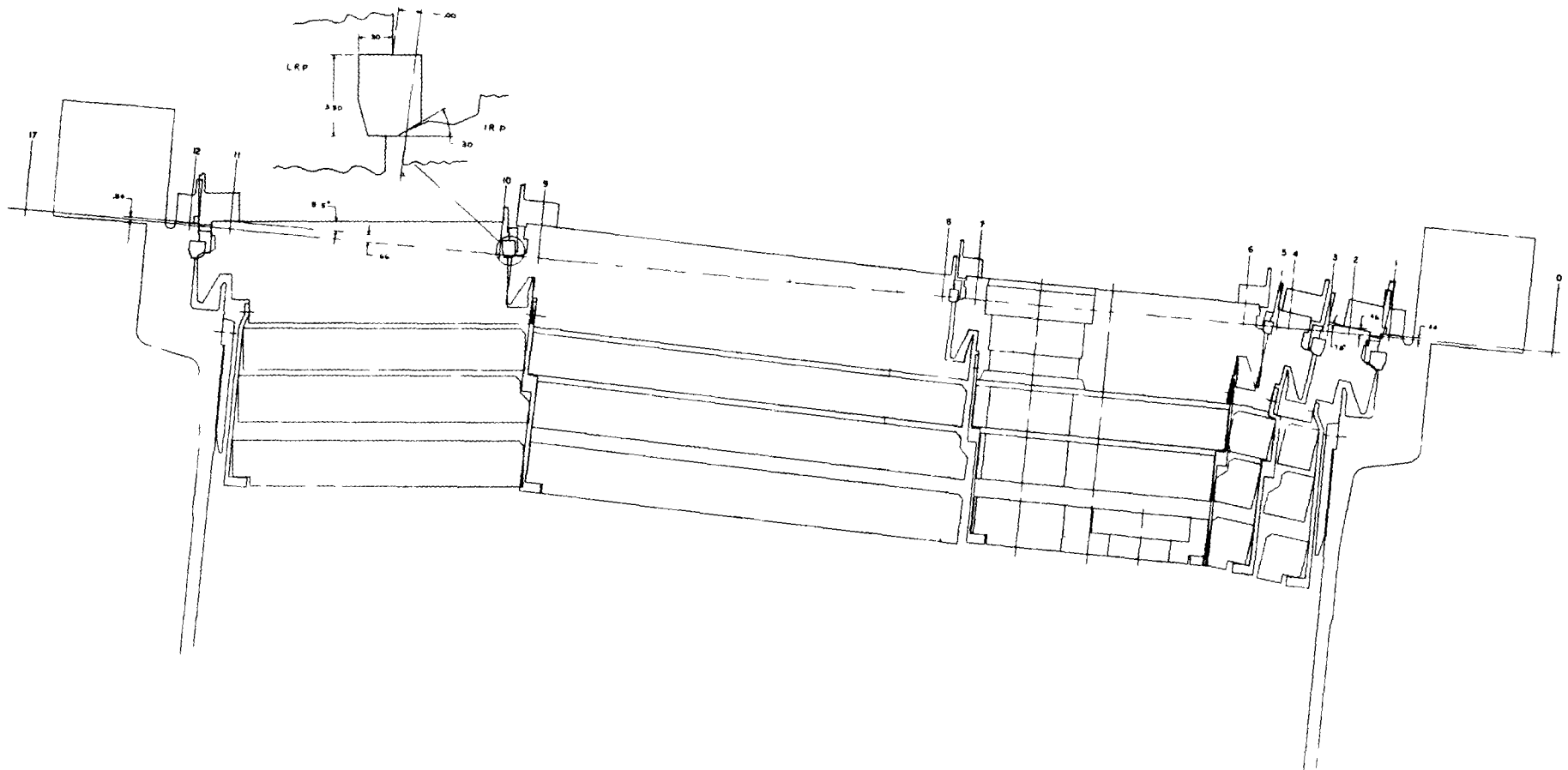


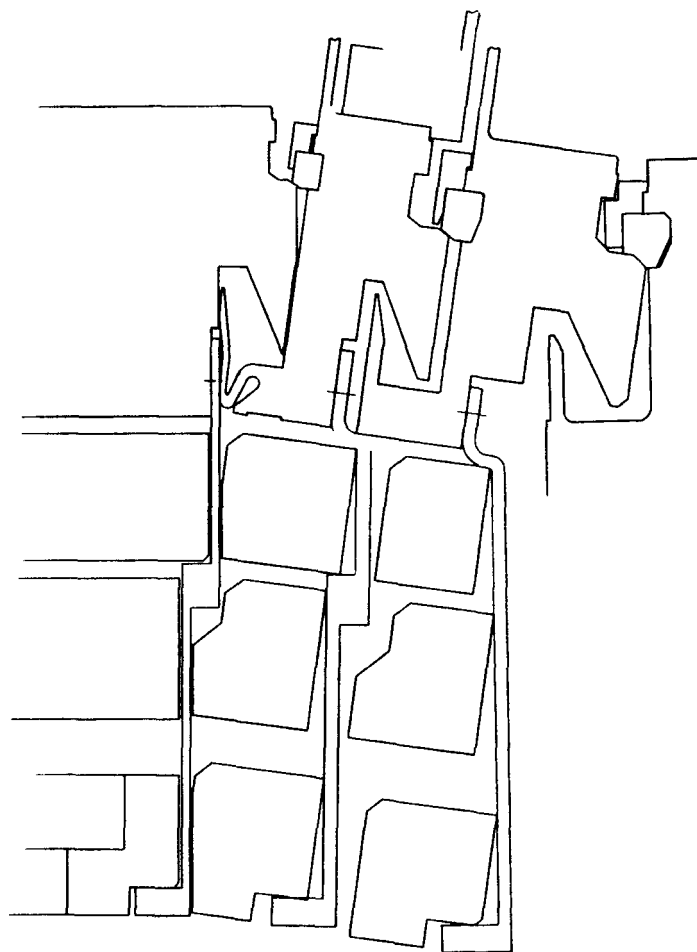
Figure 4-22
Relative Displacements Across Point of Disengagement for Tests SM 7 and SM 8



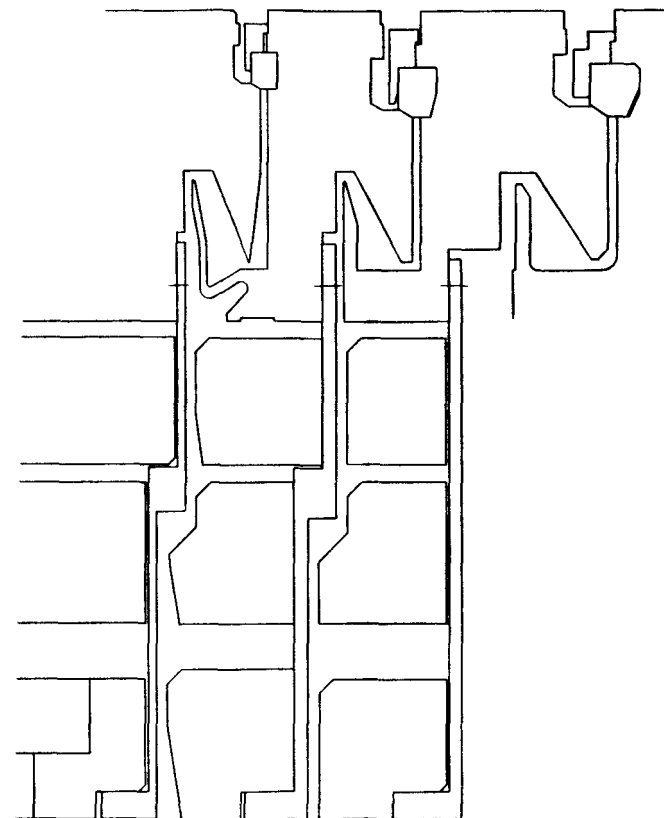
PROJECTED CLOSURE HEAD
DEFLECTIONS AND INTERFERENCES
AT 2700 PSI

Figure 4-23

Figure 4-24



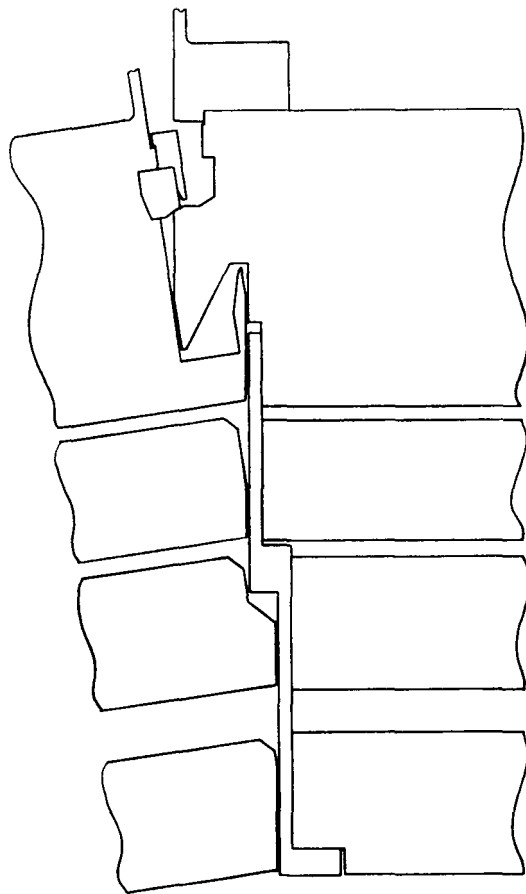
MAXIMUM (3000 PSI) DEFLECTION



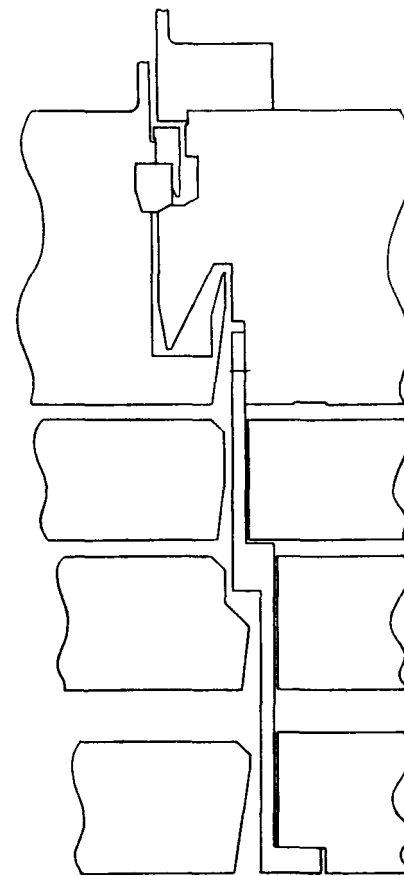
ZERO DEFLECTION

MACHINING MODIFICATIONS PROPOSED
TO ELIMINATE INTERFERENCES AT
THE NODE 5-6 INTERFACE.

Figure 4-25



MAXIMUM (3000 PSI) DEFLECTION



ZERO DEFLECTION

MACHINING MODIFICATIONS PROPOSED
TO ELIMINATE INTERFERENCES AT THE
NODE 9-10 INTERFACE

Figure 4-26

ring are shown in Figures 4-27 and 4-28. The energy absorption capability of the head for these configurations is very slightly less than that associated with the configurations of Figures 4-23 and 4-24 due to a slight reduction in head deflection, ($\Delta E < 3\%$).

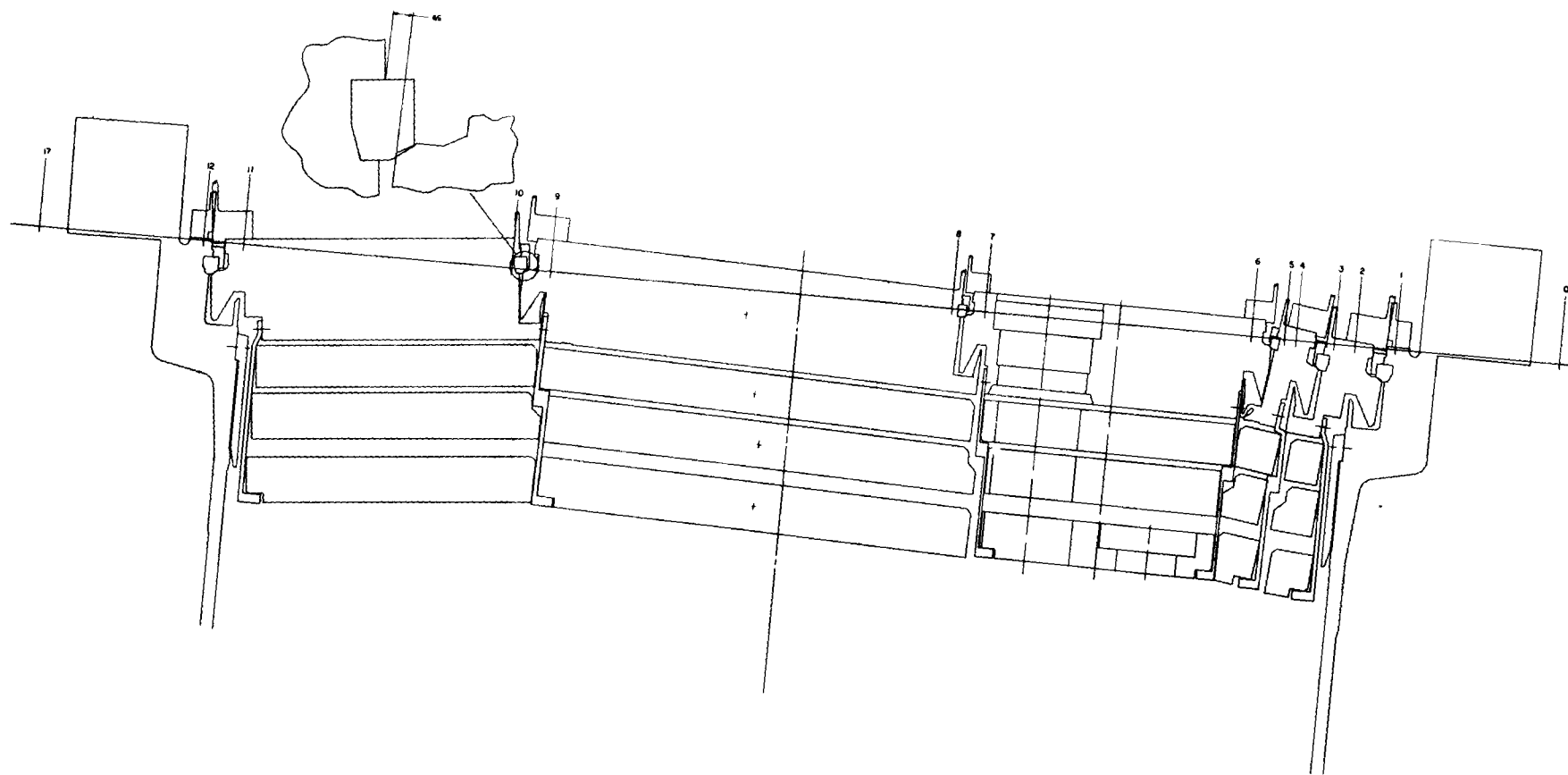
4.3.1.2.2 Limiting Shear Ring-Plug Geometry

Figure 4-19 shows the model plugs in the configuration corresponding to the failure pressure of 2010 psi. The IRP is shown at the extremes of lateral travel permitted by the inside diameter of the LRP. In one extreme of the travel the IRP ledge remains engaged with the margin shear ring, while at the other extreme no engagement remains. Visual examination of the model test pieces following the test show that they were not damaged in any significant way. This evidence suggests that the geometric condition existing in the model at the point of failure was that associated with lateral movement of the plugs such that complete disengagement had taken place. Based on this interpretation of the test data, geometric separation of the full scale plugs is considered acceptable provided that the remaining bearing and shear area is sufficient to carry the load of the plugs.

4.3.1.2.3 Limiting Pressure Retention Capability of the Modified Head

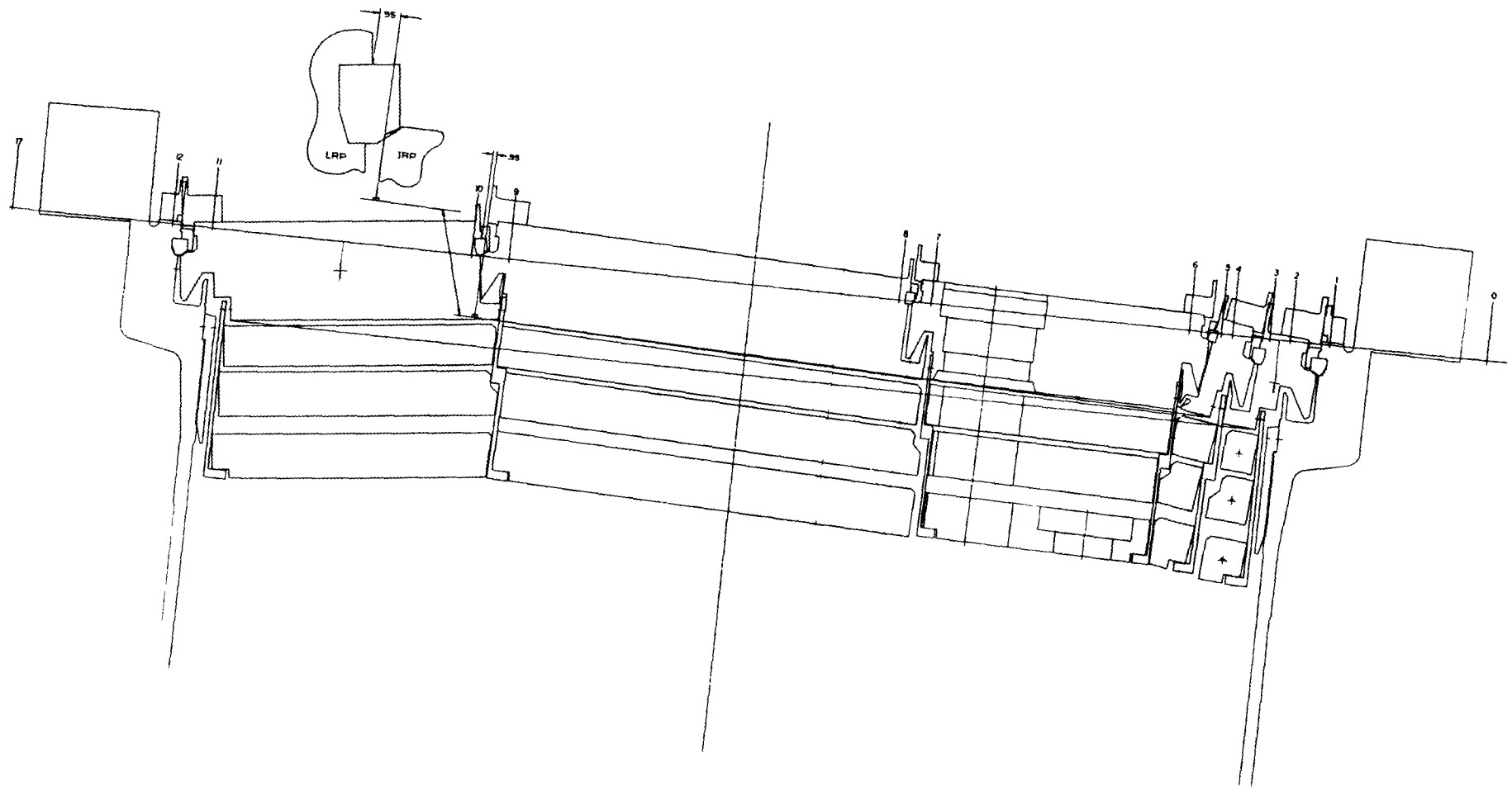
Analyses were performed to determine the limiting pressure retention capability of the closure head once modifications to eliminate the plug disengagement problem were in place. The shear joints between the plugs were identified as the limiting design features. Maximum shear loads occur in the LRP to reactor vessel shear joint. Limiting pressure retention capability of the closure head was therefore based on the shear strength of this joint.

The limiting shear strength of the LRP to reactor vessel joint was derived using the collapse load stress limits of Table F.1322.2-1 of Reference 11 coupled with materials data taken from the individual component archive test results. A summary of the acceptance criteria and materials test data is provided in Figure 4-29.



PROJECTED CLOSURE HEAD DEFLECTIONS AT
2700 PSI WITH PROTOTYPE RADIAL RESTRAINT
AT THE LRP-REACTOR VESSEL INTERFACE

Figure 4-27



PROJECTED CLOSURE HEAD DEFLECTIONS AT
3000 PSI WITH PROTOTYPE RADIAL RESTRAINT
AT THE LRP-REACTOR VESSEL INTERFACE

Figure 4-28

- The Limiting Load Is Based On Average Shear Stress At 400°F
- Allowable Shear Stress = 0.6 Allowable Tensile Strength
(ASME Sec. III, B – 3227.2 Pure Shear)
- Allowable Tensile Strength = 0.9 S_u
0.9 Is A Factor On The Collapse Load Based On The ASME Sec. III
Appendix F For Faulted Conditions
- Material Data At Critical Region:

COMPONENT	MATERIAL SPEC.	R.T. (1) UTS (KSI)	400°F ESTIMATED UTS (KSI)	DATA SOURCE
R.V. TOP FLANGE	SA-508 C1.2	88.2	78.5	P.O. CCU-239552-B LADISH CO. WE004A
LRP MARGIN RING	SA-540 B-24, C1.1	163.5	153.0	PRIORITY 4 DATA PACKAGE FAG AR-54-7CCN-239588-35
LRP	SA-508 C1.2	83.0	73.87	CHICAGO BRIDGE & IRON DATA PACKAGE P.O. 54-CCW-212567
OUTER RISER BASE FLANGE STUD	SA-540 B-23 C1.1	168.4	168.8	B&W PRIORITY 4 DATA PACKAGE P.O. 54-7CCA-237970

(1) Ultimate Tensile Strength Is The Lowest Of The Test Data Of Each Material

Figure 4-29
Acceptance Criteria

The three potential shear planes in the LRP to reactor vessel shear joint are shown in Figure 4-30. The weakest of these shear planes is in the reactor vessel flange where flange shearout would limit the allowable pressure on the closure head to 2507 psi. This shear joint however is backed up by the outer riser bolted flange, which must also fail if the overall joint is to fail. The riser flange strength is limited by the strength of its attachment studs. Analysis summarized in Figure 4-31 shows that an additional 248 psi would be required to fail the flange studs. The allowable load on the LRP to reactor vessel joint therefore is obtained as $2507 + 248 = 2755$ psi.

The acceptance criteria adopted in the analysis of the shear joint strength limit the allowable load to 90% of that which would cause failure. The predicted failure load would therefore be $2755/0.9 = 3060$ psi. Layouts of the head deformed configuration were therefore made for pressures of 2700 and 3000 psi.

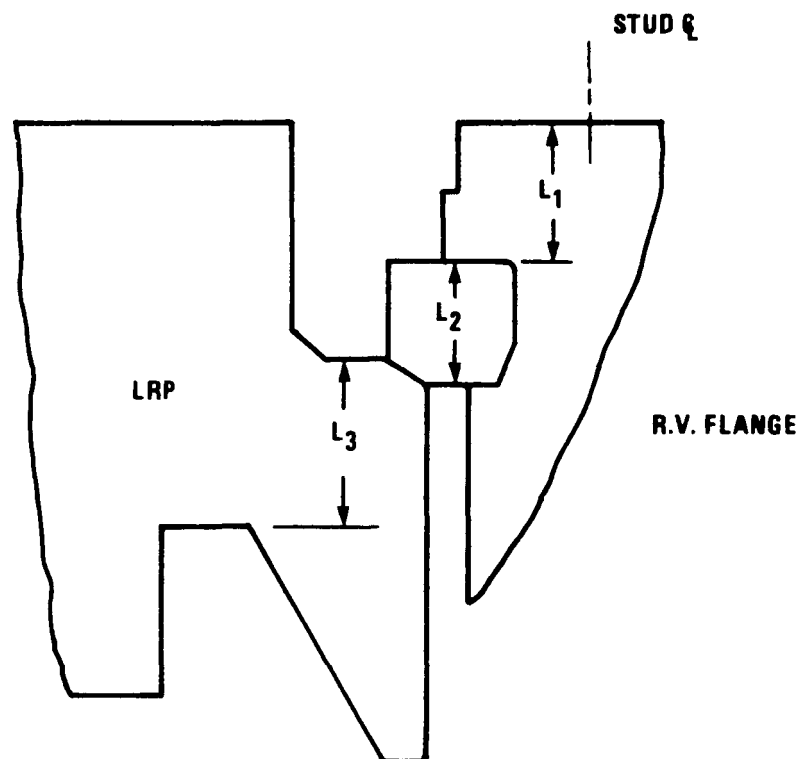
4.3.1.2.4 Limiting Plug Gap Reductions

One of the parameters considered for modification was the radial gap between the plug and the margin ring keeper ring. Closing this gap would limit the ability of the IRP to move laterally within the LRP, and thereby contribute to the delay of shear joint disengagement. The present definition of the design modifications does not include the reduction in this gap. The summary of the evaluations performed to determine the minimum acceptable gap however is provided in Figures 4-32 to 4-35. The evaluations showed that a minimum gap of 1/8 of an inch would permit free running of the plugs during normal operation and would have no impact on the seismic response of the closure head.

4.3.1.2.5 Plug Strains

An evaluation of the plug strains is important to

- A. Verify that the plug is operating in the stable region of the post yield stress-strain curve and,



LOCATION	COMPONENT	FORCE (LB/IN CIRC.)	HEIGHT (IN)	AVERAGE SHEAR STR. (PSI)	ALLOWABLE SHEAR STR. (PSI)	ALLOWABLE PRESSURE (PSI)
VESSEL/ TO LRP INTERFACE	R.V. FLANGE	64.25P	$L_1 = 3.80$	16.91 P	42390	2507
	MARGIN RING		$L_2 = 3.50$	18.36 P	82620	4500
	LRP		$L_3 = 4.60$	13.97 P	30890	2855

P = PRESSURE

THE AVERAGE SHEAR STRESSES IN THE INTERMEDIATE AND SMALL SHEAR JOINTS ARE LESS THAN THE ABOVE VALUES.

TOTAL ALLOWABLE PRESSURE = VESSEL FLANGE CAPACITY + FLANGE STUDS CAPACITY
 $= 2507 + 248 = 2755 \text{ PSI}$

Figure 4-30

Shear Joint Strength

89 Studs 1.250 – 8 UN – 2A (Shank Area = 1 in²)

S_w At 400°F = 160800 psi

**Allowable Strength = 0.9 S_u = 144720 psi
(0.9 Factor Based On Collapse Load of ASME Sec. III Appendix F)**

Total Area = 1 x 89 = 89 in²

Load Capacity = 144720 x 89 = 1.288 x 10⁷ lbs

Diameter of LRP = 257 in

$$\text{Allowable Pressure} = \frac{1.288 \times 10^7}{\frac{\pi}{4} (257)^2} = 248 \text{ psi}$$

Figure 4-31

Flange Studs Capacity

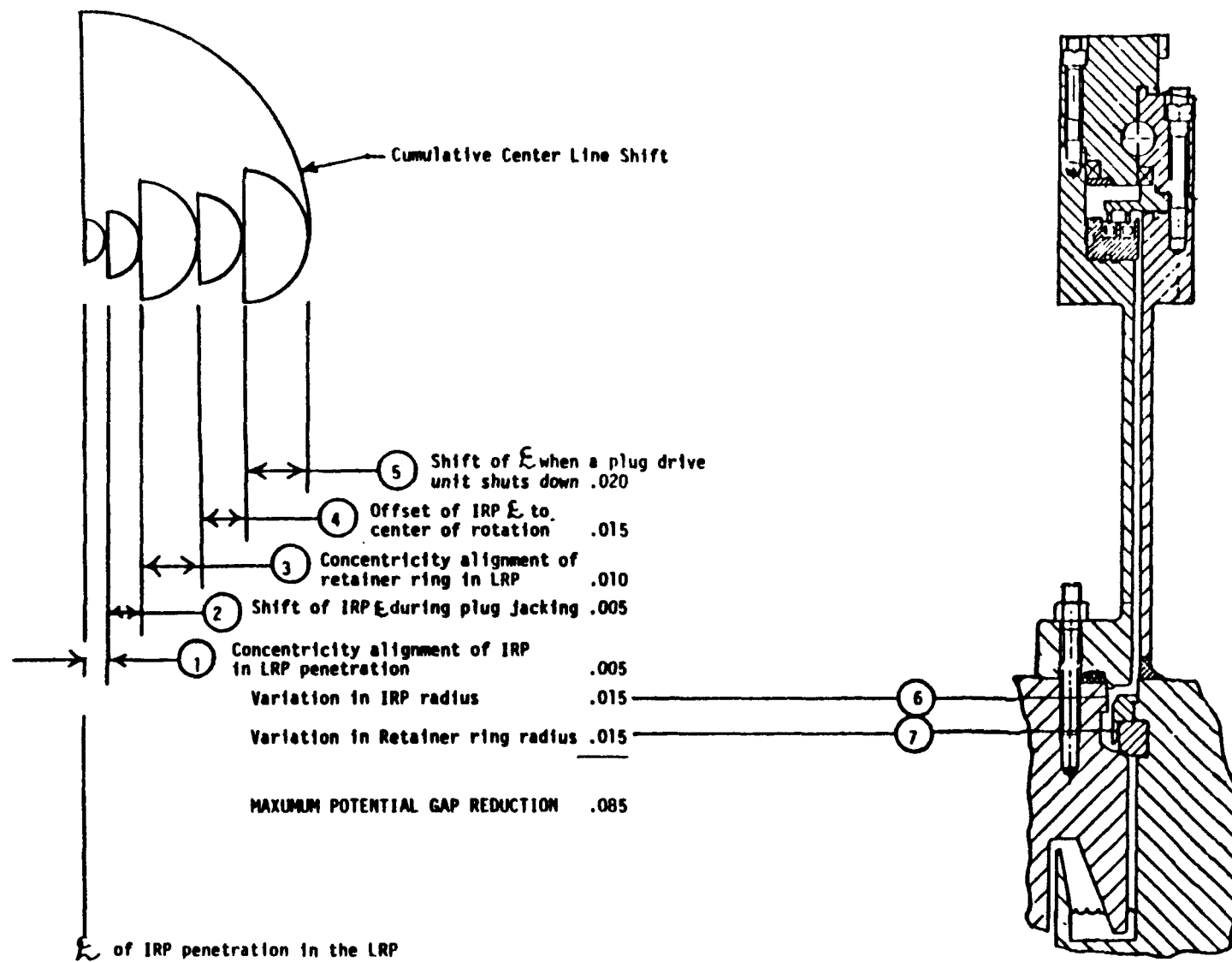


Figure 4-32

Potential Gap Reduction Factors

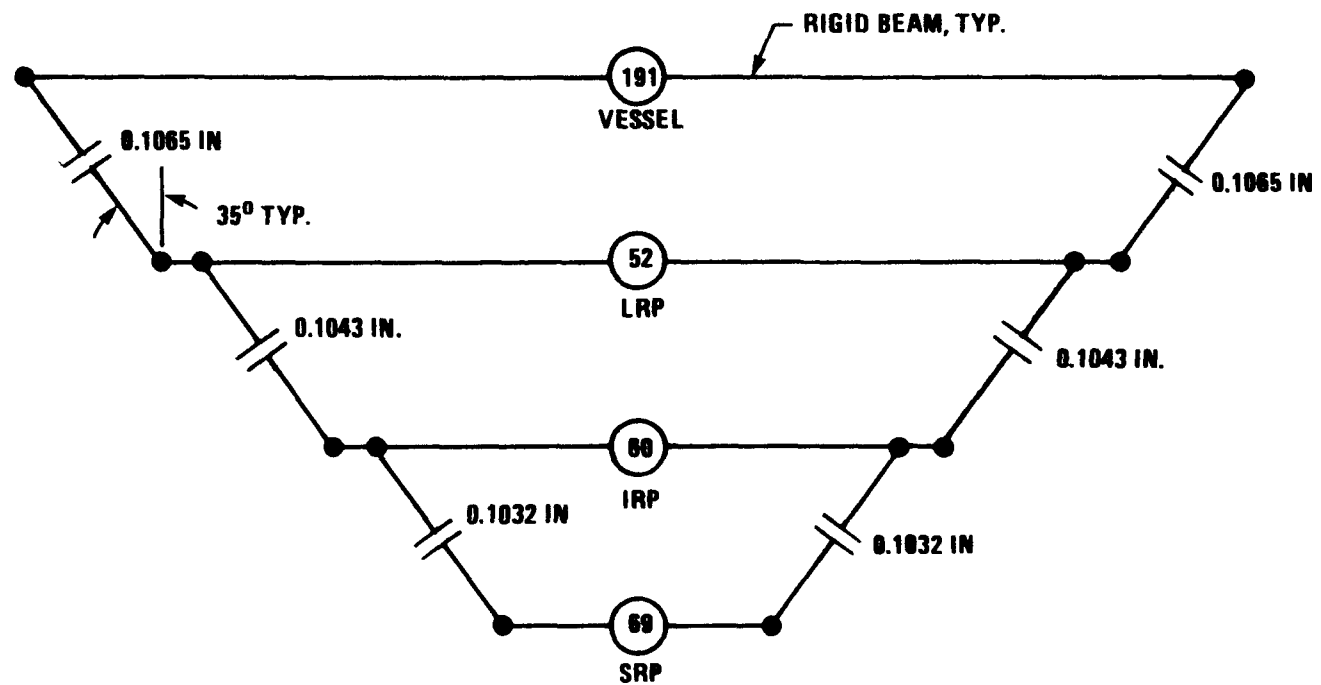


Figure 4-33

Seismic Model: From Calculation No. LRA-SA-78-77

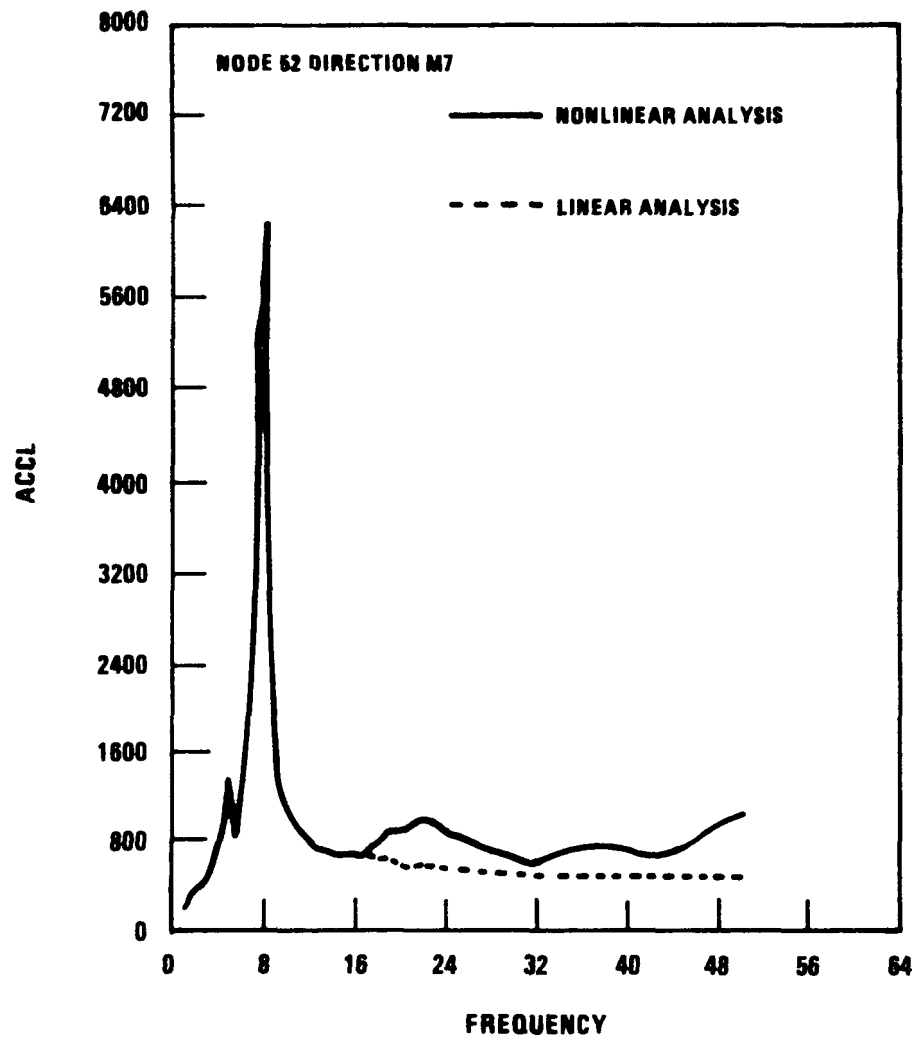


Figure 4-34

ACCELERATION VS. FREQUENCY FOR LINEAR
AND NON-LINEAR ANALYSES

- **Reactor System Seismic Model Is Linear**
- **Effect Of Inter-Plug Gaps Was Evaluated In A Separate Analysis**
- **The Results Show That There Is No Significant Difference In The Response Spectra Peaks**
- **The LRP ZPA Increased, However This Would Be Reduced With Actual Plug Distributed Mass And Stiffness**
- **It Can Be Concluded That Gap Modification Will Have An Insignificant Effect On The Seismic Response**

Figure 4-35

Head Seismic Response With Reduced Radial Gaps Between Plugs

- B. Verify that the plug strains are not approaching a level at which failure due to strain exhaustion is possible.

An evaluation of the plug strains is provided in Figure 4-36. Geometric data required to determine the strain levels in the main body of the plugs were obtained from the head deformed layout drawings. Maximum strains occur at the free edges of the plugs and are therefore uniaxial in nature. The maximum strain was found to be 1.2%. The uniaxial uniform elongation for the head structural material at the operating temperature of 400°F is 7%. This strain level is equated to the plastic instability strain of Reference 11. The allowable strain is limited to 0.7 of the plastic instability strain, and is therefore $.7 \times 7.0 = 4.9\%$. The maximum strain in the body of the plug is therefore seen to be approximately 25% of the allowable strain. The load deflection characteristics of the plugs at this relatively low post yield strain level will be stable.

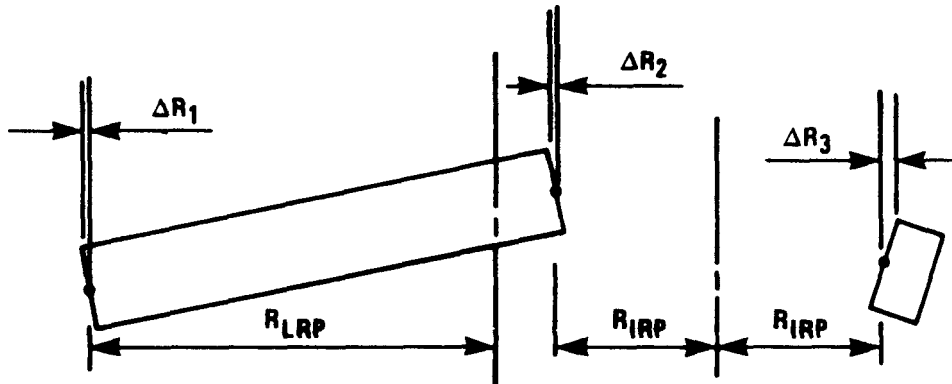
It is to be expected that some of the highest strains in the head system will be found in the shear joints where the loads are concentrated. The LRP to reactor vessel shear joint was subject to a shear load corresponding to a 2587 psi pressure load on the head during the SM-7 tests. While the stiffness of the plugs in the SM-7 test was not prototypic, representation of the shear joints was prototypic. The SM-7 tests therefore provided a valid proof test on the LRP to reactor vessel shear joint. The joint did not fail during the test. While the test verification pressure of 2587 psi is lower than the predicted limit pressure strength of the head, it is higher than the pressure strength required to demonstrate the capability to absorb the kinetic energy from a 75.1MJ slug. (The required pressure retention capability for a 75.1MJ slug is derived in Section 4 as $2180/0.9 = 2420$ psi).

4.3.1.3 Energy Absorption Performance of the Modified Head

In order to define the slug energy absorption capability of the modified head it is first necessary to define the percentage of the slug energy expended in performing work on the closure head. The kinetic energy in the combined slughead mass immediately following the impact is related to the slug kinetic energy immediately prior to impact by the law of conservation of momentum.

3000 PSI

1. Plug Primary Strains



$$\epsilon_{\max} = \Delta R/R = 1.2\% \text{ (Uniaxial)}$$

$$\epsilon_u = 7.0\% \text{ (SA508 CL 2 At } 400^\circ\text{F)}$$

$$0.7 \times \epsilon_u = 4.9\% \text{ (0.9 Factor Based On Strain Limit Load Of ASME Section III Appendix F)}$$

2. Attachment Strains

- The LRP O.D. Attachment Is The Most Highly Loaded Attachment
- The LRP O.D. Attachment Geometry Was The Same For The SM 7 And SM 8 Tests
- The Joint Withstood A 2587 PSI Load In The SM 7 Test With No Failures

Figure 4-36

Plug Strain Summary

Given a slug mass of 0.368×10^6 lb and a head mass of 1.046×10^6 lb, the energy imparted to the head is obtained as 19.5MJ, or approximately 26% of the original slug kinetic energy. A summary of the derivation of these results is provided in Figure 4-37.

In order to have confidence that the energy imparted to the head has been correctly defined, it is necessary to demonstrate that the location of the remaining 74% of the energy in the original 75MJ slug can be satisfactorily defined. In this report, only the energy converted from kinetic energy to potential energy within the sodium slug is investigated as an example of the post impact energy location.

Pressure and flow conditions within the slug in the period immediately following the slug impact can be estimated using conventional waterhammer analysis for flow in pipes with an instantaneous gate closure. The sequence of events in the fluid immediately following gate closure (slug impact) is illustrated in Figure 4-38, extracted from Reference 12. A pressure wave is seen to travel down the length of the fluid slug, leaving the entire slug pressurized at the peak impact pressure at time $t = l/a$ seconds. The impact pressure associated with 75MJ slug in a vessel with no UIS present is obtained from the SM-2 dynamic model test as 5300 psi [7]. The bulk modulus required to calculate the volumetric contraction of the plug, and the potential energy stored in the compressed slug was obtained from Reference 13. The calculation of the potential energy stored in the compressed slug is provided in Figure 4-39. The stored energy is obtained as 39.6MJ or 53% of the original slug kinetic energy content.

The closure head and slug between them thus account for approximately 79% of the slug energy in the period immediately following impact. Energy storage mechanisms such as vessel wall straining, re-directed slug flow and slug turbulence account for the balance of the slug energy. Table 5-2 of Reference 2 shows that much of the slug energy is ultimately deposited in the reactor vessel wall.

M_S = SLUG MASS ($W_S = 0.368 \times 10^6$ LB)

M_H = HEAD MASS ($W_H = 1.046 \times 10^6$ LB)

$M_{H\&S}$ = SLUG MASS + HEAD MASS

V_S = SLUG VELOCITY

$V_{H\&S}$ = VELOCITY OF THE COMBINED SLUG & HEAD MASS

KE_S = SLUG KE AT IMPACT = 75 MJ

$KE_{(H\&S)}$ = UPPER BOUND ESTIMATE OF THE KE DELIVERED TO THE HEAD

FOR CONSERVATION OF MOMENTUM: -

$$V_{M\&S} = V_S \left(\frac{M_S}{M_S + M_H} \right) \quad (1)$$

$$KE_S = \left(\frac{M_S V_S^2}{2} \right) \quad (2)$$

$$KE_{H\&S} = \left(\frac{M_S + M_H}{2} \right) \times V_S^2 \left(\frac{M_S}{M_S + M_H} \right)^2$$

$$KE_{H\&S} = \frac{V_S^2 M_S^2}{2(M_S + M_H)} = KE_S \left(\frac{M_S}{M_S + M_H} \right) \quad (3)$$

$$KE_{H\&S} = \frac{75 \times 0.368}{(1.046 + 0.368)} = 19.5 \text{ MJ}$$

$$KE_{H\&S} = 19.5 \times 8.85 \times 10^6 = 173 \times 10^6 \text{ LB.IN.}$$

Figure 4-37
Head Energy Input

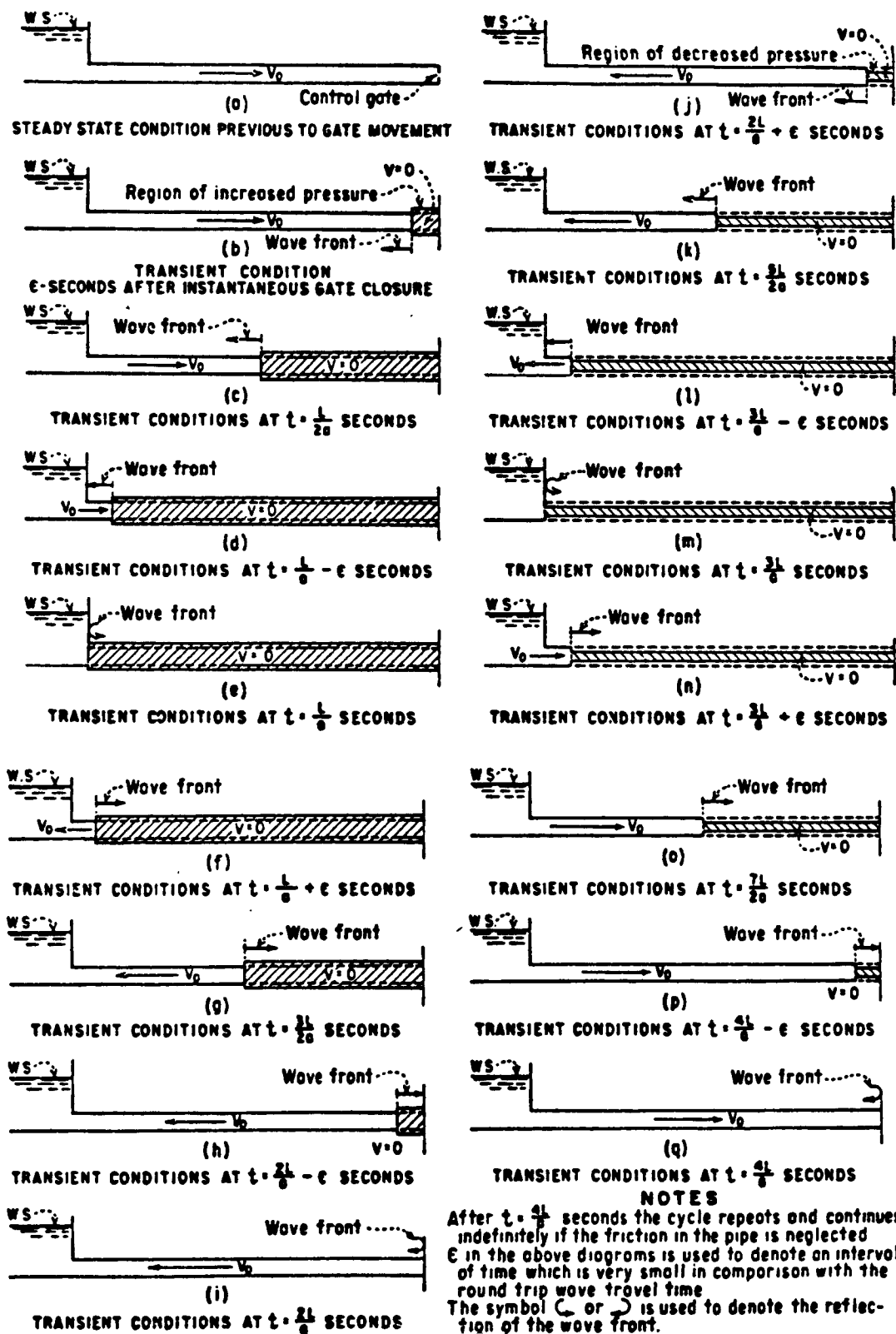


Figure 4-38
Propagation of Pressure Waves Caused by an Instantaneous Complete Gate Closure

1. DATA

SLUG WEIGHT = 0.368×10^6 lb

SLUG TEMPERATURE = 1000°F

SLUG DENSITY = $51 \text{ lb/ft}^3 = 0.029 \text{ lb/in}^3$

SODIUM COMPRESSIBILITY = $0.29/\text{GP}_a$ (SMH)

$1 \text{ GP}_a = 0.145 \times 10^6 \text{ lb/in}^2$

MAXIMUM INTERFACE PRESSURE AT IMPACT = 5300 PSI

2. ENERGY CONTENT ESTIMATE

THE "WATERHAMMER" PRESSURE THROUGHOUT THE LENGTH OF THE SLUG AT TIME $t = \frac{L}{a}$ SECONDS AFTER THE IMPACT WILL BE A UNIFORM 5300 PSI (Reference WATERHAMMER ANALYSIS, John Parmakian)

$$\text{COMPRESSION ENERGY STORED} = \frac{\Delta \cdot P}{2}$$

$$PE_s = \frac{V \cdot P^2}{2K}$$

$$K = \text{BULK MODULUS} = \frac{0.145 \times 10^6}{0.29} = 0.5 \times 10^6 \text{ lb/in}^2$$

$$\text{VOLUME} = \frac{0.368 \times 10^6 \times 12^3}{51} = 12.5 \times 10^6 \text{ in}^3$$

$$\therefore PE_s = \frac{12.5 \times 10^6 \times 5300^2}{2 \times 0.5 \times 10^6} = 351 \times 10^6 \text{ lb.in}$$

$$1 \text{ MJ} = 8.85 \times 10^6 \text{ lb.in}$$

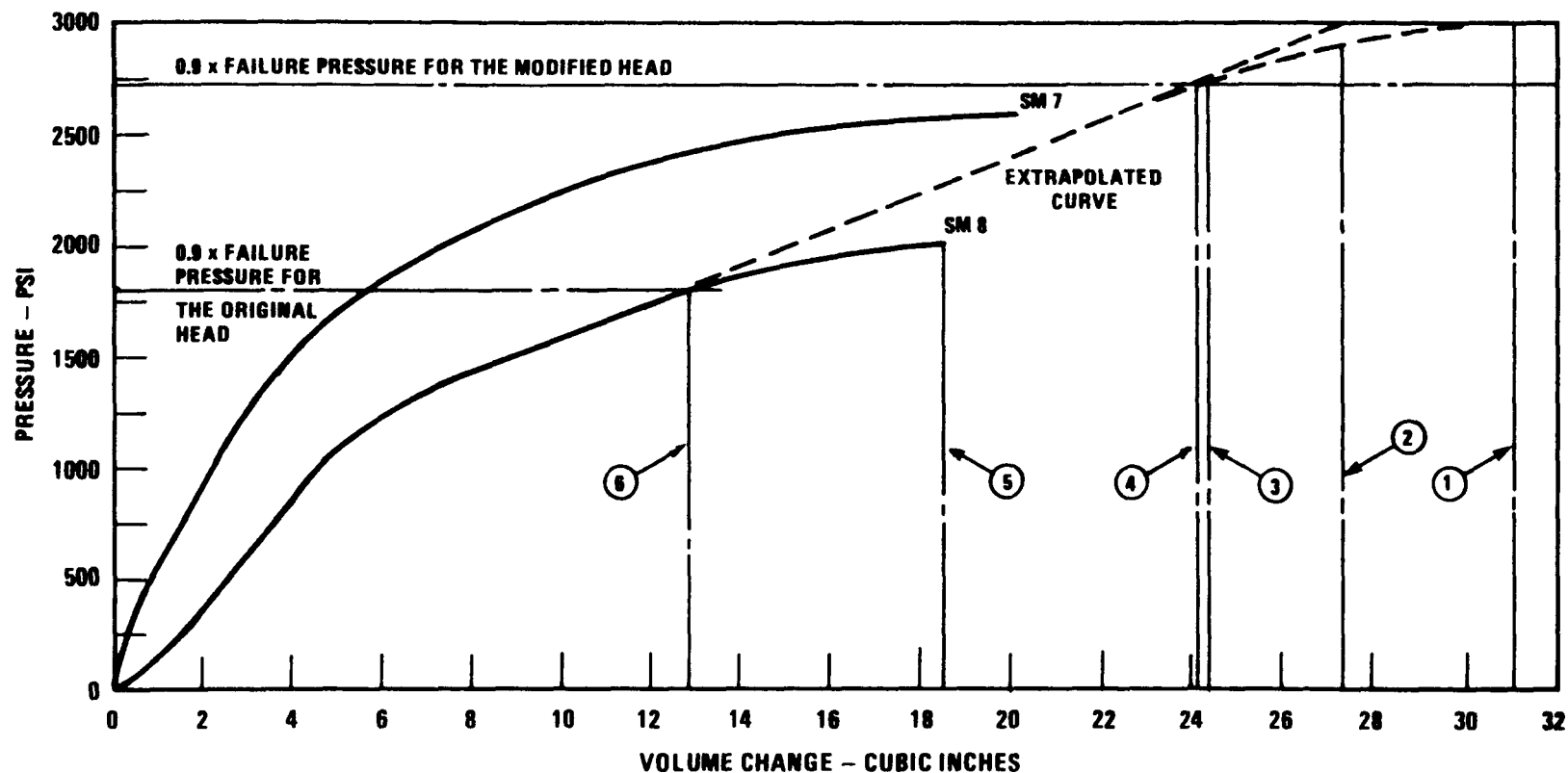
$$\therefore PE_s = \frac{351 \times 10^6}{8.85 \times 10^6} = 39.6 \text{ MJ}$$

Figure 4-39

Slug Bulk Compression Energy

Work performed in deflecting the SM-8 test head up to the configuration corresponding to its 2010 psi failure pressure can be obtained by integrating the area under the pressure-volume change curve given in Reference 9. The curve from Reference 9 is reproduced in Figure 4-24 of this report. The integral under the SM-8 curve up to the point of failure gives an energy absorption capability of 200×10^6 in-lb. The acceptance criteria of Reference 11, however, limit the maximum permissible loads to 90% of the failure loads. The indicated maximum usable energy absorption capability of the existing head is therefore limited to that associated with a head pressure of $0.9 \times 2010 = 1800$ psi. Integration of the test pressure-volume change curve over the pressure range zero to 1800 psi yields a usable head work value of 94×10^6 lb-in. Using the value derived in Figure 4-37 for the energy deposited in the head by a 75MJ slug (173×10^6 lb-in), the predicted slug energy capability of the existing closure head design is obtained as 40.8MJ.

It was shown in Section 4.3.1.2.3 of this report that the failure pressure for the modified closure head is approximately 3000 psi. With this failure pressure, the allowable maximum pressure on the closure head is $0.9 \times 3000 = 2700$ psi. The pressure-volume curve from the SM-8 test can be extrapolated to the higher failure pressure provided the mode of plug motion in the disengagement phase of the head loading is known. If the plugs start to disengage, the pressure-volume change curve flattens as failure is approached, giving an increase in the total energy absorbed for a given maximum pressure load on the head. With no plug disengagement, a straight line extrapolation of the pressure-volume change curve is obtained. This mode of extrapolation results in a lower bound estimate of the energy absorption capability of the closure head for a given maximum pressure loading. A comparison of the model plug shear ring configuration at failure (Figure 4-18), with the shear ring configuration at 3000 psi loading in Figures 4-24 and 4-28 indicates that there should be little non-linear plug disengagement motion at 3000 psi loading. Information on the energy absorption capacity at failure with non-linear plug disengagement is useful however in that it indicates the energy absorption margin available should unanticipated non-linear plug disengagement effects be encountered. Both linear and non-linear extrapolation curves are provided in Figure 4-40. Note that while the volume change at failure for these curves differ significantly



- ① VOLUME CHANGE FOR THE MODIFIED HEAD WITH NON LINEAR DISENGAGEMENT (NLD)
- ② VOLUME CHANGE FOR THE MODIFIED HEAD WITH NO NON LINEAR DISENGAGEMENT
- ③ USEABLE VOLUME CHANGE FOR THE MODIFIED HEAD WITH NLD
- ④ USEABLE VOLUME CHANGE FOR THE MODIFIED HEAD WITH NO NLD
- ⑤ VOLUME CHANGE FOR THE ORIGINAL HEAD WITH NLD
- ⑥ USEABLE VOLUME CHANGE FOR THE ORIGINAL HEAD

Figure 4-40

Pressure-Volume Change for Original & Modified Head Designs

(points 1 and 2), the curves yield similar results at the maximum pressure permitted by the acceptance criteria (points 3 and 4).

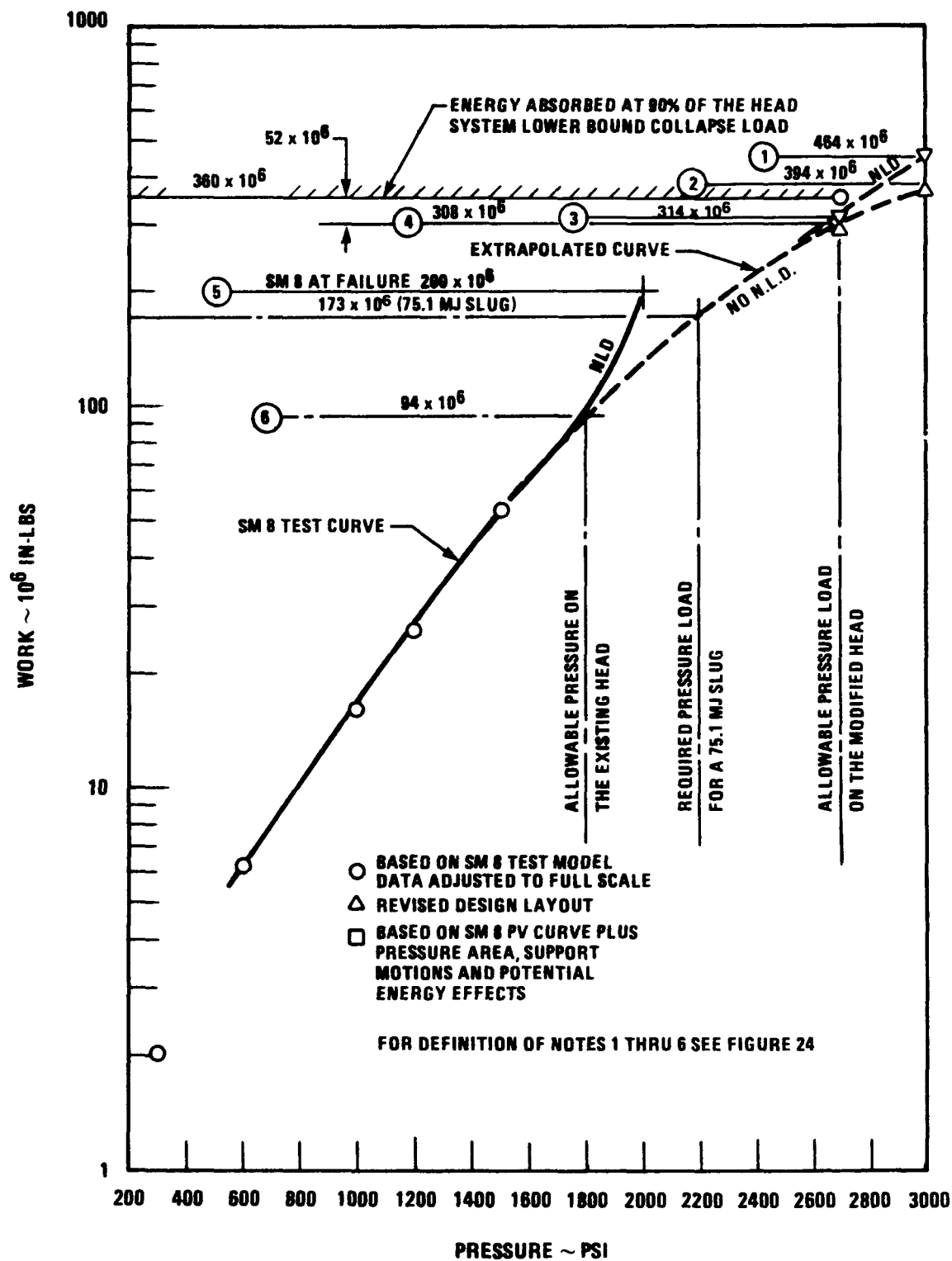
Integrating the area under the pressure-volume change curve to points 1 and 2 gives estimates of the energy absorbed at failure of 464×10^6 lb-in (201MJ) for the case of non-linear disengagement (NLD) and 394×10^6 lb-in (171MJ) for the case of no non-linear disengagement. Comparing these results with the test result for the original head configuration, the energy absorbed at failure is seen to have been increased by 132% for the NLD mode and 97% for the no NLD mode.

At the limiting collapse load pressure ($0.9 \times$ failure pressure) the energy absorbed by the modified head in the NLD mode is 314×10^6 lb-in (136MJ) (point 3) and 308×10^6 lb-in (134MJ) (point 4) in the no NLD mode. This compares with a predicted energy absorption of 94×10^6 lb-in (40.8MJ) for the existing head design at the limiting collapse load pressure of 1800 psi (point 6). The head modifications have therefore increased the head energy absorption capacity at the limiting collapse load pressure by 234% and 227% for the NLD mode and no NLD modes respectively.

Work absorbed by the closure head as derived from an extrapolation of the SM-8 model test curves is summarized in Figure 4-41.

A number of small additional sources of slug energy absorption exist in the reactor head and support system. Lifting the mass of the head as the head deforms upwards contributes 5×10^6 lb-in of potential energy. Dynamic response of vessel support system (vessel flange, attachment bolts and the steel ledge) due to out of balance SMBDB inertia loads produces a 0.43 inch upward motion of the head support. This motion consumes 11×10^6 in-lb of strain energy. The analytical model, loads, and vessel flange response used to derive support system energy absorption are shown in Figures 4-43 through 4-46.

A final correction is required to account for differences between the pressure area used in the SM-8 model and that existing in the actual closure head under SMBDB pressure loading. The pressure seal bladder is shown in Figure 4-17 to



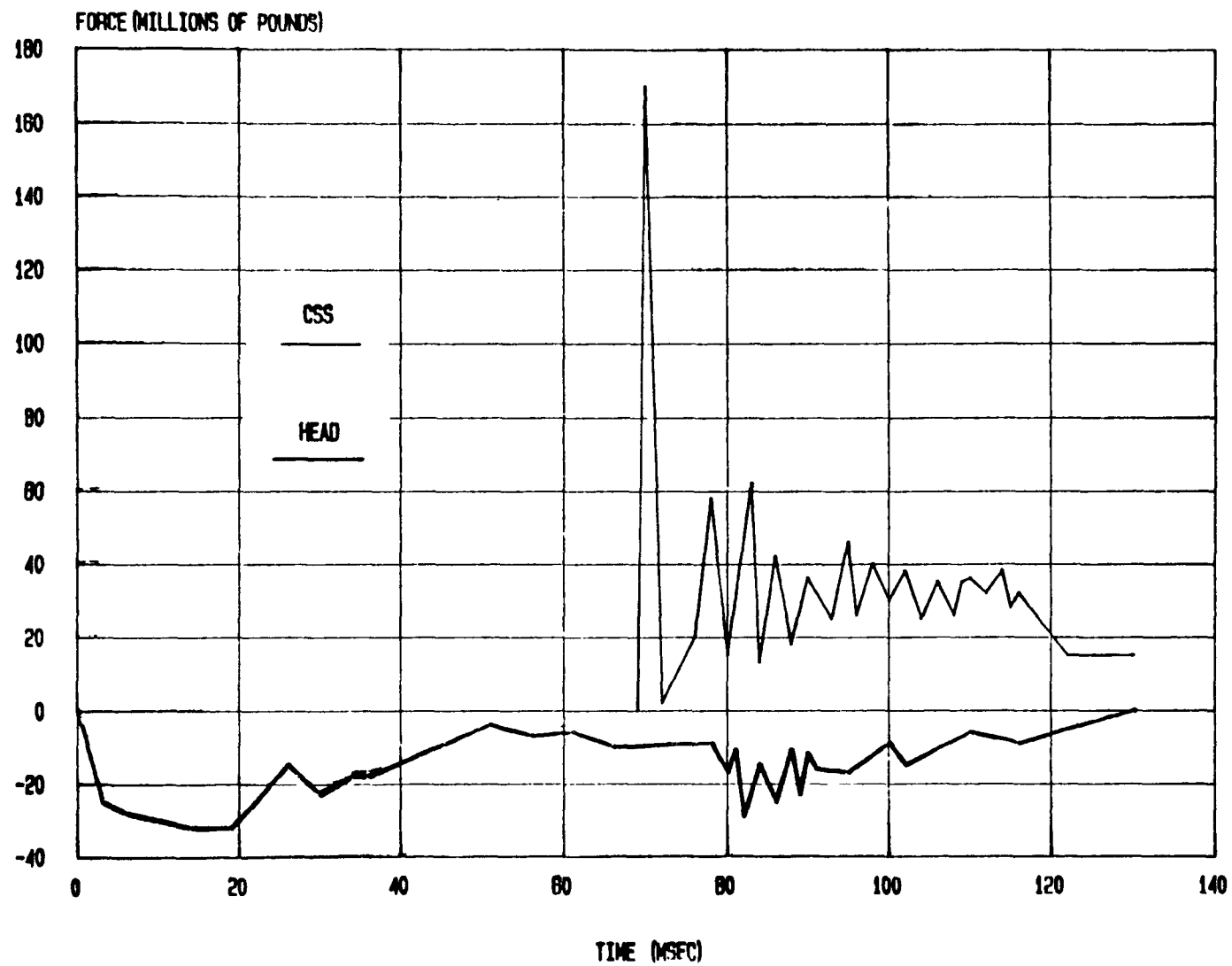


Figure 4-43
Force Time History
Head Up/Force CSS Down/Force

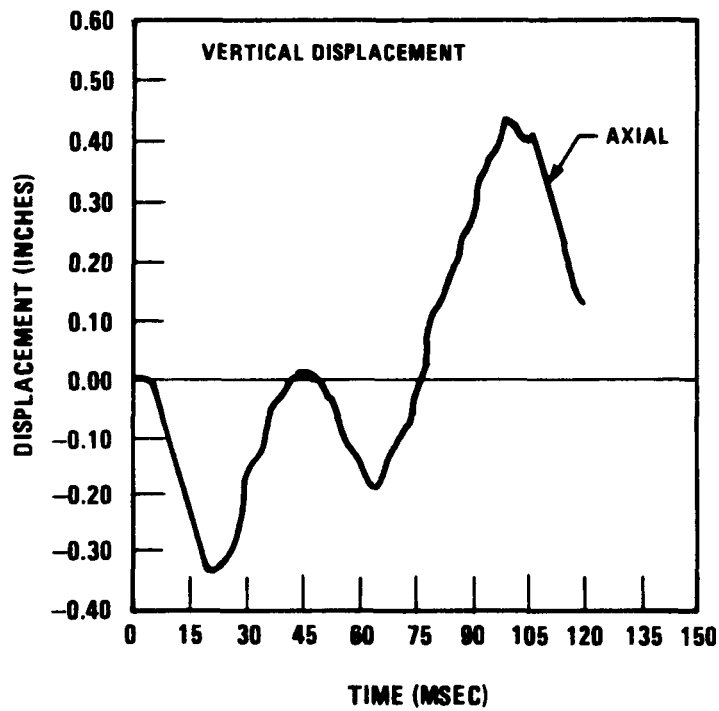
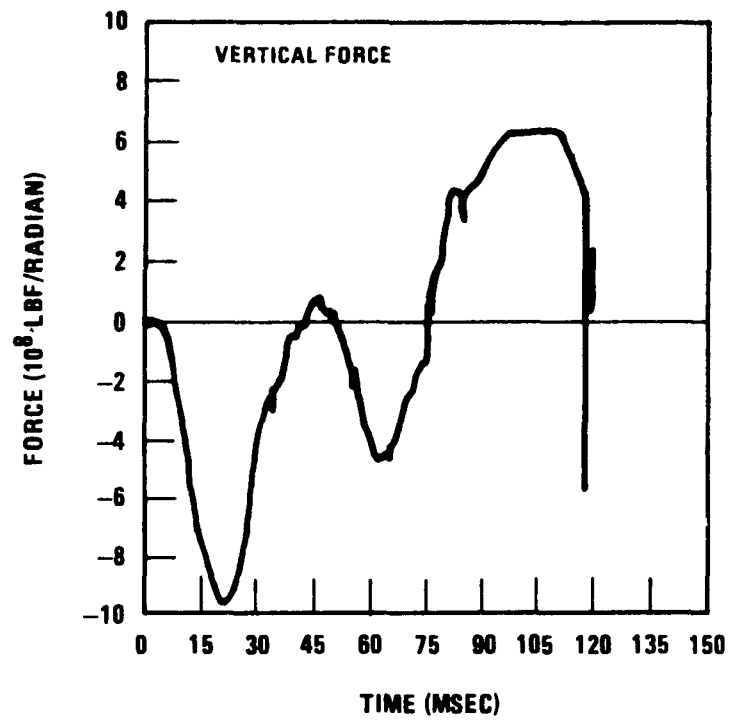


Figure 4-44

Vessel Flange SMBDB Response

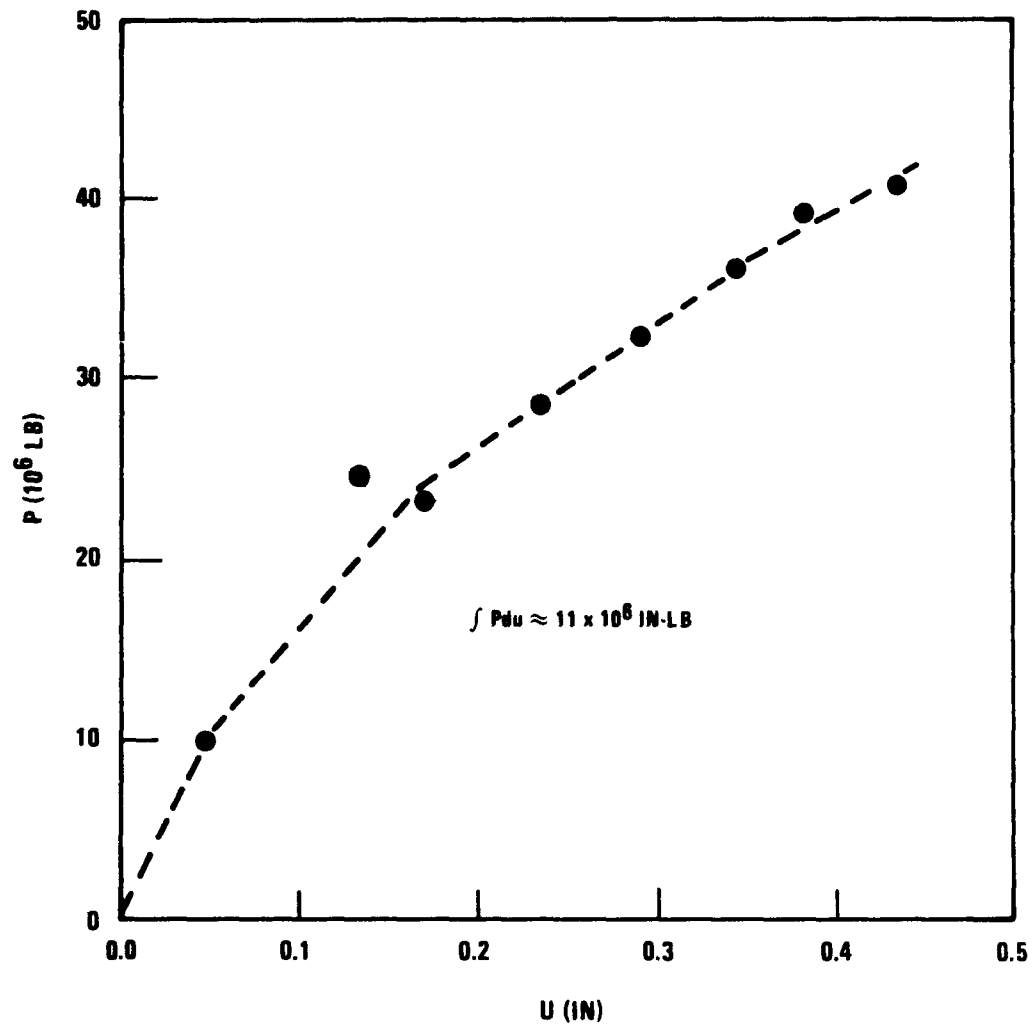


Figure 4-45

Reactor Vessel Support & Ledge Dynamic Force-Deflection Relationship (75 MJ Slug Load)

HEAD CONFIGURATION	LOADING CONDITION	HEAD PRESSURE PSI	ENERGY ABSORPTION FROM THE EXTRAPOLATED TEST P-V CURVE LB-IN	ADDITIONAL HEAD SYSTEM ENERGY CAPACITY (1) LB-IN	TOTAL HEAD SYSTEM ENERGY CAPACITY LB-IN
EXISTING HEAD	COLLAPSE LOAD	2010	130×10^6	~	~
EXISTING HEAD	90% COLLAPSE LOAD	1800	94×10^6	~	$\geq 94 \times 10^6$
MODIFIED HEAD (2)	COLLAPSE LOAD	3000	394×10^6	~	~
MODIFIED HEAD (2)	90% COLLAPSE LOAD	2700	308×10^6	$\geq 52 \times 10^6$ (2)	360×10^6

THE MODIFIED HEAD ENERGY ABSORPTION CAPABILITY
CORRESPONDS TO A SLUG ENERGY OF 156 MJ

- NOTES:
1. THE ADDITIONAL HEAD SYSTEM ENERGY ABSORPTION CAPACITY DERIVES FROM (a) HEAD SYSTEM POTENTIAL ENERGY (b) HEAD SUPPORT SYSTEM STRAIN ENERGY AND (c) ADJUSTMENTS TO ACCOUNT A SMALL NON PROTO-TYPIC REPRESENTATION OF THE HEAD PRESSURE AREA IN THE SM 8 MODEL
 2. THE ADDITIONAL HEAD SYSTEM ENERGY ABSORPTION CAPACITY WAS ESTIMATED FOR THE MODIFIED HEAD ONLY. SMALLER VALUES WOULD APPLY TO THE EXISTING HEAD DESIGN

Figure 4-46

Closure Head System Energy Absorption Summary

extend to a model radius of the vessel ID. In the closure head however the SMBDB pressures would extend outwards to the LRP margin shear ring. Extending the pressure out to the radius and multiplying the resulting load by the small deflection of this area under 2700 psi loading results in an addition of 38×10^6 lb in to the head work integral.

A summary of the head system energy absorption capability is provided in Figure 4-46.

4.3.1.4 Evaluation

Entering the head energy absorption curve of Figure 4-41 with the energy deposited in the head by a 75MJ slug, the required pressure load capacity of the head was obtained as 2180 psi. The allowable pressure load on the modified head had been calculated to be 2700 psi. The reserve factor on head strength (RF) was therefore obtained as follows.

$$RF_s = \frac{2700}{2180} = 1.24$$

The lower bound energy absorption capacity of the modified head system was obtained from Figure 4-29 as 360×10^6 lb-in. The energy deposited in the head system by a 75MJ slug was obtained from Figure 4-20 as 173×10^6 lb-in. The reserve factor on energy (RFE) absorption at the maximum allowable pressure for the modified head was therefore:

$$RF_E = \frac{360 \times 10^6}{173 \times 10^6} = 2.08$$

The controlled parameter was the energy content in the slug at the instant of contact with the head (75MJ). The reserve factor of 2.08 obtained for the energy absorption capacity of the modified head was considered more than adequate to cover any uncertainties in the evaluation procedure.

The reserve factor on head strength was considered adequate at 1.24. An important consideration in arriving at this conclusion was the demonstration

in SM-7 test of a shear strength capability well in excess of that required for the absorption of a 75MJ slug (2580 psi in the test vs a required pressure of 2180 psi).

Experience had shown that the most satisfactory means of assessing the SMBDB energy absorption capacity of the closure head was to test the closure head. A hydrostatic test of the modified head geometry was planned to confirm that all interferences which contributed to the kinematic failure mode had been satisfactorily eliminated. This was to be followed by a confirmatory dynamic test using the scale equivalent of a 75MJ slug.

4.4 Final System Status

Each closure head plug assembly was completed in its individual assembly stand. Then they were assembled to form the complete closure head assembly. The assembly was then installed in the test and shipping container.

The bolted risers, the seals, and the bull gear and bearings were previously delivered to CBI from other suppliers. These components were assembled to the plug drive and control system.

The status of the Development Program for HCDA Load Simulation is shown in Table 4-3.

4.4.1 Outstanding Changes - HCDA Closure Head Modification

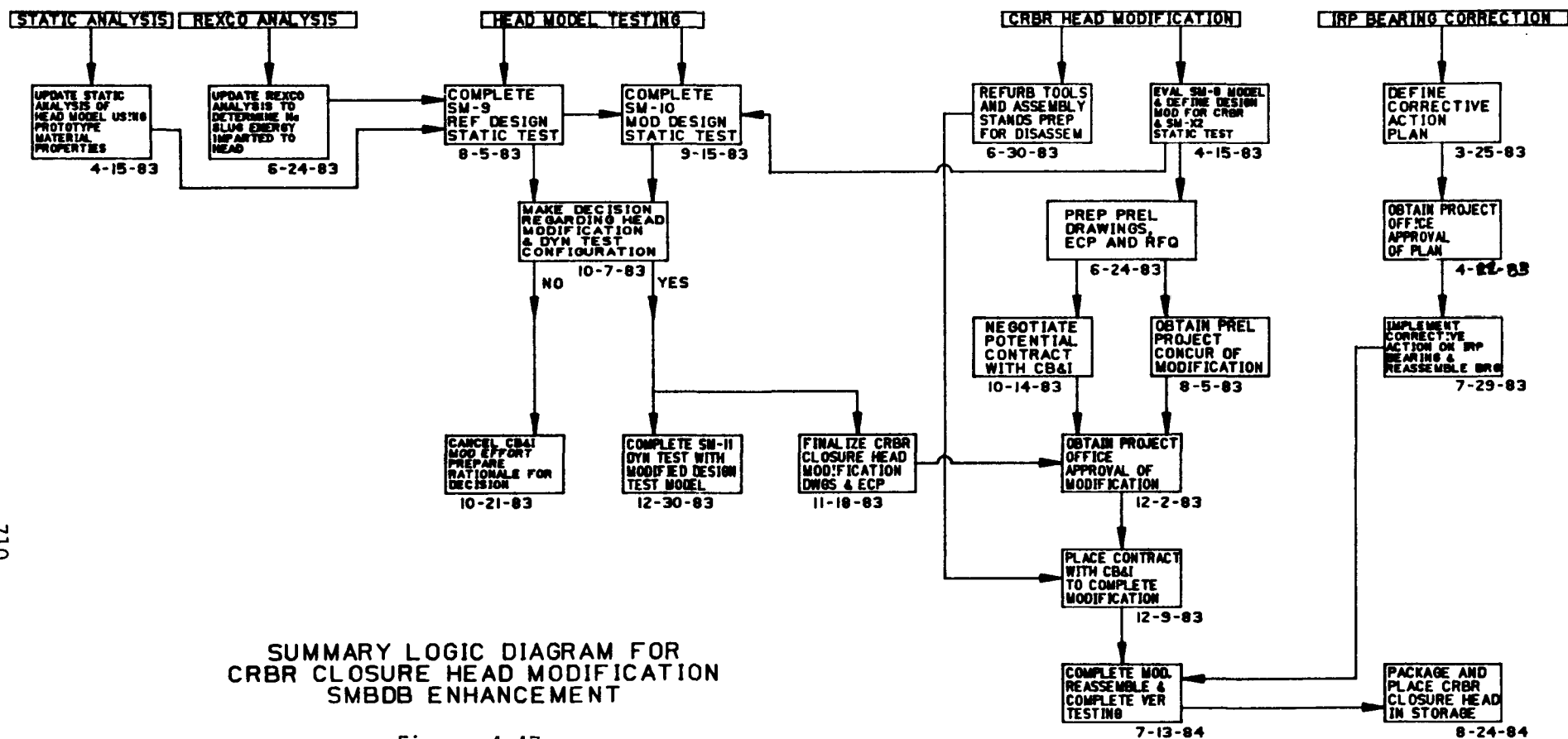
Draft ECP W-2133 [14] defined the machining required to eliminate the kinematic interferences created by doming of the plugs under SMBDB loading without degrading the function of the dip seal. The recommended machining modifications are shown on Figures 4-25 and 4-26.

4.4.2 Outstanding Activities to Complete Design

Figure 4-47 is a "Summary Logic Diagram for CRBRP Closure Head Modification, SMBDB Enhancement" as proposed in April, 1983. Between April and November, 1983 (CRBRP cancellation) the SM-10 (modified design) test was

TABLE 4-3
STATUS OF DEVELOPMENT PROGRAM FOR HCDA LOAD SIMULATION

Complete Phase 0 (Materials Test)	Completed
Complete Phase I (Source Calibration)	Completed
Data from SM-4	Completed
Assess the need and scope for complex model and high energetics experiments	Completed
Data from Phase II (Simple Model Tests)	Completed
Final Report on Phase 1 and 2 (SM-1 through SM-5)	Completed
SM-6	Not Completed
Initiate SM-7 preparation and testing	Completed
Initiate SM-8 hardware procurement	Completed
Complete SM-8 test	Completed
Preliminary data available	Completed
Final report on SM-7 and SM-8	Completed
Provide SM-9 model drawings to SRI	Completed
Provide SM-10 model drawings to SRI	Completed
Complete SM-10 Model Design	Completed
Complete SM-9 static test and submit interpretive report	Not Completed
Complete SM-10 static test	Completed
Submit interpretive report on SM-10	Not Completed
Complete static test report	Not Completed
Complete SM-11 dynamic test	Not Completed
Complete final report on SM-9, SM-10 and SM-11	Not Completed
Prepare drawings for CRBR Head Modification	Not Completed
Place contract for Head Modification	Not Completed
Complete Head Modification	Not Completed
Reassemble and complete head verification testing	Not Completed
Package and place in storage	Not Completed



SUMMARY LOGIC DIAGRAM FOR
CRBR CLOSURE HEAD MODIFICATION
SMBDB ENHANCEMENT

Figure 4-47

completed but the results were not evaluated. The decision to run SM-10 before SM-9 was made at a planning review meeting at Oak Ridge, Tenn. on May 6, 1983.

To complete the design for SMBDB enhancement of the Closure Head, SM-9 would have to be completed. After results from SM-9 and SM-10 were evaluated, a decision would be made regarding head modification and dynamic test configuration for SM-11. After a successful test, the closure would be modified per the sketches in Figures 4-25 and 4-26. These modifications are shown in Draft ECP W-2133, "HCDA Closure Head Modification" [14].

4.4.3 Outstanding Problems and Descriptions of Actions Needed

The outstanding problem was the modification of the closure head for SMBDB enhancement. The problem and actions needed to resolve the problems are described in Section 4.3.1, "Sodium Slug Energy Absorption Capability of a Modified CRBRP Closure Head". Draft ECP W-2133, "HCDA Closure Head Modification," was written but not acted upon when the project was cancelled in November, 1983.

4.5 Conclusions

Kinematic interaction between the head plugs had been identified as the cause of premature disengagement of the plugs during the SM-8 test. Local machining of the non-load bearing portions of the head plugs and shield plugs had been shown to eliminate the interferences which caused kinematic disengagement of the margin shear rings. With the elimination of the kinematic failure mode, the modified head had a potential capacity to accommodate the energy from a 156MJ slug. The SMBDB design requirement for the closure was that it accommodate the energy from a 75MJ slug. The margin existing between the potential energy absorption capacity was more than sufficient to accommodate any uncertainties in the evaluation procedures.

Finalization of the head machining modifications and confirmation of the head energy absorption capability was to be obtained in additional hydrostatic and dynamic model tests.

As a result of experience gained in the design, analysis, testing and fabrication of the CRBR Closure Head, the following conclusions and recommendations can be made for others involved in the design and analysis of reactor closure heads.

- A. The head components should be checked for possible kinematic interaction mechanisms that may occur as a result of head doming due to internal pressure.
- B. Consideration should be given to seal designs that can easily be replaced to eliminate long down times.

4.6 References

1. "Reactor Closure Head Assembly and Associated Equipment," Equipment Specification E-953011.
2. "Hypothetical Core Disruptive Accident Considerations in CRBRP," CRBRP-3, Volume 1, Rev. 5, "Energetics and Structural Margin Beyond the Design Base," February, 1983.
3. "CRBRP Margin Beyond the Design Base Requirements," WARD-D-0156, Rev. 9, September 26, 1982.
4. "CRBRP Closure Head Capability for Structural Margin Beyond Design Base Loading," WARD-D-0178, November, 1978.
5. "Structural Response of CRBRP Scale Models to a Simulated Hypothetical Core Disruptive Accident," CRBRP-ARD-0218, October 1978.
6. "Hypothetical Core Disruptive Accident Dynamic Load Simulation in an LMFBR Scale Model," DRS 02.20.00, Rev. II. August 30, 1983.
7. "Structural Response of 1/20 Scale Models of the Clinch River Breeder Reactor to a Simulated Hypothetical Core Disruptive Accident," Tech Report #4, October, 1978, C. M. Romander and D. S. Castiostro, SRI International, 33 Ravenswood Ave., Menlo Park, California 94025. [DRS 02.20 TR1]
8. "Static Response of a 1/20 Scale Model of the Clinch River Breeder Reactor Head," Third Technical Report, July 1977. [DRS 02.20 TR0]
9. "Static Tests of 1/20-Scale Model of the Clinch River Breeder Reactor Head in Support on the LMFBR Safety Program," Final Report, December 1982, C. M. Romander and Y. D. Murray, SRI International. [DRS 02.20 TR2]
10. Engineering Study Report "Sodium Slug Energy Absorption Capability of a Modified CRBRP Closure Head," ES-LRD-83-3029, W. E. Pennell, February 10, 1983.
11. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Appendix F.
12. "Waterhammer Analysis," John Parmakian, 1965, Dover Publications, 180 Varick St., New York, NY 10014.
13. "Nuclear Systems Materials Handbook," TID 26666, Volume 1 - "Design Data," Property Code 3307.
14. Draft ECP W-2133, "HCDA Closure Head Modification."

List of ECPs Referenced in Text:

1. Draft ECP W-2133, "HCDA Closure Head Modification."

5.0 REACTOR REFUELING SYSTEM - SYSTEM 41

5.1 Overall System Scope

The reactor refueling system (RRS) provides for insertion and/or replacement of reactor core assemblies, including fuel, blanket, control, and radial shield assemblies. The system consists of the equipment and facilities needed to accomplish the normal scheduled refueling operations, and all other functions pertaining to the handling of core assemblies. These include receiving and unloading new core assemblies, inspection, temporary storage, transferring both new and irradiated core assemblies between storage facilities and the reactor, transferring core assemblies within the reactor, providing inventory control of all core assemblies, removing and examining irradiated core assemblies, and preparing and loading irradiated core assemblies for shipment offsite.

The reactor refueling system has been divided into seven subsystems, based on functions and simplification of intrasystem interfaces, plus shipping containers: (1) 41-1--overall refueling, (2) 41-2--in-vessel handling, (3) 41-3--ex-vessel handling, (4) 41-4--ex-vessel storage, (5) 41-5--conditioning and service, (6) 41-6--receiving and shipping, (7) 41-7--instrumentation and control, and (8) shipping containers.

The first of these subsystems (41-1) is an overall system management and engineering activity; the other six are hardware subsystems. The instrumentation and control subsystem (41-7) contains the central controls, as well as instrumentation and control items mounted on equipment or in facilities, that are parts of the other five hardware subsystems. Shipping containers include both new fuel shipping containers and spent fuel shipping casks designed and procured separately from the remainder of the RRS.

5.2 Functions and Description

5.2.1 Functions

The RRS provides the means of transporting, storing, and handling reactor core assemblies, including fuel, blanket control, and radial shield assemblies, within the CRBRP. The system consists of the facilities and equipment needed to accomplish the normal scheduled refueling operations, and all other functions incident to handling of core assemblies. It is described in more detail in System Design Description (SDD) 41, Reactor Refueling System (Ref. 1).

Major equipment and facilities needed to perform functions incident to the handling of core assemblies and their functions are as follows:

- In-vessel transfer machine (IVTM)--transfer core assemblies between the core and in-vessel storage or transfer positions within the reactor
- Ex-vessel transfer machine (EVTM)--transfer core assemblies between facilities, i.e., EVST reactor, fuel handling cell (FHC), and new fuel unloading stations
- Ex-vessel storage tank (EVST)--provide storage for irradiated core assemblies
- Fuel handling cell (FHC) and spent fuel shipping cask (SFSC)--examine and prepare irradiated core assemblies for shipment
- Auxiliary handling machine (AHM)--handle equipment needed in fuel handling and serve as a maintenance cask for some components
- Reactor Containment Building (RCB) and Reactor Service Building (RSB) plug storage facilities--store port plugs when removed from the reactor, EVST, or FHC

- Floor valves--seal port openings in the reactor, EVST, and FHC when port plugs are removed
- Fuel transfer port cooling inserts--provide cooling for removal of decay heat if an irradiated fuel assembly becomes immobilized in port during transfer
- New fuel shipping containers, new fuel shipping container unloading station, and new core assembly inspection equipment--receive, inspect, store, and prepare new core assemblies for insertion in reactor
- Refueling instrumentation and control system--provide inventory control of all core assemblies.

New core assemblies enter the RSB, are unloaded from their shipping containers and inspected, and are transferred by the EVTM to the sodium-filled ex-vessel storage tank for preheating and storage. After reactor shutdown for refueling, the EVTM transfers new core assemblies from the EVST to the reactor and transfers spent core assemblies from the reactor to the EVST, on a one-for-one basis. The IVTM, in conjunction with the reactor rotatable plugs, transfers spent core assemblies from the core to the in-vessel transfer positions and transfers new core assemblies from the in-vessel transfer positions to the core, on a one-for-one basis. After completion of refueling, and a suitable period for decay, the spent core components are removed from the EVST to the fuel handling cell, examined if desired, and loaded into the spent fuel shipping cask for shipment to a fuel reprocessor.

5.2.2 Description

5.2.2.1 Overall Refueling Subsystem 41-1

5.2.2.1.1 General Features

The major equipment and facilities and the general arrangement of the RRS are shown in perspective in Figure 5-1. The majority of the equipment and facilities are located in the reactor service building, which is adjacent to the

78

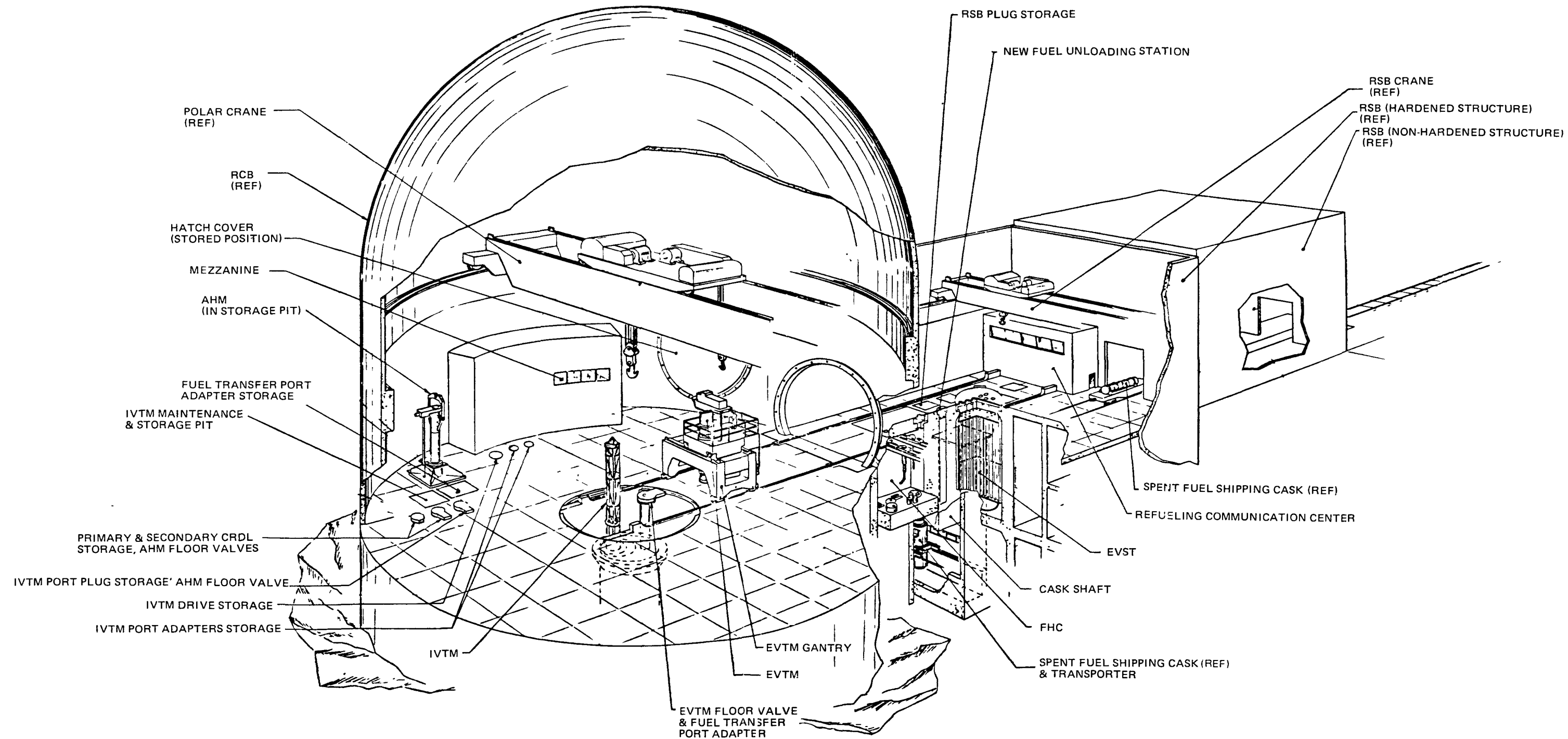


Figure 5-1. Reactor Refueling System (Perspective)

reactor containment building. Important fuel handling facilities and equipment located in the RSB include the ex-vessel storage tank, the fuel handling cell, new and spent fuel unloading and loading facilities, miscellaneous storage facilities, and a communications center from which refueling and other fuel handling operations* are coordinated. Equipment and facilities located in the RCB are limited to those involved with fuel handling in the reactor; they include the in-vessel transfer machine, the auxiliary handling machine, and miscellaneous storage facilities.

Fuel movement between buildings is accomplished during reactor shutdown using an EVTM mounted on the EVTM gantry-trolley. The EVTM and gantry-trolley pass through the RCB equipment hatch, which is opened for these operations. The EVTM and gantry-trolley remain in the RSB during reactor operation. All storage locations are accessible by the RCB polar crane or the RSB crane. The subsystem hardware items and their operations are described in subsequent paragraphs under the appropriate subsystem. The overall system is designed and arranged to meet the inservice inspection (ISI) requirements of Ref. 2 and the as low as reasonably achievable (ALARA) criteria of Ref. 3. The seismic design criteria are given in Ref. 4.

5.2.2.1.2 Safety and Code Classification

The final safety and American Society of Mechanical Engineers (ASME) code classifications of System 41 components are listed in Table 5-1. The safety classes are defined in Section 1.3.1.6 of Overall Plant Design Description (OPDD) 10 (Ref. 5.).

*Refueling is generally defined as reactor refueling (i.e., the operations involving transfer of fuel within, or into, or out of the reactor, and which take place while the reactor is shut down.) Fuel handling is a general term, referring to any operation involving the handling of fuel or other core assemblies.

TABLE 5-1
COMPONENT SAFETY AND CODE CLASSIFICATION--SYSTEM 41

Subsystem or Component	Safety Classifi- cation	ASME Code Class
In-vessel handling subsystem		
IVTM pressure boundary	None	VIII-1
IVTM grapple and stem	None	III-NF2
AHM pressure boundary	None	VIII-1
AHM grapple and hoist drive	None	III-NF3
AHM floor valve (FV) pressure boundary	None	VIII-1
Ex-vessel handling subsystem		
EVTM pressure boundary	SC-3	III-3
EVTM grapple	None	III-NF3
EVTM FV	SC-3	VIII-1
Core component pot (CCP)	SC-2	III-2
Rotating guide tube pressure boundary	SC-1	III-1
Fuel transfer port adapter pressure boundary	None	VIII-1
Ex-vessel storage subsystem		
Storage vessel and closure head	SC-2	III-2
Turntable	SC-2	III-NF2
Guard tank	SC-3	III-3
EVST adapter pressure boundary	None	VIII-1
Conditioning and service subsystem		
Shielded windows	None	VIII-1
Core assembly cooling grapples	None	III-NF3
Spent fuel transfer station	None	III-NF3
Argon circulation system (ACS)		
main filters	None	VIII-1
Spent fuel transfer port assembly	None	VIII-1
Fuel examination station port assembly	None	VIII-1
28-in. maintenance FV	None	VIII-1
Roof closure	None	VIII-1
Penetrations	None	VIII-1
Receiving and Shipping Subsystem		
Cask-FHC seal assembly	None	VIII-1
Instrument and control subsystem		
	None	Commercial

5.2.2.2 In-vessel Handling Subsystem (IVHS) 41-2

The primary function of the IVHS is to move core assemblies between their normal positions in the reactor core and the transfer positions outside the core, but within the reactor vessel. In the transfer positions, the core assemblies are accessible for transfer in or out of the reactor vessel.

The IVHS includes the in-vessel transfer machine, the auxiliary handling machine, AHM floor valves, and other minor equipment and facilities. The major components are briefly described in the following sections. For a more detailed description, refer to Refs. 6, 7, 8, 9, 10, and 11.

5.2.2.2.1 In-Vessel Transfer Machine (IVTM)

The IVTM implements the primary IVHS function mentioned previously by the following operational capabilities:

- Grapple and release core assemblies
- Raise and lower core assemblies
- Provide holddown of adjacent core assemblies
- Uniquely identify core assemblies
- Orient and center core assemblies for insertion in the core.

The IVTM consists of two principal subassemblies--the in-vessel section and the ex-vessel or drive section (see Figure 5-2). The in-vessel section contains the grapple, the centering device, the identification mechanism, the holddown mechanism, and the seals for reactor cover gas containment. The ex-vessel section contains the drive equipment that powers the in-vessel section mechanisms. The IVTM is a straight push-pull machine; horizontal translation is accomplished by rotating the plugs in the reactor head.

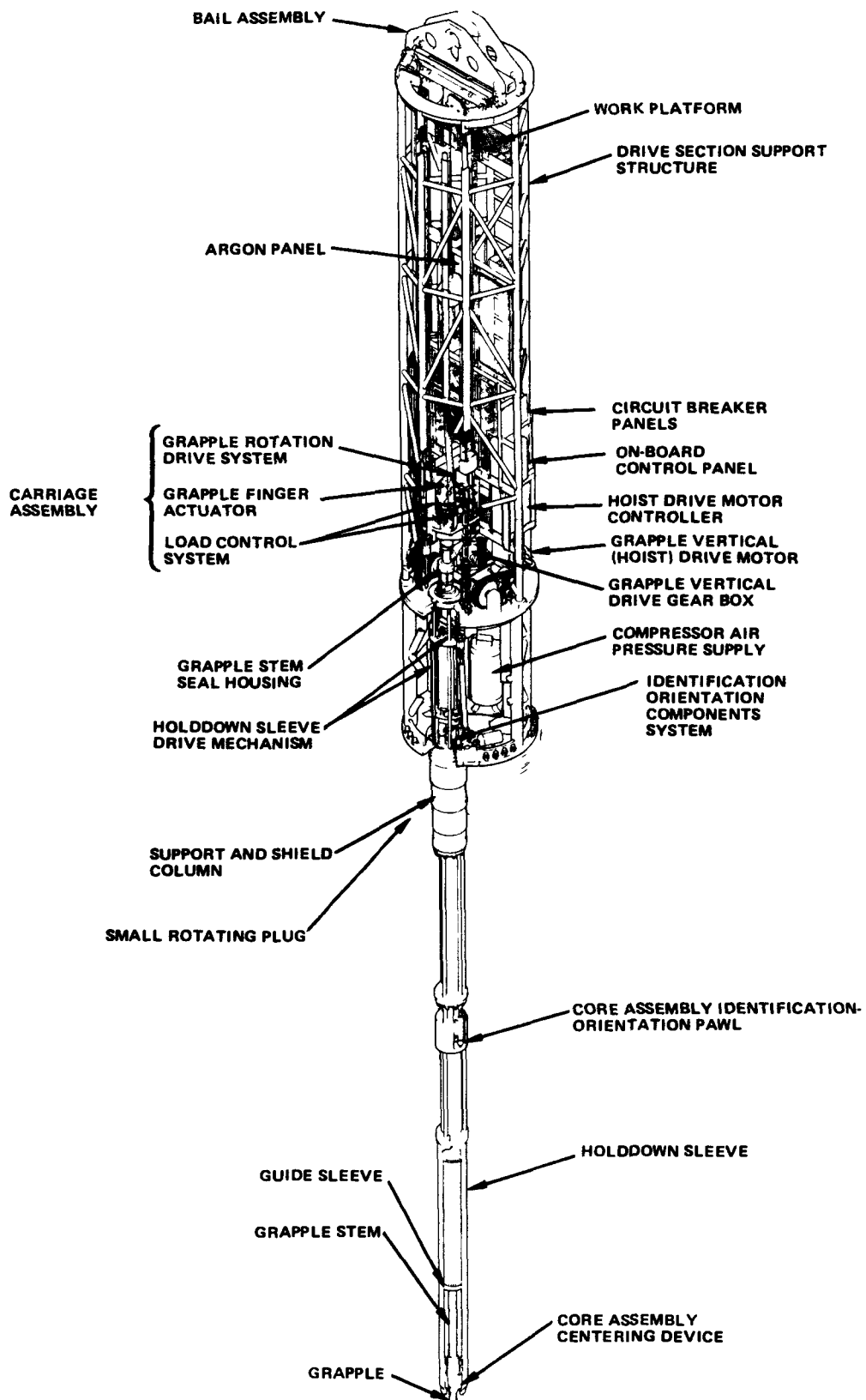


Figure 5-2. In-Vessel Transfer Machine

The lower portion of the in-vessel section is exposed to reactor sodium and cover gas and must be installed and removed using the AHM, which provides a shielded, inert gas environment. The drive section is exposed only to the RCB air environment and can be installed using the polar crane.

5.2.2.2.2 Auxiliary Handling Machine (AHM)

The AHM, shown in Figure 5-3, is a single-barrel handling machine which provides an inert atmosphere and radiation shielding for transferring sodium-wetted radioactive components between the reactor and storage facilities within the RCB.

These operations are primarily involved in preparation for (and termination of) refueling: (1) removing the IVTM port plug and installing the IVTM in-vessel section, and (2) returning these items after refueling. The AHM will also be used to support maintenance functions for System 31 (i.e., remove and install control rod drive lines and core inlet modules). The special tools, grapples, adapters, etc., required for these operations are provided by System 31.

5.2.2.2.3 AHM Floor Valve

The function of the floor valve (Figure 5-4) is to seal a reactor or other facility access port during various phases of the refueling operation when the port plug is removed. The floor valve prevents the release of radioactivity-contaminated gas and shields the building operating floor environment from radiation. The floor valve is a portable item of equipment.

5.2.2.3 Ex-Vessel Handling Subsystem (EVHS) 41-3

The primary function of the EVHS is to transfer core assemblies between the reactor, the ex-vessel storage tank, the new fuel unloading stations, and the fuel handling cell.

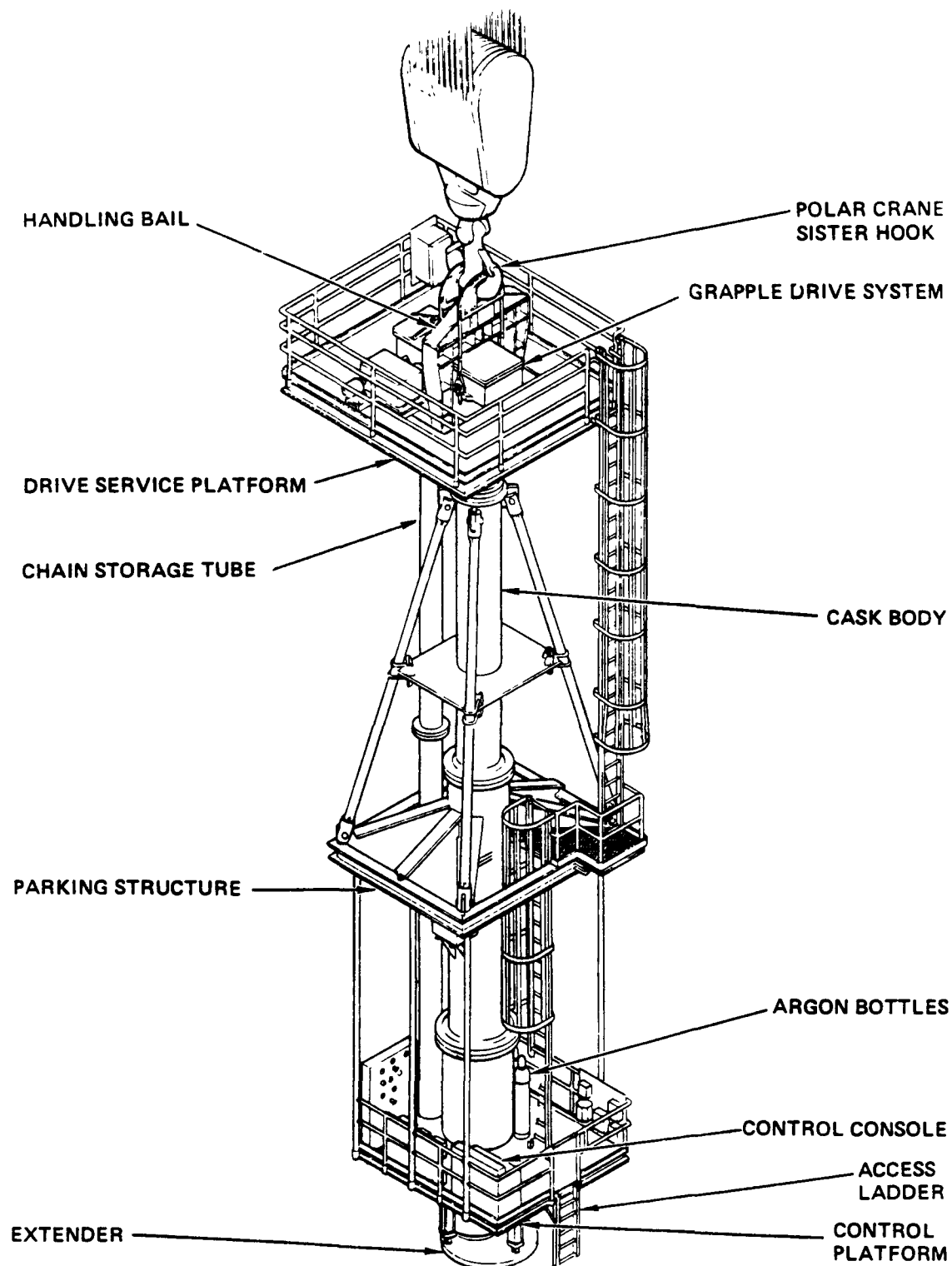


Figure 5-3. Auxiliary Handling Machine (AHM)

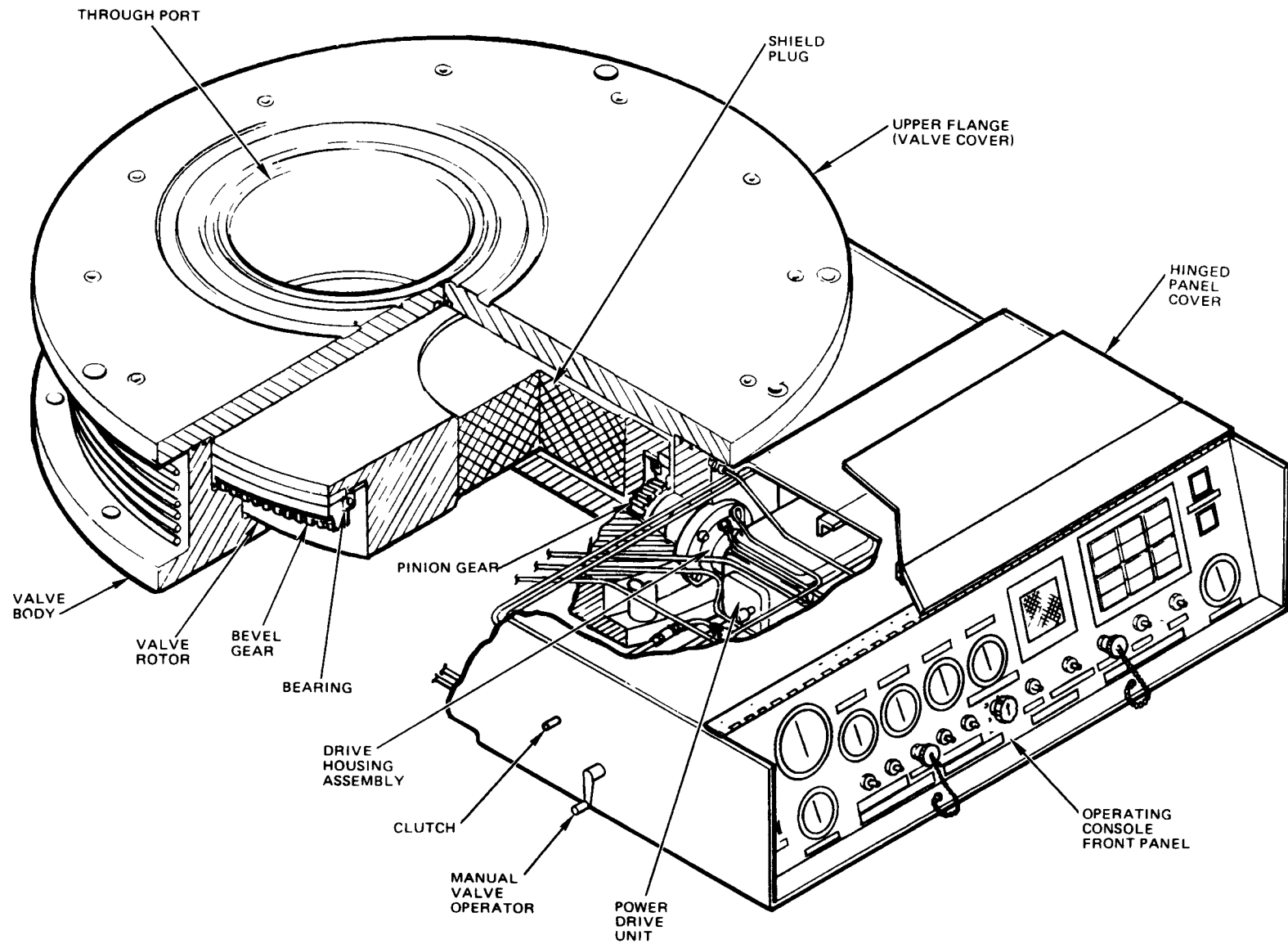


Figure 5-4. AHM Floor Valve

The EVHS includes the EVTM, the EVTM gantry-trolley, EVTM floor valves, core component pots, a plug storage facility in the reactor service building, the reactor fuel transfer port adapter, rotating guide tube, and miscellaneous adapters. The major components of the subsystem are briefly described in the following paragraphs. For a more detailed description, see Refs. 12 through 19.

5.2.2.3.1 Ex-Vessel Transfer Machine (EVTM)

The primary function of the EVTM is to handle and transfer both new and irradiated core assemblies. The EVTM implements the primary function by the following operational capabilities:

- Grapple and release core assemblies, core component pots, and port plugs
- Raise and lower core assemblies, core component pots, and port plugs
- Provide containment of radioactive cover gas
- Maintain an argon environment
- Maintain preheat temperature for new core assemblies
- Provide cooling for spent core assemblies
- Provide radiation shielding.

The EVTM, shown in Figure 5-5, is a shielded, inerted, single-barrel machine. The machine has provisions for maintaining the preheat of new core assemblies and for cooling irradiated core assemblies.

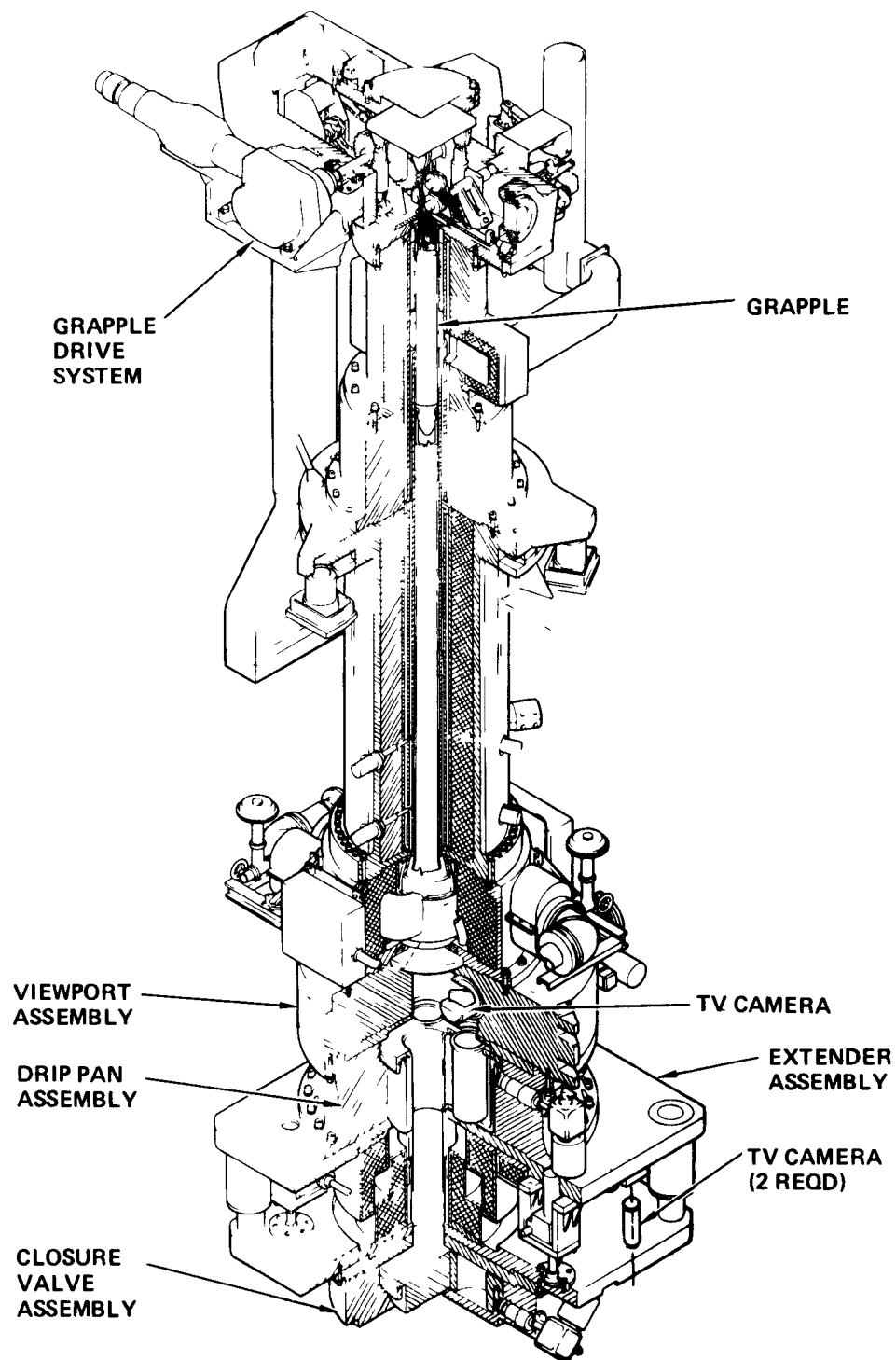


Figure 5-5. Ex-Vessel Transfer Machine (EVTM)

The EVTM is mounted on a trolley which, in turn, is positioned on rails on top of a gantry. The gantry moves on crane rails between the RCB and the RSB. The trolley rails are perpendicular to the gantry rails, allowing complete indexing of the EVTM.

Irradiated core assemblies are handled only in sodium-filled core component pots. New assemblies may be handled either in pots (between the reactor and the ex-vessel storage tank) or bare (between the new fuel unloading stations and the preheat and storage positions in the ex-vessel storage tank).

5.2.2.3.2 EVTM Gantry-Trolley

The function of the EVTM gantry-trolley (see Figure 5-6) is to transport, support, and position the EVTM over the reactor, the ex-vessel storage tank, the new fuel unloading stations, the RSB plug storage, and the fuel handling cell, as required.

The gantry-trolley is essentially a conventional gantry crane without a hoisting system and is similar in concept to the Fast Flux Test Facility Closed-Loop Ex-Vessel Machine (FFTF CLEM) transporter, although smaller in size and lighter in weight. The major assemblies consist of a gantry, trolley, seismic restraint system, operator's cab with controls, and a cable system. The EVTM is mounted on the trolley, and the trolley travels in an east-west direction on top of the gantry. The gantry travels in a north-south direction on rails that are mounted in trenches in the operating floor. The control cab is attached to the trolley on the south side of the gantry, away from the reactor. There are two levels to the cab: the upper level contains the EVTM and the gantry-trolley motor control center; the lower level contains the EVTM and gantry control console.

The EVTM, gantry, and trolley have been sized so that they pass through the 44.5-ft-diameter RCB equipment hatch when the trolley is centered on the gantry.

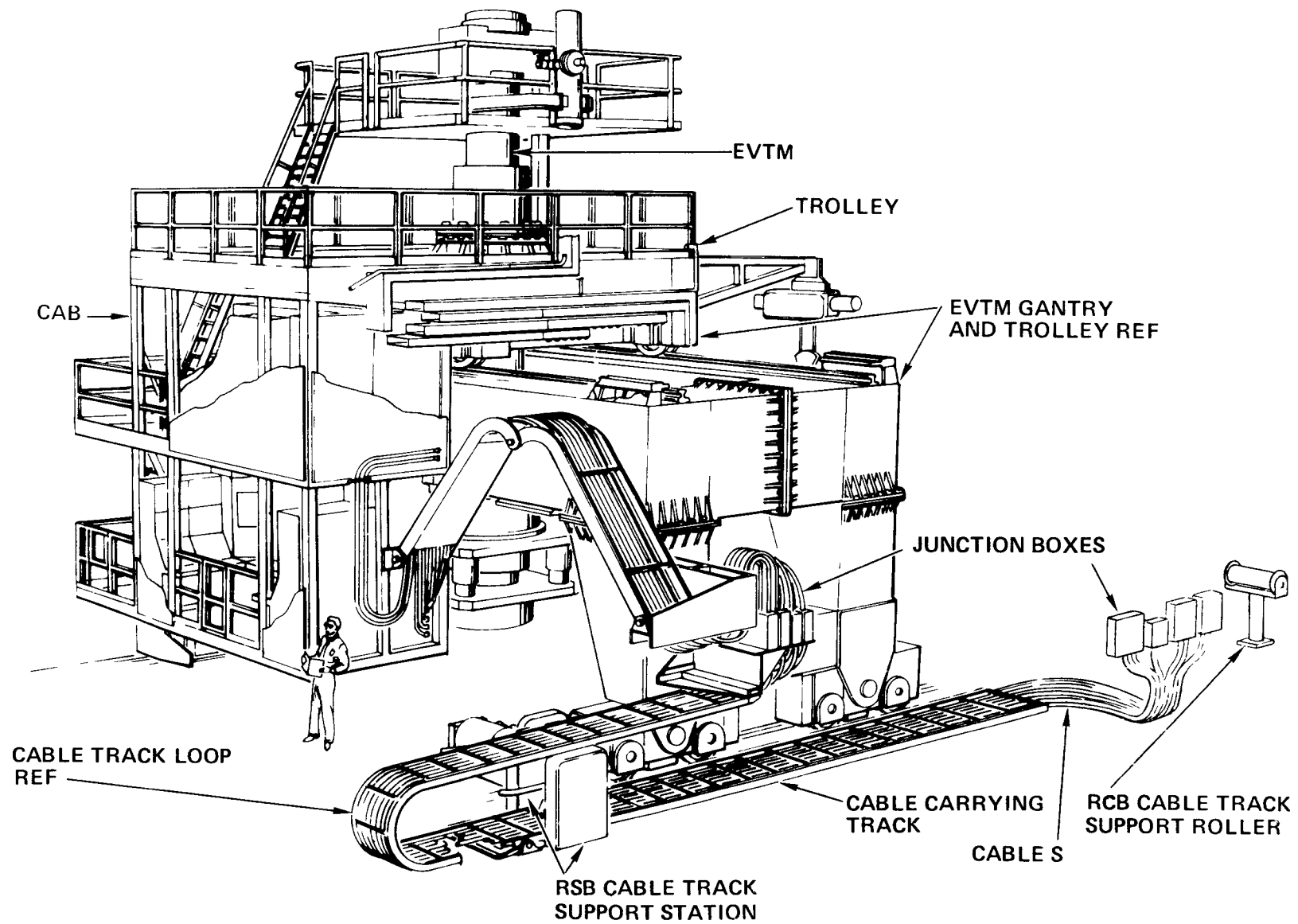


Figure 5-6. EVTM Gantry-Trolley

5.2.2.3.3 EVTM Floor Valve

The EVTM floor valve is essentially identical in function and external geometry to the AHM floor valve and is described in Section 5.2.2.2.3 (see Figure 5-4).

Three EVTM floor valves are required during refueling--one in the RCB mounted on the reactor fuel transfer port, and two in the RSB on EVST fuel transfer ports. For fuel handling operations taking place while the reactor is operating, the EVTM floor valves will be on the EVST and/or the fuel handling cell. A floor valve is not required at a new fuel unloading station or the RSB plug storage inasmuch as the EVTM mates directly with adapters provided at each of these facilities which serve the same purpose.

5.2.2.3.4 Core Component Pot (CCP)

The function of the CCP is to provide a sodium environment for core assemblies in transit between the reactor vessel and fuel handling facilities.

The CCP (see Figure 5-7) is a tubular steel container, closed at the bottom, with a handling socket at the upper end that can be grappled by the EVTM or the fuel handling cell grapple. Near the top of the CCP is a siphon assembly, designed to remove a predetermined volume of sodium at the time the pot is raised above the facility sodium level, thereby allowing for subsequent sodium thermal expansion due to the maximum powered irradiated fuel assembly decay heat. This feature prevents overflow and greatly reduces sodium drippage and splash, with their consequent operational and contamination problems.

The CCP is used to receive core assemblies during in-vessel transfer, to transport them in the EVTM, and to store them in the EVST and FHC. A CCP is needed for every storage position in the EVST.

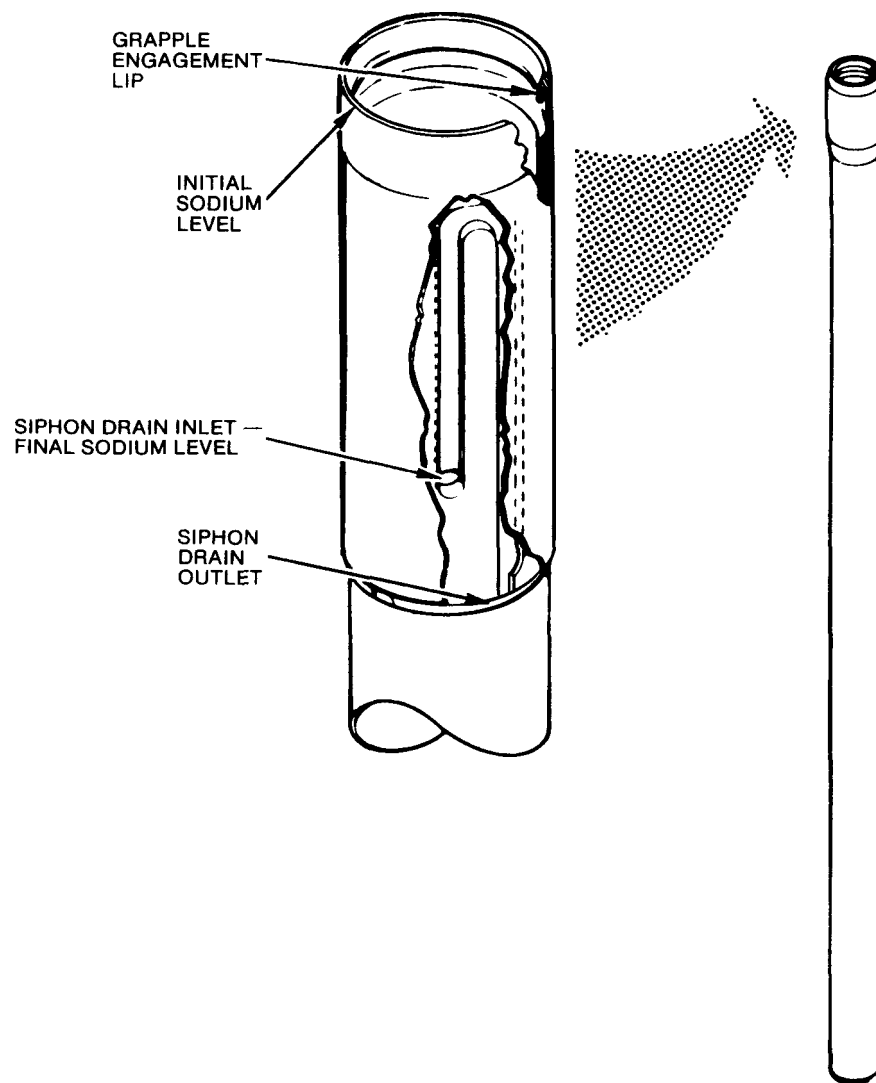


Figure 5-7. Core Component Pot (CCP)

5.2.2.4 Ex-Vessel Storage Subsystem (EVSS) 41-4

The EVSS performs the functions of preparing new core assemblies for insertion into the reactor and of providing safe, controlled storage for new and irradiated core assemblies. The subsystem includes one ex-vessel storage tank (EVST) (see Figure 5-8).

The EVST capabilities required to implement the previously mentioned EVSS functions are as follows:

- Preheating of new core assemblies
- Storing new and irradiated core assemblies under sodium
- Cooling irradiated core assemblies
- Containing cover gas
- Providing structural support and physical separation of fuel assemblies to maintain their subcriticality
- Radiation shielding.

The EVST is a sodium-filled storage facility with a two-tier rotating turntable and a fixed closure head with fuel transfer ports. The EVST is located in the RSB between the EVTM gantry rails in a nitrogen-gas-filled concrete vault. For additional details, see Refs. 20 and 21.

5.2.2.5 Conditioning and Service Subsystem (CSS) 41-5

The primary functions of the conditioning and service subsystem performed by the FHC are to: (1) receive irradiated core assemblies from the EVTM, (2) provide limited interim storage for these irradiated assemblies, (3) examine selected irradiated assemblies, and (4) load irradiated core

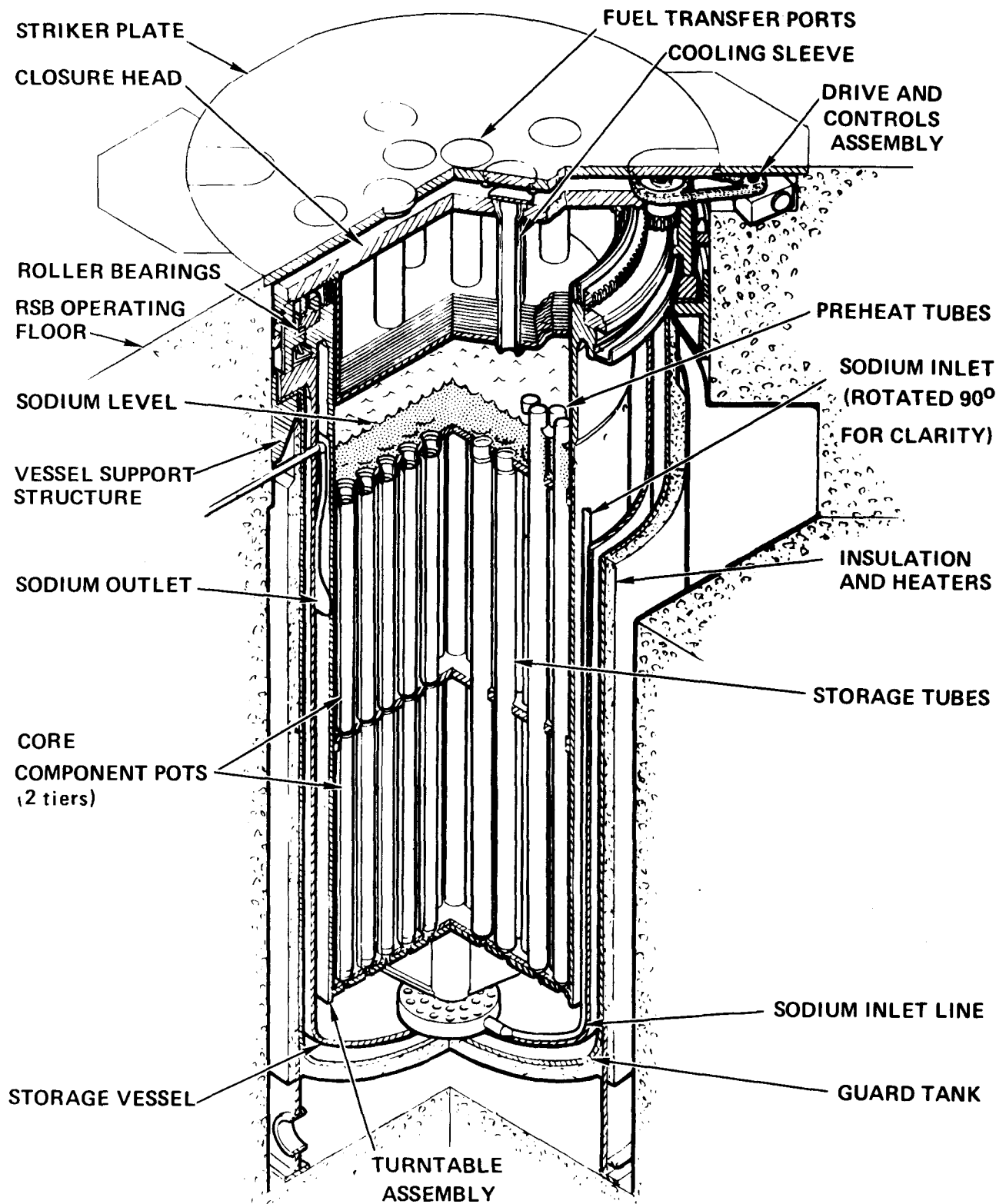


Figure 5-8. Ex-Vessel Storage Tank (EVST)

assemblies into casks for shipment off site. Other functions of the CSS, also performed by the FHC, are providing service and maintenance of radioactive fuel handling equipment (e.g., grapple replacement and drip pan change-out for the ex-vessel transfer machine). The CSS also provides contingency storage for low-heat-producing core assemblies in the event of a complete core unloading.

The CSS includes the fuel handling cell, operating and maintenance galleries, handling and servicing equipment within these areas, the FHC gas circulation system, and low-heat-producing assembly storage. Additional details pertaining to specific components or equipment groups are given in the following sections and Refs. 22 through 29.

5.2.2.5.1 Fuel Handling Cell (FHC)

A perspective of the FHC is given in Figure 5-9. The FHC is a shielded, inerted, alpha-tight hot cell. A steel lining provides ease of decontamination and a gastight enclosure. A grid system provides a separation of at least 4 in. between the heated zone of an accidentally dropped core assembly and the FHC floor liner. The wall between the cell and the operating gallery is of high-density concrete to provide biological shielding and to limit wall thickness so that commercially available master-slave manipulators may be used. Master-slave manipulators penetrate this wall, which also contains shielded viewing windows. In-cell equipment includes a remote-controlled crane and a powered manipulator (i.e., mechanical arm). All operations in the cell are observed and controlled from the operating gallery. In addition to the shielded viewing windows, visual access to the cell interior is provided by both a periscope and closed-circuit television.

In a typical spent fuel handling sequence, spent fuel in a CCP is lowered through the transfer port by the EVTm, through an EVTm floor valve, into the spent fuel transfer station directly below the port. When the assembly is ready for removal from the transfer station, the in-cell crane with a gas-cooling grapple attached is used to lift the assembly from the CCP. The

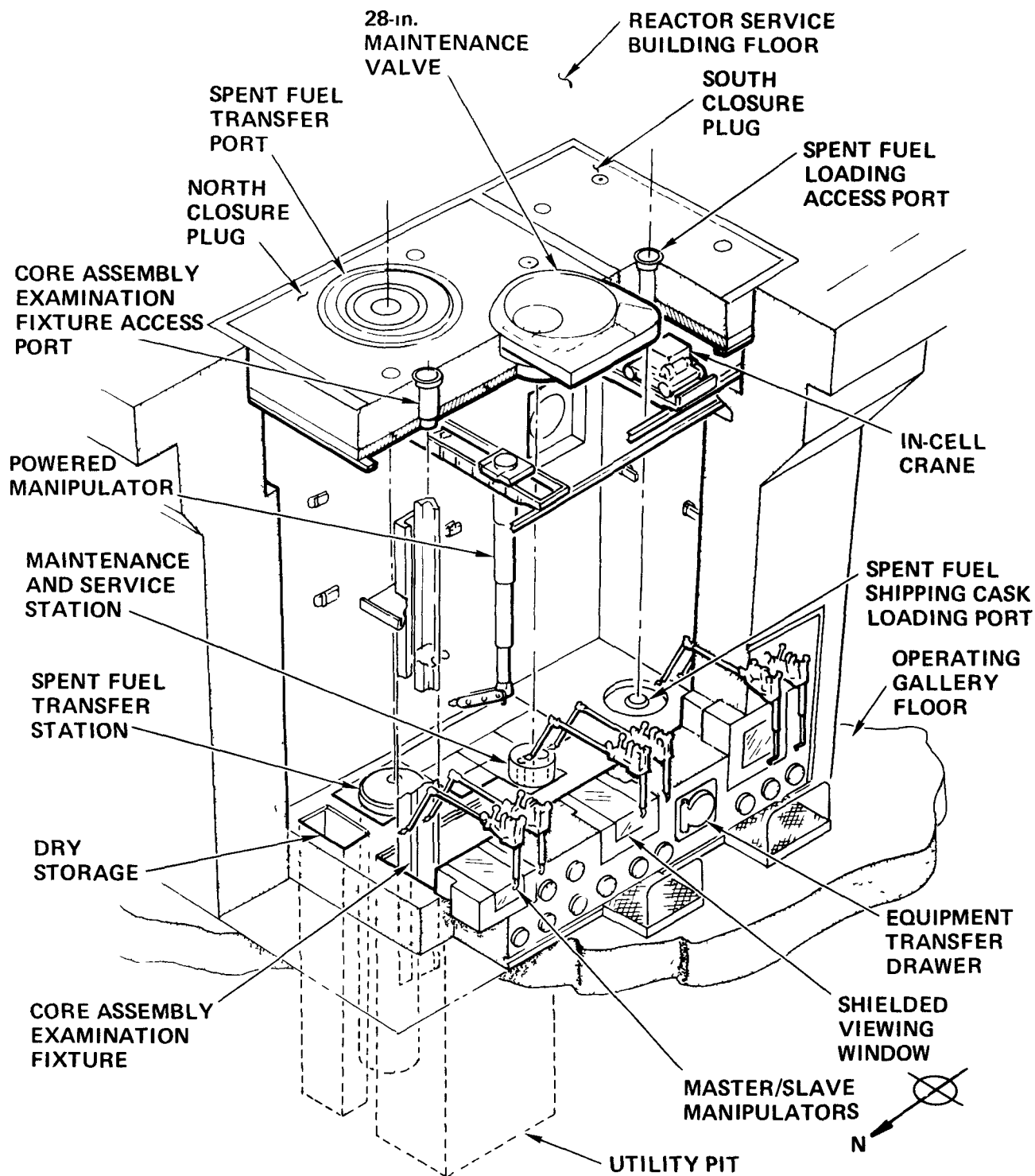


Figure 5-9. Fuel Handling Cell (FHC)

bare assembly is allowed to drip dry, during which time cooling is provided to the assembly by means of forced convection of argon from a gas-cooling grapple hung from the crane. The assembly is then identified. If desired, it may be moved to the examination station where visual inspection and gauging can be performed.

When the spent fuel has completed the preceding sequence, it is lowered into the spent fuel shipping cask located in a corridor below the cell floor. The sequence is repeated as many times as necessary to fill the shipping cask.

5.2.2.6 Receiving and Shipping Subsystems (RRS) 41-6

The function of the receiving and shipping subsystem is to receive, handle, and unload new fuel shipping containers; inspect new core assemblies, and receive, handle, and load spent fuel shipping casks. The subsystem consists of new fuel unloading stations, new core assembly inspection equipment, a cask corridor/shaft and transporter, spent fuel shipping cask preparation and closeout station, and other miscellaneous equipment and facilities.

5.2.2.6.1 New Fuel Unloading Stations

Two new fuel unloading stations are located between the EVTM gantry-trolley rails in the RSB. Each station supports and positions, in a vertical orientation, new core assembly shipping containers and the universal container to permit loading or unloading of a core assembly by the EVTM.

Each station consists of a pit in the operating floor with a structural steel support for the shipping containers (see Figure 5-10). An adapter which houses the 24-in.-diameter unloading station valve is mounted on the upper end of the container holder to provide an interface between the container and the EVTM. For additional details, see Ref. 30.

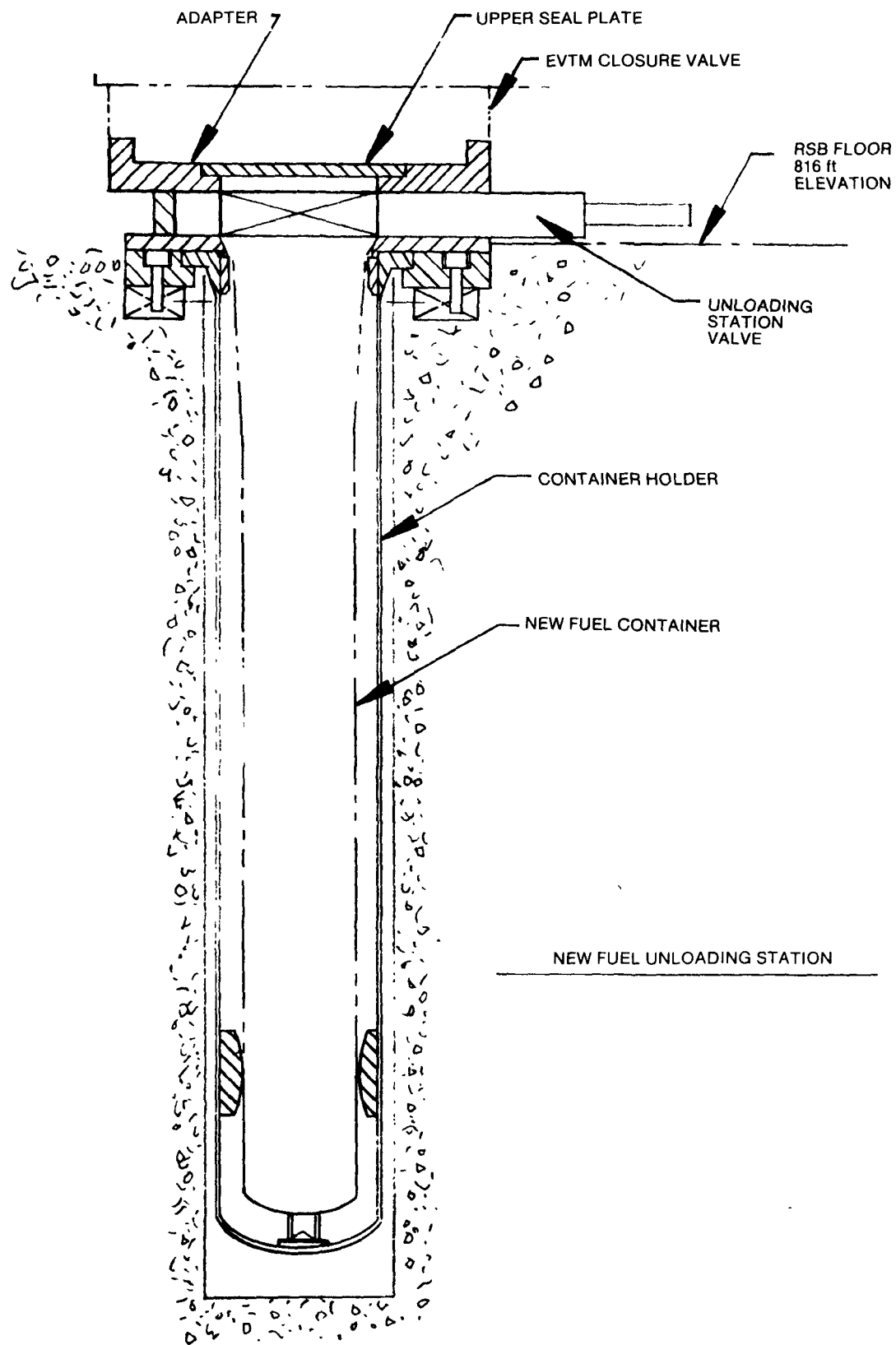


Figure 5-10. New Fuel Unloading Station

5.2.2.6.2 New Core Assembly Inspection Equipment

The function of the new core assembly inspection equipment (NCAIE) is to perform the required receiving inspection of new core assemblies, primarily new fuel assemblies which may be slightly radioactive.

The major component of the equipment is a lightly shielded structure installed above a new fuel unloading station (see Figure 5-11). The shielding permits personnel to perform the required visual and manual inspection operations without exceeding the 125 mrem/quarter allowable dose. The EVTM gantry-trolley incorporates a hoist and the NCAIE a grapple to raise and lower the core assembly. For additional details, see Ref. 31.

5.2.2.7 Instrumentation and Control (I&C) Subsystem 41-7

The functions of the instrumentation and control subsystem are primarily to optimize control of the RRS operations to minimize reactor downtime and to provide inventory control of all core assemblies.

The I&C subsystem includes all electrical, electronic, electromechanical, and pneumatic sensors, instruments, actuators, and other components required for control and monitoring of all RRS subsystems.

A pictorial representation of the I&C for the RRS is shown in Figure 5-12. The refueling communications center (RCC) is shown schematically in the upper right portion of the figure. Coordination of the operations of the in-vessel handling subsystem, the ex-vessel storage subsystem, and the conditioning and service subsystem by means of the RCC is indicated. The figure also indicates the relationships between: (1) the plant data handling and display system (PDH&DS, System 91) and the RCC, and (2) the reactor enclosure system (System 32) and the IVTM. Also shown are three digital mini-computers--one for control of the IVTM, one for the EVTM, and one for control and data acquisition in the RCC. Controls for manual operations are also indicated schematically.

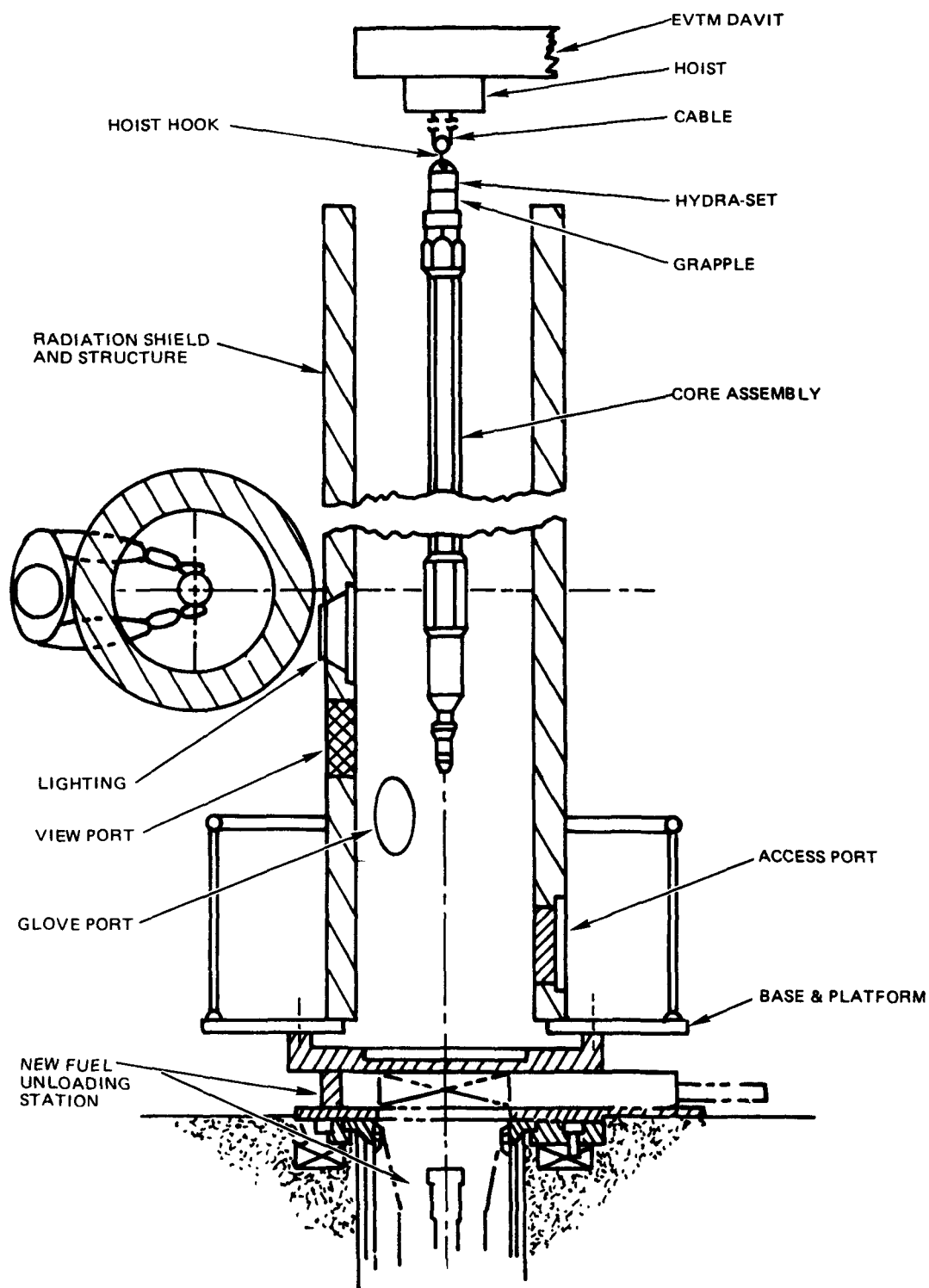


Figure 5-11. New Core Assembly Inspection Equipment

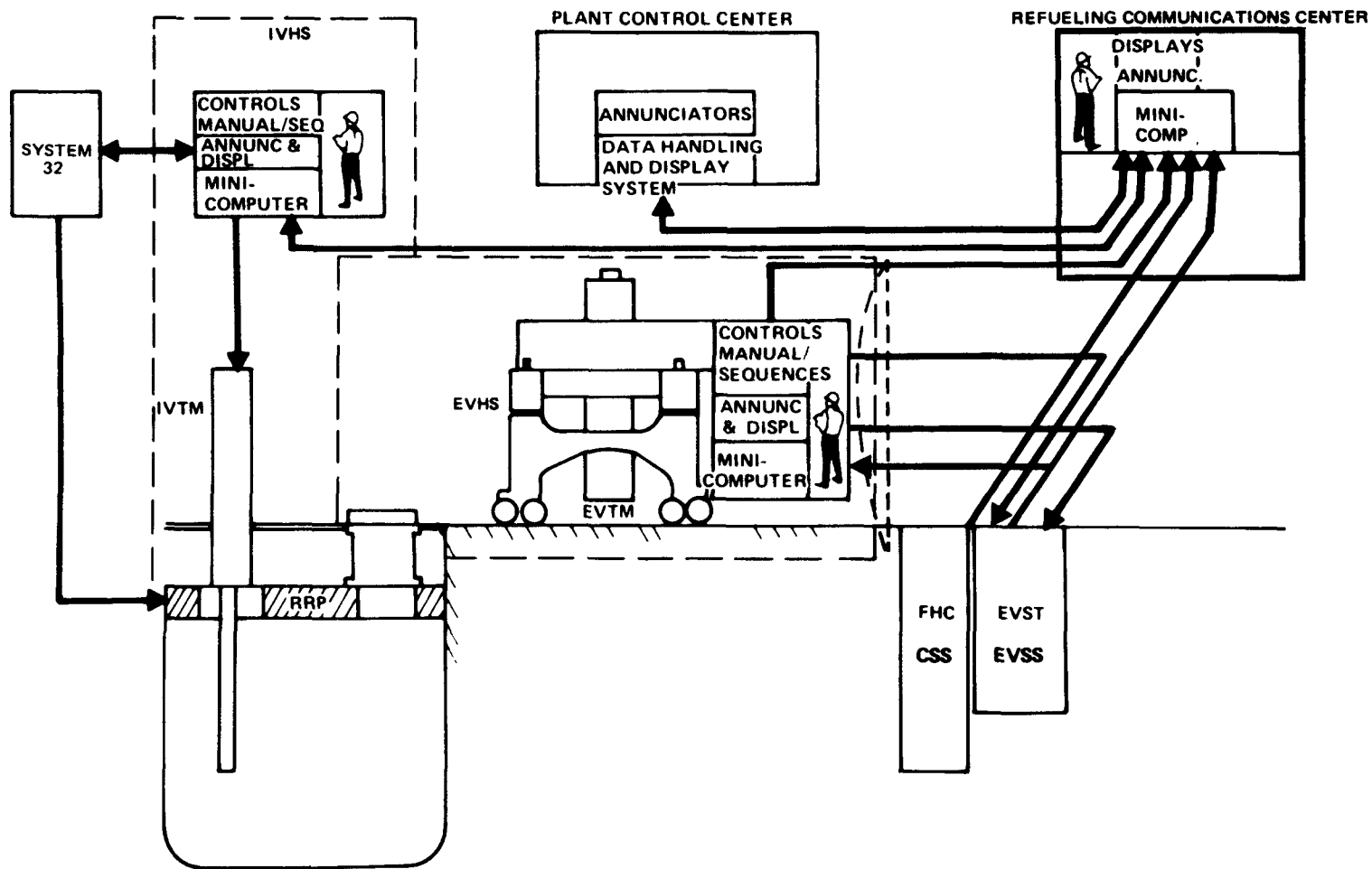


Figure 5-12. Reactor Refueling System (RRS) Control Concept

The RCC is an enclosed structure located at the south end of the gantry tracks in the RSB and elevated above the operating floor level.

It contains the RCC supervisory console, the RCC computer system, the EVST control panels and equipment, auxiliary liquid metal (System 81), and inert gas receiving and processing (System 82) I&C equipment, record storage, desks, and tables.

The RCC equipment is used to perform a variety of functions. The supervisory console contains displays that show the current status of RRS machine and facilities. Control panels provide for initiating control functions performed by the RCC computer system. The master annunciator panel provides overall RRS machine annunciatory status. Voice communication to all points in the RRS and CRBRP is provided.

The windows in the RCC permit direct viewing of the RSB above-floor areas.

The RCC block diagram, Figure 5-13, shows the major equipment of the RCC and the signal paths between this equipment and equipment in other locations in the CRBRP. The RCC computer is shown linked to computers located at the EVTM and IVTM control panels, as well as to the PDH&DS computers and to a remote terminal located in the plant control room. This terminal permits communication with the RCC computer to verify that the machines and facilities of the RRS are in a safe state, in the event it is ever necessary to evacuate the RCB and RSB.

The RCC computer is also shown to be the interface between the RRS and the PDH&DS (System 91) computers. The PDH&DS serves as a repository for historical information on core assemblies and fuel handling operations and, additionally, performs IVTM position calibration computations based on input data received from System 41. The RCC computer receives data from the EVHS and IVTM computers and RRS control panels and drives data displays on the supervisory console. For additional details, see Refs. 32 and 33.

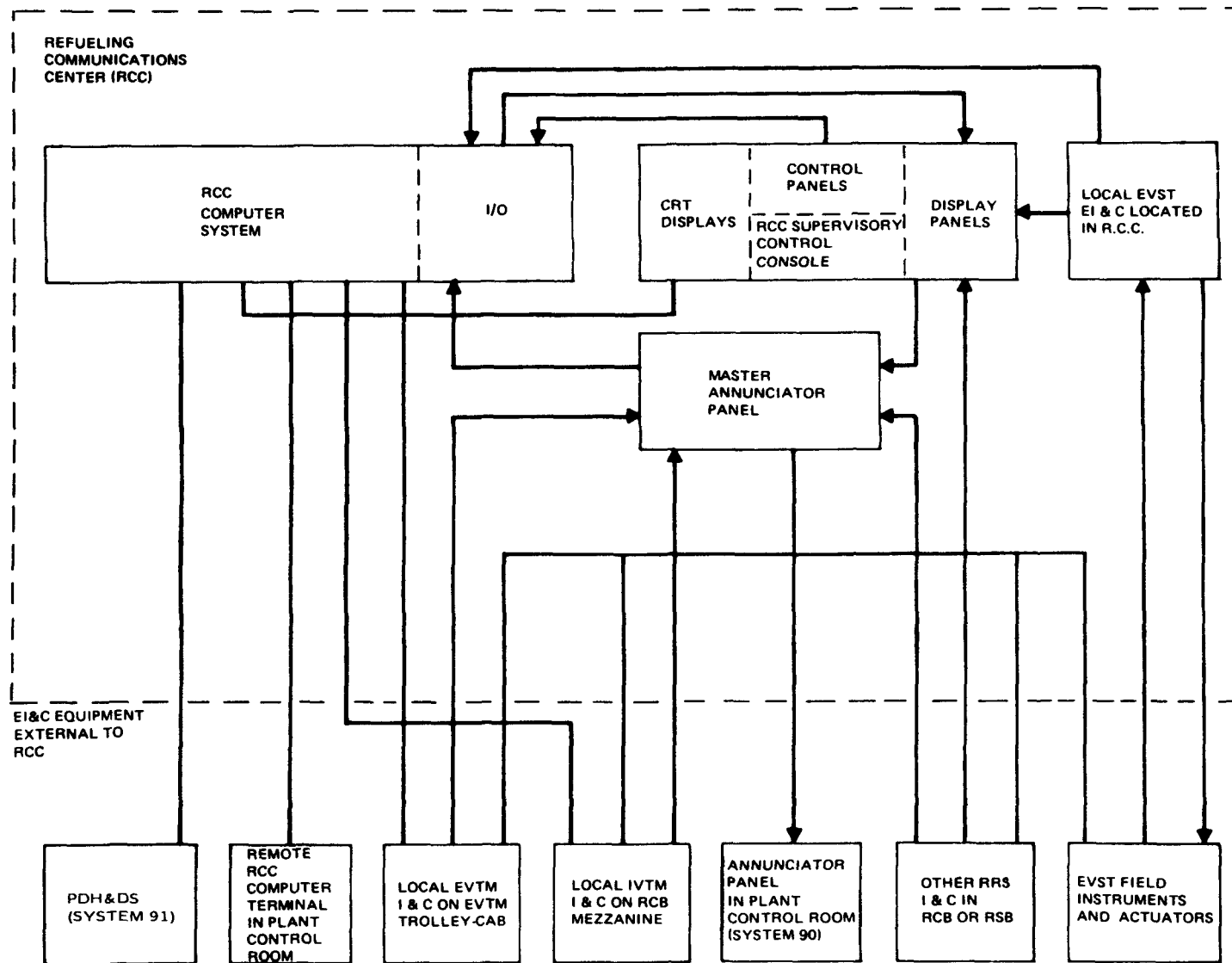


Figure 5-13. Refueling Communications Center (RCC) Block Diagram

5.3 Evolution of Design

The design of the reactor refueling system (RRS) for CRBRP began in 1973. The CRBRP reactor refueling system design utilizes major features of the refueling systems in earlier liquid-metal-cooled reactors in the United States--the Fast Flux Test Facility (FFTF), Hallam, Sodium Reactor Experiment, Fermi, and EBR-II reactors--as well as features from the German, French, and British LMFBR programs. Maximum use of FFTF technology in fuel-handling equipment design was a key factor in selection of the reference design concept, minimizing the required development of the CRBRP system.

The initial reactor refueling system concept grew out of a 60-day study by the reactor manufacturers (Ref. 34). This concept, which was called the reference design, was based on FFTF studies, reactor manufacturer program definitions for design and system requirements, and Project Management Corporation (PMC) guidelines.

Beginning in October 1973, a refueling task force reviewed the concept and identified a number of studies for additional review. The following studies were then undertaken:

- IVTM design options for a straight-pull machine working with the triple-rotating plug, reactor head design
- Alternate methods for transporting the auxiliary handling machine (AHM)
- Consideration of a single versus a multifunction AHM
- Alternate methods for cooling spent fuel assemblies in the EVTM
- Means for reducing EVTM size and weight
- Ex-vessel storage tank (EVST) design options

- Alternate methods for loading the spent fuel shipping cask (SFSC)
- Determination of the refueling equipment to be located in the reactor containment building
- Selection and arrangement of refueling equipment.

In December 1973, an independent review task force identified a number of potential improvements in the refueling system, and in mid-January the designers began to estimate the total cost of the reference system. At the same time, efforts to optimize the system and reduce its cost were under way. Continuing through July 1974, these efforts reduced CRBRP costs more than \$35 million for the reactor refueling system alone. The cost-cutting efforts also produced savings in interfacing systems, primarily the reactor containment building and the reactor services building.

5.3.1 Conceptual Design

In 1974, the refueling system design was based on the following concepts.

Most of the reactor refueling equipment and handling and storage facilities were located in the reactor services building (RSB), adjacent to the reactor containment building (RCB). The ex-vessel transfer machine (EVTM), mounted on a gantry, moves fuel between buildings during reactor shutdown, passing through the opened RCB equipment hatch.

Within the reactor vessel, a straight pull in-vessel transfer machine (IVTM) handles the fuel assemblies. The IVTM mounts on the small rotating plug that forms part of the triple-rotating plug assembly in the head of the reactor vessel. The machine grapples a fuel assembly and raises it above the core. Then the plugs rotate, and the fuel is placed in a transfer pot in the reactor vessel from which the ex-vessel transfer machine can remove it from the reactor vessel.

The EVTM provides either heating or cooling, as required, during transfer of new or irradiated fuel between reactor and ex-vessel fuel handling facilities and provides shielding for plant refueling and maintenance personnel. The in-vessel transfer machine will remain in the reactor only during refueling operations. It will be installed and removed by an auxiliary handling machine (AHM), also designed to remove port plugs from the RCB facilities. When not in use, both the AHM and IVTM will be stored in the reactor containment building.

Fuel handling facilities in the reactor service building include a sodium-filled ex-vessel storage tank (EVST) for interim decay of spent fuel, a fuel handling cell (FHC) for examination and shipment of spent core components, a new fuel storage cell (NFSC), and the refueling control room.

Conceptual design reviews of the major refueling system equipment were held as follows:

Item	Conceptual Design Review Date
SDD-41	October 1974
In-vessel transfer machine	October 1974
Auxiliary handling machine	July 1975
Ex-vessel transfer machine	February 1975
Core component pot	July 1975
Ex-vessel storage tank	August 1975
Fuel handling cell	February 1975
New-fuel shipping	May 1975
Spent-fuel shipping	June 1975
Refueling instrumentation and control	December 1974

The new-fuel storage cell (NFSC) and new-fuel transfer machine (NFTM) were deleted from the reactor refueling system (RRS) reference design as a result of trade studies, which revealed that their elimination would be cost effective. Substantial capital cost reductions resulted in both cases.

5.3.2 Preliminary Design

Trade studies and cost optimization as well as results from the development testing continued to modify the design of the RRS slightly. The addition of a rotating guide tube (formerly known as the pivoting in-vessel transfer position) was approved, and design of this assembly began. The rotating guide tube is located over the transfer positions in the reactor. Its purpose is to reduce refueling time by reducing the number of EVTm coupling and decoupling movements required at the reactor.

The reference design of the EVST included a bottom seismic key to prevent unacceptable displacement of the vessel under the loading produced by seismic forces. Three alternate concepts eliminating the seismic key were evaluated: a two-tier rotating turntable similar to the previous reference design, a two-tier rotating head similar to the reactor, and a single-tier rotating turntable. The results of the study showed that a seismic key was not required. The two-tier rotating turntable concept was chosen to be the reference design. Its principal advantages are greater reliability and maintainability and less complicated fuel handling operations.

A cost reduction of \$710,000 resulted from substitution of a natural-convection-cooled spent-fuel transfer station for the sodium-filled spent-fuel storage tank in the FHC.

Preliminary design reviews of the major refueling system equipment were held as follows:

Item	Preliminary Design Release Date
In-vessel transfer machine	December 1975
Fuel handling cell	March 1976
Refueling instrumentation and control	October 1976
Ex-vessel storage tank	December 1976
Auxiliary handling machine	March 1977
Auxiliary handling machine floor valve	July 1977
Ex-vessel storage tank	December 1976
Ex-vessel transfer machine	July 1977
Ex-vessel transfer machine floor valve	July 1977
Core component pot	April 1977
Refueling instrumentation and control computer software	March 1977
Ex-vessel transfer machine gantry-trolley	December 1976
Reactor service building plug storage facility	July 1977
FHC gas cooling grapple	November 1976
FHC maintenance station	November 1977
New core assembly inspection equipment	September 1980
New fuel unloading station	September 1980
EVHS floor service stations	December 1977

5.3.3 Final Design

Final design of each major item of equipment began after preliminary design and extended to release for procurement. The final design effort was not completed until the results of development testing were available. See Section 5.3.4. Final design releases of the major refueling system equipment were as follows:

Item	Final Design Release Date
In-vessel transfer machine	June 1977
Ex-vessel storage tank major components: storage vessel, guard tank, closure head, turntable	April 1977
FHC gas cooling grapple	July 1977
FHC roof components	March 1977
FHC wall and floor penetrations	April 1977
Auxiliary handling machine	May 1978
Auxiliary handling machine floor valve	June 1978
Ex-vessel transfer machine	January 1979
Ex-vessel transfer machine floor valve	January 1979
Core component pot	January 1978
FHC maintenance station	June 1979
FHC equipment transfer lock	June 1979
IVTM panels	March 1978

5.3.4 Development Testing

In support of the RRS design, several features were tested to provide confidence in the design or to develop solutions to particular problems as follows:

<u>Test</u>	<u>Purpose</u>	<u>Remarks</u>
IVTM drive components	To determine the ability of the IVTM drive components to meet design loading and life requirements and to confirm manufacturers' performance data.	System performance of the grapple hoist and stem/grapple rotary drive was measured. It was determined that the components met all specified operational requirements.
IVTM grapple	To establish grapple wear characteristics and ability to operate under misaligned conditions.	Grapple functions properly under normal and misaligned conditions and exhibits little wear with prolonged use.

<u>Test</u>	<u>Purpose</u>	<u>Remarks</u>
IVTM seal and bushing	To demonstrate the seals' and bushings' abilities to operate under reactor refueling conditions, and to establish seal temperature limits (see Refs. 35, 36, and 37).	Aluminum-bronze bushing and elastomeric seals qualified for operation. Seal temperature limit established at 375 degrees.
IVTM prototype	To determine the ability of the IVTM to perform its tasks under simulated refueling conditions.	Performance testing of the in-vessel transfer machine in a sodium environment simulating refueling conditions was completed in June 1980 at the Energy Technology Engineering Center (ETEC). Core assembly exchanges equivalent to more than five refueling periods were performed during the tests, with the IVTM subjected to the sodium environment for about 130 days. The machine was operated remotely in both the automatic and manual modes from its control consoles located in a separate building. The tests were successful in demonstrating reliable machine operation, providing valuable information for the operations and maintenance manual, and identifying minor problem areas that were corrected during refurbishment. Sodium and sodium vapor deposits were removed from the IVTM by using ETEC alcohol cleaning process.

<u>Test</u>	<u>Purpose</u>	<u>Remarks</u>
AHM grapple	To verify proper kinematic functioning of the grapple to ensure that accidental release of the payload is not possible for any conceivable malfunction.	The grapple was tested under all postulated operational conditions. It showed no signs of excessive wear or deformation. It was verified that design of the grapple ensured positive retention of the payload.
EVTM chain/tape	To determine if chain or tape and their drive systems will operate in a sodium environment without malfunction or excessive wear. Previous testing had shown problems with sodium buildup and wear of the chain drive wheel.	A 3/8-in. chain was tested for a simulated 5-yr duty cycle with a front-wheel-drive grapple drive system. The results indicated that the chain operated successfully and would attain a duty cycle of at least 10 yr before replacement.
CCP siphon	Water tests were used as a design aid to develop the present design and showed it to be adequate. Tests will determine the adequacy of the design with sodium at refueling conditions.	A prototype CCP and siphon were tested in sodium. The test results indicated that the siphon is self-priming and will drain the CCP to the desired level before the CCP reaches the first seal location.
EVTM heat transfer	To demonstrate the heat transfer capability of the cold wall.	Heat transfer is adequate for core assemblies with decay heats of 20 kW or less. (EVTM design requirement is 20 kW. Frequent cleaning of the cold wall is not required.)

<u>Test</u>	<u>Purpose</u>	<u>Remarks</u>
FHC mockup	To check out prototype FHC equipment, assist in developing the cell arrangement, and provide FHC operator training.	Testing has demonstrated that a spent core assembly can be satisfactorily loaded from the FHC into the shipping cask under conditions of maximum misalignment. The tests demonstrated the capability of loading the spent-fuel shipping cask with a simulated gas cooling grapple.

5.4 Key Problems and Resolutions

5.4.1 In-Vessel Transfer Machine (IVTM) Problems Experienced During Performance Testing

The in-vessel transfer machine (IVTM), part of the CRBRP's in-vessel fuel handling system, was designed and built by the Energy Systems Group (ESG) of Rockwell International.

In May 1979, the IVTM was delivered to the Energy Technology Engineering Center (ETEC) for testing first in a dry ambient environment and then in a sodium environment simulating reactor conditions expected during refueling.

These tests were performed by ETEC between May 1979 and September 1980. The results of the tests, problems experienced, and corrections made are documented in Refs. 38 and 39.

Two problems encountered in these tests and the corrective actions taken are described in this report because these problems were associated with off-the-shelf purchased equipment commonly used in the industry and may be of benefit to possible future users of such equipment in avoiding similar difficulties.

5.4.1.1 IVTM Grapple Hoist Motor Controller Problems and Corrective Actions

The IVTM grapple hoist drive employs a pneumatic load control system as the primary method for controlling the load on the core assembly and for shutting off the drive motor. Load cells in series with the pneumatic load control system are used to monitor the load and serve as a backup for shutting the motor off. The basic principle upon which this system operates is to couple the motor to the driven core assembly through a pressure-controlled air column (pneumatic cylinders) in order to provide precise load control and the compliance necessary to isolate the core assembly from excessive forces that could be generated by the hoist motor during the period between the command to stop and motor deenergization.

The design of the pneumatic load control system is such that it is armed to protect the load in the direction of core assembly/grapple travel (raise or lower) and the controls preclude reversing the direction of travel unless first the pneumatic system is armed in that direction. Any uncontrollable motor reversal, as was the case with a malfunction of the motor controller experienced during IVTM testing at ETEC, will in effect bypass all the protective features designed into the machine and could cause substantial damage.

A chronological description of the problems encountered with the motor controller and the corrections made to preclude future problems are given below.

The motor and motor controller used on the IVTM grapple hoist drive system were procured from WER Industrial Corporation, Division of Emerson Electric Company, Grand Islands, New York.

During ETEC testing of the IVTM in the period July through December 1979, the grapple hoist controller exhibited numerous Instantaneous Fault Trips (IFT). When IFT occurs, the hoist drive motor is automatically stopped by the controller and can only be restarted by opening the controller enclosure door and actuating the IFT reset switch. In all cases of IFT, the cause of the trip had "cleared" by the time the reset procedure had been implemented.

As the problem of IFT was more of a nuisance than hazardous, testing was continued while communications took place with the controller supplier (WER Industrial) and its local service representative (Devoe Engineering).

The position taken by the supplier's field engineer was that the problem was temperature-related due to the borderline high ambient temperature inside the test facility (100-105°F versus 104°F rating). WER suggested to first have the local service technician attempt to isolate the cause of the IFT by taking measurements and observing oscilloscope patterns. Accordingly, ESG contracted with Devoe to perform this analysis. During the first visit, the technician did not find any faulty components but did determine "incorrect" settings of field voltage and armature current limit which he adjusted. Adjustments were made to obtain the proper operational parameters, and the grapple hoist was run several times with no IFT problems.

Testing was resumed and, shortly thereafter, several IFTs were again experienced. Devoe was again contracted to attempt to desensitize the IFT circuit by increasing the size of a capacitor across the gate of a Silicon Controlled Rectifier (SCR) in the IFT circuit. After increasing this capacitance by a factor of 5 to 0.54 μ f and blowing warm air across the general area of the IFT circuit, the hoist drive was repeatedly tested without IFT occurrence. Dry checkout testing of the IVTM was completed without any further IFT problems.

The test facility was loaded with sodium and performance testing of the IVTM started on December 14, 1979. IFT occurrences again slowly built up until their frequency per shift increased to the point where it was decided to have the factory and a local WER Service Representative on board once again but with the provision that the IFT problem be thoroughly analyzed to ascertain and implement a permanent fix. Preparations were made to have the diagnosis and fix made during the next scheduled shutdown of the test.

On January 23, 1980, which was just prior to the time planned for the diagnosis and fix, ETEC experienced an anomaly whereby the hoist drive had momentarily reversed its direction of travel.

On January 24, 1980, investigation by ESG Engineering, including numerous attempts to duplicate the reversal, failed to reveal any problem areas or repeats of the reversal. Shortly thereafter, on the same day, the grapple was in the down-drive mode when a reversal again occurred driving the grapple upward and thus bypassing the pneumatic load protection shutoff circuit, which was armed to protect for the commanded down-drive mode. Upon reaching full up, the hoist drive was automatically shut off by the backup grapple load cell circuit.

After making several basic measurements and observations to determine possible mechanical damage to the hoist drive components, the high load was relieved and the grapple carriage assembly was lowered via manual cranking of the hoist drive motor. Although the load measured at the time of the incident was more than twice the load cell comparator setting, none of the components were damaged. The Devoe and WER Industrial people were contacted, and the local service technician was hired to determine the cause of the reversal.

The investigation showed that a printed circuit board in the controller called "Accel/Decel" board (1074-097) had failed by ignoring the direction signal sent by the control console and generated its own arbitrary, oscillating direction signal. The board was removed and sent to the supplier for further analysis, and a failure analysis report was requested (see Ref. 40 for the supplier's findings and recommendations). ESG also requested that the replacement board be burned in for 168 hours to preclude a reversal problem due to infant mortality.

In order to perform analysis of the recurring IFT problem, the supplier sent Mr. Lyle Eveleigh, the Field Service Engineer from the St. Louis regional area. Mr. Eveleigh spent 2 days, February 11 and 12, 1980, and determined

that the IFT nuisance tripping was primarily due to improper adjustments of the IR Comp and Current Limit potentiometers. The supplier recommended the replacement of the entire regulator control board on the basis that the new board was much improved. He also pointed out that the old board, having been repaired in the past, may have had some incipient failure pending.

Concurrent with the factory failure analysis of the Accel/Decel board, ESG communicated with WER Industrial our desire to make the controller "fail-safe" to the extent that it cannot unintentionally reverse direction of the hoist drive. As a result of several discussions with WER, they proposed a modification package that may be added on the controller with the wiring changes to be done internal to the controller enclosure. The basic objective of the circuit is that the command signal polarity (which determines the direction of hoist travel chosen by the operator) is compared to the armature polarity which directly indicates the direction of motor rotation and, hence, hoist travel direction. If the signals do not agree, relay contacts will open the run (or jog) circuit, thereby dropping out the armature contactor. The relay logic in the modification is set up such that if any of the relays fail to an "incorrect" configuration, the hoist is prevented from starting.

An antireversal interlock module was procured from the motor controller supplier, along with a new properly burned-in regulator control board and Accel/Decel board. These items were installed and testing of the IVTM was successfully completed without any further IFT or motor-reversal problems.

From the experience attained during testing of the IVTM, it is recommended that motor controllers used on machinery similar to the IVTM include the anti-reversal interlock. There is no evidence that IFT occurrences were in any way connected with, nor led to the Accel/Decel board failure, but rather was a separate problem attributed to an older, inferior model, regulator control board, which has been once repaired and improperly adjusted.

5.4.1.2 IVTM Holddown Sleeve Slip Clutch Problem and Correction Action

The IVTM holddown sleeve-drive system incorporates a spring-adjustable, friction-type slip clutch between the drive motor and the gear reducer for overload protection of the components downstream from the gear reducer. The slip clutch used is Model L4-1-322-XA purchased from the Hilliard Corporation, Elmira, New York.

During testing of the IVTM, the test performer, ETEC, reported that the slip clutch operated noisily, the set screw, which retains the slip clutch position, loosened on several occasions and, on one occasion, the friction plate gears disengaged from the coupling sleeve.

Investigation of the slip clutch problems and review of the supplier's technical literature resulted in the following findings.

The motor shaft to the gear reducer alignment (lateral and angular) exceeded the maximum permissible values recommended by the supplier of the clutch. The shaft misalignment, predominantly the angular misalignment, creates a condition whereby the friction plate gear teeth meshing with the internal gear of the coupling sleeve would axially oscillate during shaft rotation. This oscillation created the noise experienced and generated an axial force (due to friction at the gear teeth), which eventually overcame the holding action of the set screw and caused the separation of the slip clutch.

After completion of the IVTM dry ambient tests and before the in-sodium tests started, the motor/gear reducer assembly was removed from the machine and the shafts were realigned to meet the suppliers specified limits. Additionally, the slip clutch was modified by adding a set screw hole at 90 degrees from the existing set screw hole. The torque setting was bench-adjusted and the slip clutch reinstalled. To preclude possible set screw loosening, back-to-back set screws were installed at each set screw hole (the second set screw provides the locking feature for the first).

The modified slip clutch and realigned shaft assembly were reinstalled on the machine and performance tests of the IVTM (equivalent to five reactor refuelings) were completed without encountering the slip clutch problems experienced earlier. The slip clutch noise, however, although reduced noticeably, could not be eliminated completely.

Based on the experience with the type of slip clutch used on the IVTM, the following is concluded:

- The mating shafts must be aligned very precisely, otherwise the clutch will operate noisily, excessive wear of the clutch gear teeth can be expected, and the clutch may separate unless the two clutch halves are securely locked.
- Locking the clutch with "clamp-tite" shaft collars at each end of the clutch is recommended. Where space is not available for locking collars, the alternate method is to use two set screw holes and back-to-back set screws as was done on the IVTM.
- The noise exhibited by this type of a clutch can be expected since perfect alignment of the shafts is difficult to attain and, in most practical applications, impossible to achieve.

5.4.2 Ex-Vessel Transfer Machine (EVTM) Problems and Resolutions

5.4.2.1 Design Changes and Code Verification Tests Resulting from Heat Transfer Test Problems

Initial heat transfer calculations were based on the analyses performed during the design of the closed loop ex-vessel machine (CLEM) for the Fast Flux Test Facility (FFTF). These calculations indicated the CRBRP ex-vessel transfer machine (EVTM) design was marginal to handle 20-kWt fuel assemblies. To confirm the analytical model, a subscale parametric heat transfer test was

performed (Ref. 41). The test results showed that, besides the convective flow through the fuel assembly, a series of secondary convective loops from the outside of the fuel duct to the wall of the core component pot (CCP) exist to assist the heat transfer in the sodium-filled CCP. Predictions indicated that an FFTF-size CCP was still marginal and that a larger diameter CCP would enhance the overall thermal performance. The CCP tubing was increased from 6-5/8 to 7-1/4 in. OD.

A full-scale test of a combined EVTM and CLEM fuel assembly was performed which confirmed the EVTM design (Ref. 42). The thermal model of the fuel assembly in the CCP was utilized for the EVTM design.

The coupon emissivity (Ref. 43) and subscale emissivity tests (Ref. 44) investigated the emissivity of the CCP and cold wall and the possible changes in emissivity due to the deposition of sodium and sodium oxide deposits. The results indicated that, during normal transfer times, the sodium would evaporate off of the CCP. In addition, the sodium buildup on the cold wall would not significantly affect the overall conductance over a 1- to 2-year operating cycle, and the frost film could be effectively removed by cold wall heatup with the cold wall heaters.

5.4.2.2 Prevention of Unintentional Grapple Actuation

The CRBR EVTM grapples contain a positive latching feature to preclude inadvertent grapple actuation and potential dropping of a spent fuel assembly. The grapple contains a locking finger which prevents movement of the actuating rods and cams. The locking finger is pivoted out of the way when the grapple is set down. With the locking finger retracted, the actuating rod may be pulled up or lowered, moving the grapple fingers in or out. Once the grapple is elevated (about 1 in.), the locking finger is pivoted back to the lock position, preventing actuation rod motion (Ref. 45).

The grapples and grapple drive were fabricated early in the program to allow testing of the positive latching feature of the grapples in sodium. The testing is documented in Ref. 46. As documented in the test report, the grapple operated as designed with one drawback. Once either of the grapples was immersed in sodium and the sodium allowed to solidify, the locking finger would not move, and the grapple fingers could not be retracted. Each morning prior to initiation of the day's testing, the grapple had to be soaked in sodium to unfreeze the sodium in the grapple to allow proper operation. Westinghouse and Project Office personnel were aware of the difficulty but considered that it would be an insignificant operating penalty.

5.4.2.3 Break-Away Joint Design to Accommodate Seismic Response Spectra Changes

The increasing seismic response spectra changes resulted in a negative design margin in the bolted joint between the reactor head and the reactor fuel transfer port (RFTP) adapter when the EVTM was coupled to the floor valve on the adapter. The break-away or slip joint was devised as a means to limit the lateral load on the floor valve and the RFTP adapter. The slip joint, as described in Ref. 47, is located between the extender and the closure valve. Shear pins are used to position the closure valve on the extender. If a seismic event occurs while the closure valve is mated to a floor valve, the shear pins may fail, limiting the load imparted on the floor valve.

5.4.2.4 Gantry Trolley Seismic Restraint Redesign

Changes which occurred during the gantry design are documented in Ref. 48. Initially the seismic restraints used for the gantry and trolley were similar to those used for the FFTF CLEM transporter. Increased seismic spectra required the redesign of the seismic restraints to initially prevent gantry trolley vertical movement and finally offload the gantry and trolley wheels to prevent wheel failure during the seismic event. Additional seismic clamps were required to secure the EVTM viewpoint to the gantry beams (two clamps,

one on each side of the viewport) and two clamps to tie the cab to the bottom of the gantry beam. Tapered rails are used in the rail trench to increase the gantry restraint load capability. Tapered rails are also used on the gantry for the trolley seismic restraints. Flat (parallel-sided) rails are used underneath the gantry beams for the viewport and cab seismic restraints.

5.4.2.5 Syphon Feature to Control Sodium Dripping

The development of the core component pot syphon is described in Ref. 49. Initial thermal calculations indicated that thermal expansion of the sodium in the CCPs would overflow the CCPs and fill 15 drip pans in the EVTM, requiring five emptyings of the three drip pan pots into the fuel handling cell. As one exchange of pots into the FHC takes about one 8-h shift, reactor refueling would be extended by five shifts each year. A syphon was considered for use in the FFTF CCPs but was abandoned as unreliable. Initial testing was with an aluminum and plastic model to visualize the action in the syphon. Testing was performed in heated water (140°F) to simulate the viscosity of 400°F sodium. The initial test results were disappointing. Hoisting speed determined whether the syphon would work. With a slow speed, the syphons would not work at all. Normally, the EVTM grapple speed would be at 16 or 24 fpm. At these hoisting speeds, sometimes the syphon would not work. Watching the water flow through the syphon showed that sometimes the trapped air in the syphon would not be expelled by the water, and in fact some air would leak back into the passageway against the flow of the water, airlocking the syphon. Several modifications were tested. When the syphon exit hole was partially blocked, causing the exiting water to swirl and fill the port, blocking the inleakage of air, the syphon worked. Further testing optimized the exit port configuration. A stainless steel, full-size mockup of the CCP top was made and tested, first in water and then in sodium. The test was attached to the bottom of the ongoing EVTM grapple chain and tape tests. The in-sodium tests confirmed the design of the CCP syphons (Ref. 50).

5.4.2.6 Resolution of Problems with Grapple Drive Sprocket

Initial testing of the EVTM grapple support member, chain, or tape was to select the best support member for the EVTM. During fabrication and the initiation of the test, the reduction in size of the EVTM (about 60 ft tall) was investigated as a potential cost savings. In addition, the EVST and FHC were relocated outside the reactor containment building to the reactor service building. The hatch size between the buildings required as small an EVTM and gantry trolley as possible. At the same time, a rotating offset guide tube was selected to speed up refueling. The grapple support member (three chains or three tapes) will bunch and rub the side of the guide tube when the grapple is down in the offset position. The two edges of the tape rubbing (scrapping) the side of the guide tube was not a viable design. The chain drive system was, therefore, the viable design choice.

Following fabrication completion, the tape test weight was safety chained to the chain test weight in case of breakage of either the chain or tape to preclude damage of a dropped weight into the sodium test vessel. Testing was performed with the drives synchronized.

Results of the test are documented in Ref. 51. The initial 3/8-in.-diameter chain drive design, which was based on the chain vendor's recommendations, did not work with stainless steel chain, which was too soft for the much harder drive wheels. Utilization of the more standard alloy steel crane chain improved the performance somewhat, to about a 1-year chain life, which was much less than the 5-year minimum chain life goal. It was determined that the driving wheel should be the first wheel to minimize loaded chain articulations (one instead of seven), which should increase chain life seven times.

While the front drive wheel system was being designed and fabricated, a series of tests were performed on a commercially available 5/16-in.-diameter chain four-pocket wheel versus the reference EVTM drive of a 3/8-in.-diameter chain eight-pocket wheel. These tests confirmed that the four-pocket wheel wore

more rapidly than an eight-pocket wheel would have, and that clean chain or chain with the vendor's standard lubricant wore out at about the same rate. As soon as the chain wore through its case-hardened surface, extremely high wear rates occurred. Lubrication with commercially available LPS-3 lubricant decreased the wear rate by a factor exceeding 25.

Wear patterns of the initial chain test indicated that chain which had been immersed in sodium had a much lower wear rate than clean-dry chain, i.e., sodium exhibits some lubricative properties.

During the pocket wheel redesign, enlarged layouts of the pocket wheel and chain were made. It was determined that the vendor's computer-code-designed pocket wheel resulted in the chain load being taken by the first pocket. As the chain begins to wear, it tends to ride up and out of the pocket, loading the front end of the pocket sprocket, causing local deformation of the wheel. By reducing the pitch diameter slightly, the last pocket could be made to take the load. As chain wear occurs, the longer chain would cause successive pockets to take the load until ultimately the situation reverts back to the original vendor configuration. Longer chain life should be expected.

Testing in air with the new front wheel drive showed excessive chain guide plate wear accompanied by chain pileup and bunching. The cloverleaf chain stripper was removed, and a bayonet-style stripper was installed. A 100-lb counterweight was also installed to assist in chain removal from the pockets. The utilization of LPS-3 lubricant corrected the chain guide plate wear problem. Prior to sodium testing, it was determined that the guide plate had been heat treated to the original design hardness rather than the new design hardness. A new pocket wheel and chain guide (reheat-treated) were installed in the test fixture. Testing of the unlubricated chain over a 5-year operating life indicated the chain wear life to be in excess of 11 years.

The prototype unit design specified that the chain would be prelubricated with LPS-3. Tests had shown the LPS-3 to be compatible with sodium, the sodium

dissolving the LPS-3, resulting in a slightly higher carbon content from the paraffinic lubricant residue.

Testing of the prototype chain drive indicated no chain or drive problems over the 1-year refueling testing. Examination of the pocket wheels and guides also indicated no wear problems. The only evidence of wear was a slight polishing of the guides.

5.4.2.7 Resolution of Problems with Inflatable Seals

Based on the difficulties of the FFTF CLEM inflatable seals, the seals for the CRBRP were changed from the FFTF design. A larger groove was selected, as well as adoption of a different style inflation stem. These changes were made based on the development recommendations documented in Ref. 52.

Significant difficulties were encountered developing a list of potential bidders. The inflatable seal vendors were busy with other commercial work and did not want to get involved with a specialty item. Two potential bidders were selected who were willing to bid on the CRBRP inflatable seals. During the bid cycle, one of the bidders felt he could not meet the specification requirements, resulting in a sole source situation and contract.

The vendor initially made two sets of two sizes of seals for qualification testing. The rubber batch from our supplied B. F. Goodrich recipe for the seals exhibited a higher than desired heat resistance hardness, which is an indication of what the seal's properties may be if subjected to a higher than normal operating temperature for an extended period of time. Also, there were some areas of nonbonding where the base of the seal was bonded to the main bladder of the seal. These were small areas along the lip of the base. The testing showed no increase of separation or peeling of the base from the seal. Performance data from the seal test indicated that the seals performed extremely well. After the test rig was debugged, at no time was the specification limit for leak rate even approached. The supplementary tests,

especially, showed the seals to be very effective over a broad range of gaps, tilts, internal seal pressures, and even after heat aging (Ref. 53). Destructive tests of the seals (sectioning) showed the seals to be thinner in the sidewalls than specified on the drawing (0.07 in. drawing versus 0.031 in. at one point on the seal). Discussions with the vendor indicated the seal could be thickened up, but the seal would not be as flexible and would be much more susceptible to fatigue cracking.

Further discussions with the FFTF CLEM personnel indicated that FFTF performance tests the seals as they are received from the vendor. About 20% of the pairs of seals fail (leak) and are scrapped. Our vendor agreed to perform the FFTF tests at his facility prior to acceptance for CRBRP usage.

Testing of the rubber from our supplied recipe prepared for the production run showed similar results to the batch used for the qualification test seals. A second rubber batch was prepared with, again, similar results. The seals operate at ambient conditions. An accident, subjecting the seals to a higher temperature for an extended period of time, would be an upset or emergency condition. The qualification seals were heat aged during the testing process and did become harder (more rigid) but still functioned (sealed) perfectly. Based on the above, it was decided to utilize the second batch of rubber for the plant unit inflatable seals. Every seal produced for the order passed the "FFTF performance test." The seals were packaged and stored in the vendor's cold room until required for usage in the CRBRP fuel handling equipment.

5.4.3 Inservice Inspection

5.4.3.1 Design Status of Articulating CCTV System for Inservice Inspection of CRBRP Component Welds and Pipe Supports

The responsibility for the design, fabrication and delivery of the articulating closed-circuit television (CCTV) system equipment for remote inservice inspection (ISI) of welds of the reactor inlet/outlet downcomers welds, primary sodium pumps (PSP) welds, intermediate heat exchangers (IHX) welds and

pipng supports, snubbers/hangers within the primary heat transport system (PHTS) cells were initially assigned to the Energy Systems Group (ESG) of Rockwell International.

As part of the CRBRP consolidation, about the middle of Fiscal Year 1982, the customer (Westinghouse) directed that ESG finish the conceptual design of the CCTV ISI system and on completion turn over the responsibility for the preliminary design, final design, fabrication, and delivery to Westinghouse.

Accordingly, the conceptual design of the CCTV ISI equipment was completed and the layouts, reports, and specifications were delivered to Westinghouse in December 1982.

Since the subsequent design stage was under Westinghouse control, this report is limited to briefly describe the equipment status and problem areas as of the end of the conceptual design phase.

The documents of the remote viewing articulating CCTV system submitted to Westinghouse describe the conceptual design of the following components:

- 5.5-in.-diameter CCTV articulating column assembly
- Control and monitoring console
- Transporter/handling machine and auxiliary platform
- Floor valve, gas-lock, and installation/removal bags
- PSP port adapter.

The requirements, functions, and detail description of the components listed above are given in Ref. 54, and the designs are shown in Ref. 55 through 61 layout drawings.

The following provides some of the background history that led to the selection for ISI use on CRBRP, the viewing system described in Ref. 54.

Prior to starting the conceptual design, several studies had been performed to identify candidate remote viewing CCTV concepts that could possibly satisfy the requirements for remote inservice inspection of the reactor vessel inlet/outlet pipe and guard pipe welds, the PSP tank and guard vessel welds, the IHX shell and guard vessel welds, and the PHTS pipe supports, snubbers, and hangers. ISI equipment access to the areas requiring this inspection is provided through a number of 6-in.-diameter floor penetrations inside the reactor containment building (RCB).

A concept evaluation trade study, Ref. 62, and a subsequent geometry study, Ref. 63, were performed to identify the most promising candidate concepts that could provide essentially 100% inspectability of all the welds as required by Section XI, Division 3, of the ASME B&P Vessel Code, establish the articulation motions required and dimensional limits that must be imposed on each concept to permit maneuverability into restricted spaces. These scoping studies resulted in the recommendation to pursue the following three concepts and to evaluate in more detail their capability to satisfy the inspection requirements of the ASME code.

CONCEPT I--5.5-IN.-DIAMETER CCTV ARTICULATING COLUMN ASSEMBLY

The TV camera of this concept is equipped with a zoom lens and the lighting required to inspect objects at distances up to 22 ft away from the TV camera. To house the zoom lens and the light, a 5.5-in.-diameter envelope is required. The TV camera can be articulated along a vertical plane through a 180-degree arc. The plane of articulation can be changed by rotating about the vertical axis of the column assembly, and, with the capability of mirror rotation about the axis of the TV camera, provides universal viewing capability.

CONCEPT II--4.0-IN.-DIAMETER CCTV ARTICULATION COLUMN ASSEMBLY

This concept is identical to Concept I except the TV camera is not equipped with a zoom lens, and articulation along a vertical plane is limited to a 90-degree arc. The capability of this concept for inspecting welds is limited to a distance of up to 2 ft away from the camera but, because of its smaller envelope, it has greater maneuverability in tight spaces (reactor inlet/outlet downcomer bends) than Concept I.

CONCEPT III--TRACK-GUIDED CCTV ARTICULATING TRANSPORTER

This concept has the same viewing capability as Concept II. It has been included for further evaluation as a competitor to Concept II because of greater maneuverability and, hence, ability to reach areas not accessible to Concept II. The description and method of articulation of this concept is given in Ref. 64.

As a result of the preliminary evaluation performed in Ref. 63, it was concluded that Concept I would satisfy the ISI requirements for the piping supports, snubbers, and hangers in the PHTS cells, but the requirement for inspection coverage of essentially 100% of the reactor vessel inlet/outlet pipe/guard pipe welds, the PSP tank/guard vessel welds, and the IHX shell/guard vessel welds could not be attained.

Based on these findings, a detail study was performed to determine the optimum combination of remote viewing concepts that would be required to satisfy the weld inspectability requirements of the ASME code. Ref. 64 study lists the weld inspectability percentages obtained using Concept II or Concept III in combination with Concept I. Based on the data developed in Ref. 64, it was concluded that additional floor penetrations are required at the PSP area, the IHX area, and the reactor vessel inlet/outlet downcomers, and the ISI system should include both Concept I and Concept III.

This recommendation was presented to the customer (Westinghouse) in November 1981. The customer rejected the recommendation of Ref. 64 because the adaptation of Concept III would impact already fabricated reactor components and the cost of the redesign, coupled with the cost of the additional floor penetration, would be unacceptably high.

The customer requested that ESG redo the study and determine new weld inspectability percentages using Concept I alone without the additional floor penetrations and to reduce the viewing angle (between the TV camera line of sight and the surface being inspected) from the 30 degrees minimum used in the previous study to zero degrees.

The study was redone using the new ground rules, and the results are documented in Appendix A of Ref. 64. The conclusions reached in the earlier study were not changed by the new weld inspectability data developed in the restudy; 100% weld inspectability could not be attained even when Concept II was considered in those areas where weld inspectability of Concept I was very poor.

The results of the restudy, (Appendix A of Ref. 64) and the following list of problems were presented to the customer (Westinghouse) in April 1982.

List of problems requiring resolution:

- Section XI, Division 3, of the ASME B&P Vessel Code states that "essentially 100% of the weld" shall be inspectable for VTM-2 and VTM-3 acceptance standards. ESG interprets this to mean 100%, and this was the basis for all the studies and resultant recommendations. The interpretation of the word "essentially" is at issue, and a definition of what is exactly acceptable is required.
- Some of the floor penetrations in the reactor head access area (HAA) are blocked by overhead obstructions such as the EVTM gantry beams

and personnel platforms. Modifications to the above HAA components are needed in these areas to provide access for ISI of the reactor vessel inlet/outlet downcomers.

- A weld address system needs to be prepared, and the components should be marked accordingly adjacent to the welds requiring remote ISI. The weld-addressing markings are required to provide the means for navigating the articulating ISI apparatus and for identification/recording of the area being inspected.
- The orientations of the image displayed on the TV monitor will constantly vary as the TV camera is articulated into position. This condition would be very confusing and could disorient the ISI equipment operator. Therefore, it is recommended that a visual aid, such as a plastic model duplicating the welds and weld-addressing markings of the area being inspected, be provided.

Upon reviewing the data presented, the customer directed ESG to proceed and complete the conceptual design of Concept I only. The rationale for selecting an ISI system with less than 100% weld inspectability and the resolution of the problems delineated above will be undertaken by Westinghouse.

The above background history is presented here to highlight the importance of considering the ISI requirements very early in the design of the reactor plant components. The inclusion of ISI requirements during the early stages of the Plant design would avoid accessibility and interference problems, ensure that component inspectability to satisfy the requirements of the ASME code are met, and enhance the navigation and data interpretation of remote-viewing articulating devices.

5.5 Final Design Status

5.5.1 Status at Project Termination

5.5.1.1 General System Status

At the time of project termination, the basic design of the reactor refueling system was complete except for spent fuel shipping cask (SFSC) related equipment. All sections of the SDD had been baselined, all major equipment items were ready for procurement (some were delivered, some on order, and some delayed by the National Policy Debate and/or lack of funding), and very few holds remained in System 41 documentation (except for SFSC-related equipment).

The final working system design review (FWSDR) was held on August 27, 1981, for all parts of System 41 except for the SFSC-related equipment. The SFSC was to have been provided by the Department of Energy (DOE). Even though the cask to be provided had not been fully definitized, certain features necessary to complete the design of CRBRP interfacing systems had been specified by means of interface control documents. Other features of the SFSC needed to complete the design remained on "hold" in the SDD and other documents.

5.5.1.2 Documentation Status

At project termination, the published version of the System Design Description, SDD-41, was Revision 27 (Ref. 1). All sections and appendices of the SDD were baselined including Sections 3 and 6.

The principal drawings and equipment specifications are listed in Figure 5-14. In general, only the installation requirements documents and several operation and maintenance manuals remained to be prepared.

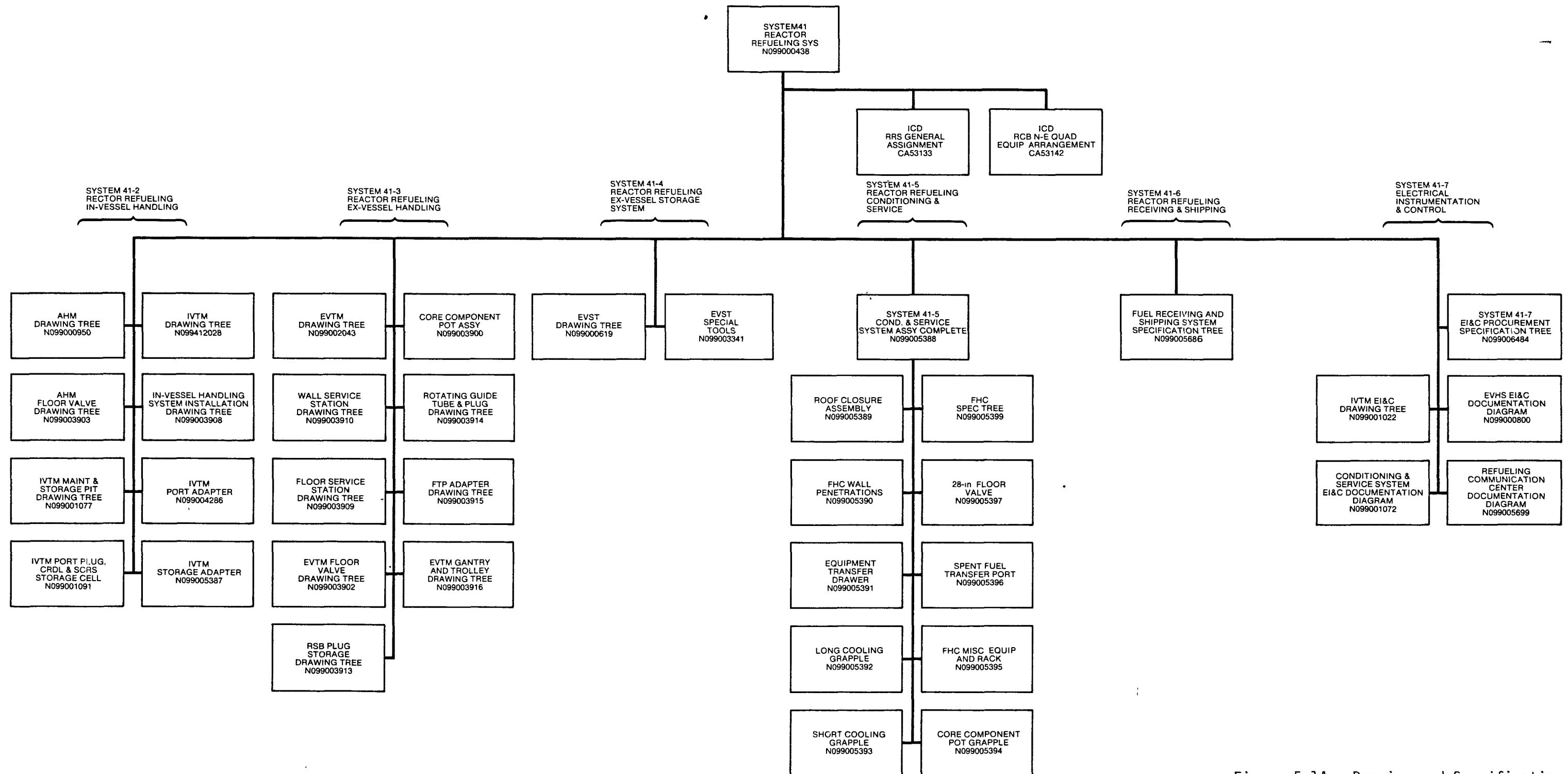


Figure 5-14. Drawing and Specification Tree for the Reactor Refueling System

5.5.1.3 Supporting Documentation and Reference Sources

The general design bases for the System 41 design and components are contained in the system design description (Ref. 1) and design reports, Refs. 6 through 33. These documents, particularly the design reports, provide all requirements and their verification, the overall rationale, and experience on which the designs are based, and also reference the supporting system and component documentation generated during the design.

Ref. 65 gives the source of requirements stated in SDD-41. The principal interfaces between System 41 and interfacing systems are given in Refs. 66 through 69.

5.5.1.4 Component Status

The component status at project termination is summarized in Table 5-2. As can be seen, a large number of components had been delivered or were nearly complete. In addition, several other items of equipment, Table 5-3, had been ready for procurement, but procurement was delayed, first by the National Policy Debate and later by funding restrictions.

5.5.2 Outstanding Approved Changes

At termination, there were five approved Class 1 ECPs that had not been completely incorporated into project documents. They are as follows:

A41358	Delete Requirements for System 91 Plant Data Handling and Display System to Store System 41 Computer Data
A41493	Update SDD-41 Primary and Secondary Interfaces
W1765	Revision of System 90 Plant Annunciator System Interface Control Documents

TABLE 5-2
REACTOR REFUELING SYSTEM COMPONENT STATUS
(Sheet 1 of 5)

Component	Item	Status/ Delivery Date	Vendor
In-vessel transfer machine	Drive section, in-vessel section, control consoles	Complete/ May 1979	Atomics International
Ex-vessel transfer machine	Grapple and grapple drive	Complete/ April 1982	Custom Machine
Ex-vessel transfer machine	Cask body, cold wall drip pan ports, etc.	90% complete	Custom Machine
Ex-vessel transfer machine floor valves	Three floor valves	90% complete	Custom Machine
Gantry-trolley system	Cable track system	Complete/ September 1982	Thames Valley Steel
	Cables for cable track	Complete/ April 1982	Boston Wire
	Gantry-trolley	90% complete	Ederer Corporation
Wall and floor service stations	Pneumatic and electrical hardware	Complete/ February 1983	Newbrook Machine Corporation
EVTM, AHM, FHC, and RSB plug storage	Inflatable seals	Complete/ October 1983	Seal Master Corporation
Core component pot	Long-lead material (tubing)	Complete/ May 1977	Trent Tube

TABLE 5-2
REACTOR REFUELING SYSTEM COMPONENT STATUS
(Sheet 2 of 5)

Component	Item	Status/ Delivery Date	Vendor
Ex-vessel storage tank (Complete items delivered to interim storage)	Bearing supports and gear	Complete/ March 1978	Kobe Steel
	Storage tank	Complete/ June 1982	Babcock & Wilcox
	Guard tank	Complete/ June 1982	Babcock & Wilcox
	Closure head	Complete/ June 1982	Babcock & Wilcox
	Turntable	Complete/ June 1982	Babcock & Wilcox
	Bearing and gear assembly	Complete/ June 1982	Rotek, Inc.
	Bearing match plate	Complete/ June 1979	American Bridge Company
	Embedment match plate	Complete/ February 1980	Hitco
	Storage tube and preheat tube (partial order)	Complete/ June 1982	Joseph Oat Company

TABLE 5-2
REACTOR REFUELING SYSTEM COMPONENT STATUS
(Sheet 3 of 5)

Component	Item	Status/ Delivery Date	Vendor
Ex-vessel storage tank (Complete items delivered to interim storage) (continued)	Pinion shaft assembly (partial order)	Complete/ June 1982	Process Equipment Company
EVST shipping and han- dling equipment (Complete items delivered to interim storage)	Guard tank skid assembly	Complete/ June 1982	Babcock & Wilcox
	Guard tank lift beam assembly	Complete/ June 1982	Babcock & Wilcox
	Storage vessel skid assembly	Complete/ June 1982	Babcock & Wilcox
	Storage vessel lift beam assembly	Complete/ June 1982	Babcock & Wilcox
	Turntable skid assembly	Complete/ June 1982	Babcock & Wilcox
	Closure head lift beam	Complete/ June 1982	Babcock & Wilcox
	Closure head shipping container	Complete/ June 1982	Babcock & Wilcox
	Upender	Complete/ June 1982	Babcock & Wilcox
	Upender support	Complete/ June 1982	Babcock & Wilcox

TABLE 5-2
REACTOR REFUELING SYSTEM COMPONENT STATUS
(Sheet 4 of 5)

Component	Item	Status/ Delivery Date	Vendor
EVST shipping and handling equipment (Complete items delivered to interim storage) (continued)	Optical spider	Complete/ June 1982	Babcock & Wilcox
	Bearing assembly and bull gear handling equipment	Complete/ June 1982	Rotek, Inc.
	Remaining turntable and encoder drive system	Complete/ October 1983	High Vacuum Equipment Corporation
Fuel handling cell	Crane and powered manipulator	Complete/ January 1978	Programmed and Remote Systems
	Master/slave manipulators	Complete/ January 1978	Central Research Laboratories
	Viewing windows	Complete/ December 1983	Nuclear Pacific, Inc.
	Spent fuel transfer port assembly	Complete/ September 1983	Newbrook Machine Corporation
	28-in. maintenance valve	70% complete	Lakeside Bridge & Steel
	FHC penetrations	Complete/ November 1983	Excelco Developments, Inc.
	Wall periscope	80% complete	Nuclear Pacific, Inc.

TABLE 5-2
REACTOR REFUELING SYSTEM COMPONENT STATUS
(Sheet 5 of 5)

Component	Item	Status/ Delivery Date	Vendor
Core assembly receiving and shipping	New core assembly inspection equipment	Complete/ June 1982	Newbrook Machine Corporation
	New-fuel unloading station	Complete/ December 1982	Newbrook Machine Corporation
Refueling instrumenta- tion and control	Computer system	Complete/ September 1978	Hewlett-Packard
	IVTM control panels	Complete/ February 1977	JPW Industries
	EVTM control consoles	Complete/ September 1981	Remco
	EVTM motor control center	Complete/ September 1981	Wesco
	EVTM logic panel	Complete/ September 1981	Wesco

TABLE 5-3
 REACTOR REFUELING SYSTEM EQUIPMENT READY FOR PROCUREMENT BUT
 DELAYED BY NATIONAL POLICY DEBATE OR FUNDING RESTRICTIONS

Subsystem	Items
In-vessel handling system	Auxiliary handling machine and AHM floor valves, port and storage adapters, IVTM storage and maintenance facility, IVTM port plug storage facility, IVTM in-vessel spare
Ex-vessel handling system	Core component pots, plug storage facility
Ex-vessel storage system	Storage tubes (balance of order), striker plate, floor valve adapters, storage vessel embedments, site maintenance tools, valve pit components
Conditioning and service system	Maintenance and service station, argon circulation system (filters, prefilter, blowers, valves, and flanges), TV systems, lights, roof closure
Receiving and shipping system	None
Refueling instrumentation and control	Controls and panels for AHM, EVTM (balance of order), EVST, FHC, and argon circulation system

A41438 Spent Fuel Shipping Cask Interfaces

B25A-100 Location of System 41 Blowers and Exhaust Air Pits with Drains
and Relocation of EVST Equipment Air Correction

There were also eight approved Class 3 (minor) ECPs that had not been completely incorporated into project documents. They are as follows:

A41502 IVTM Operation and Maintenance Manual Assembly and Installation
Section Revision

A41515 Argon Circulation System Process and Instrumentation Diagram
Update

A41524 FHC Argon Circulation System Grill Requirement Addition

A41525 Reactor Fuel Transfer Port Adapter Modification

A41526 SFSC Travel Path

A41527 IVTM Storage and Maintenance Facility Layout Update

A41531 Lifting Fixture Redesign

A41533 IVTM Storage Adapter Floating Seal Ring

5.5.3 Outstanding Activities to Complete

The major item that had not been resolved in the fuel handling system is the SFSC. ECP A41438 established features that permitted the completion of design of CRBRP interfacing systems. However, a hold still exists in Section 2.2.6.3 of SDD-41 (Ref. 1) concerning details of the SFSC. Except for this aspect, the design of the CRBRP fuel handling system was essentially complete. Procurement of those items listed in Table 5.3 remained. This would, of course,

involve updating of some drawings and interfaces to be consistent with the as-built component and the writing of operation and maintenance manuals.

The EVTM, gantry-trolley, cable track, and related I&C were to have been assembled in the K-25 building at Oak Ridge for checkout prior to installation in the plant. This activity was in the planning stage at termination.

5.5.4 Recommendations for Future Design

The CRBRP fuel handling system is a well-developed design based on many years of experience with other reactors (see Section 5.3). Future designs would do well to not depart too much from these proven features without a thorough development program to support any necessary or desired changes. It is important in the early conceptual design phase to study all aspects of an integrated RRS (as was done for CRBRP in Ref. 34), and identify very early the key features and requirements and the options available to satisfy those requirements. Any needed development effort should be conducted as soon as possible.

One area that should be studied in detail to avoid future problems is the scheduling of the activities taking place in and around the fuel handling area inside the reactor containment building. Overall system studies of this nature may uncover serious problems of congestion and overutilization of equipment. These problems need to be identified and resolved before the plant arrangement becomes firm.

5.6 References

1. SDD-41, Revision 27, "System Design Description--Reactor Refueling System," July 27, 1983
2. WARD-D-0244, Rev. 3, "CRBRP Preservice and Inservice Inspection Requirements for Sodium and Cover Gas Systems," dated January 7, 1983

3. WARD-D-177, Rev. 14, "CRBRP ALARA Criteria and Guidelines," dated July 14, 1983
4. WARD-D-0037, Rev. 1, "Seismic Design Criteria for the Clinch River Breeder Reactor Plant" dated May 25, 1977
5. OPDD-10, Rev. 146, "Overall Plant Design Description," dated August 1, 1983
6. N099ER412005, IVTM Design Report, May 16, 1980
7. ER099412010, Auxiliary Handling Machine (AHM) Design Report, May 12, 1980
8. N099ER412014, AHM Floor Valve Final Design Report, August 22, 1979
9. N099ER412015, IVTM Port and Storage Adapters Design Report, January 6, 1982
10. N099ER412016, IVTM Port Plug and AHM Floor Valve Storage Cell Final Design Report, August 26, 1980
11. N099ER412007, IVTM Storage and Maintenance Facility Design Report, August 21, 1979
12. N099ER413021, Ex-Vessel Transfer Machine (EVTM) Final Design Report, July 25, 1980
13. N099ER413012, EVTM Gantry-Trolley Preliminary Design Report, September 11, 1979
14. N099ER413022, EVTM Floor Valve Final Design Report, August 31, 1979
15. N099ER413024, Core Component Pot (CCP) Final Design Report, September 24, 1979

16. N099ER413017, RSB Plug Storage Facility Preliminary Design Report, September 11, 1979
17. N099ER413018, Floor Service Station (FSS) and Wall Service Stations (WSS) Design Report, October 10, 1979
18. N099ER413019, Reactor Fuel Transfer Port (RFTP) Rotation Guide Tube (RGT) Preliminary Design Report, September 21, 1983
19. N099ER413020, Reactor Fuel Transfer Port (RFTP) Adapter Preliminary Design Report, May 9, 1980
20. N099ER414004, EVST Final Design Report, October 12, 1979
21. N099ER414003, EVST Special Tools Design Report, March 10, 1977
22. N099ER415026, Conditioning and Service System 41-5 Fuel Handling Cell Final Design Report, August 29, 1983
23. N099ER415027, Conditioning and Service System 41-5 FHC Core Assembly Examination Fixture (CAEF) Preliminary Design Report, December 18, 1981
24. N099ER415024, Conditioning and Service System 41-5 FHC Spent Fuel Transfer Port Final Design Report, October 27, 1980
25. N099ER415019, Conditioning and Service System 41-5 Maintenance and Service Station Final Design Report, June 18, 1979
26. N099ER415020, Conditioning and Service System 41-5 Fuel Handling Cell Lighting and TV Viewing Systems Final Design Report, November 14, 1978
27. N099ER415021, Conditioning and Service System 41-5 Equipment Transfer Drawer Final Design Report, October 13, 1980

28. N099ER415022, Conditioning and Service System 41-5 FHC Roof Closure Final Design Report, May 21, 1980
29. N099ER415023, Fuel Handling Cell System 41-5 28-in. Maintenance Floor Valve Final Design Report, February 5, 1981
30. N099ER416022, New Fuel Unloading Station Design Report, July 17, 1980
31. N099ER416021, New Core Assembly Inspection Equipment Design Report, July 17, 1980
32. N099TI417021, Software Preliminary Design for Reactor Refueling System, February 9, 1977
33. N099TI417035, Software Detail Design for Reactor Refueling, August 22, 1977
34. TI099-410-001, "Reactor Refueling System Conceptual Studies," April 1973. To be published as ESG-DOE-13437.
35. TP-097-241-003, "Test Plan--FTM Stem Seals and Bushings, Phase II, Na Environment Tests," February 23, 1973
36. TR-097-241-003, "Test Report, IVTM Seals and Bushings, Phase III, Na Environment Tests," June 1, 1974
37. TR-097-412-001, "Test Report, IVTM Seals and Bushings, Phase IV, Na High Temperature," September 11, 1974
38. ETEC-TDR-80-17, "Test Report on the In-Vessel Transfer Machine (IVTM)," October 29, 1980
39. N099TR412006, "Final Report on the In-Vessel Transfer Machine (IVTM)," September 20, 1982

40. N099TI417044, "IVTM Hoist Drive Controller (SC049) Failure Analysis," March 7, 1980
41. N099TR413005, Rev. NC, "Test Report EVTM Subscale," October 9, 1975
42. N099TI413039, Rev. NC, "EVTM/CLEM Full-Scale Test Analysis," September 16, 1975
43. N099TR413009, Rev. NC, "EVTM Coupon Emissivity Test Report," December 10, 1975
44. N099TI413042, Rev. NC, "Subscale Emissivity Test Analysis," October 21, 1975
45. N099TI413093, Rev. NC, Change 1, "EVTM Grappling and Degrappling Operating Description," July 1, 1981
46. N099TR000009, Rev. NC, "EVTM Positive Latching Grapple Prototype Test Report," May 4, 1984
47. N099ER413021, Rev. A, "Ex-Vessel Transfer Machine Final Design Report," July 25, 1980
48. N099ER413012, Rev. A, "EVTM Gantry Trolley Preliminary Design Report," September 11, 1979
49. N099ER413024, Rev. NC, "Core Component Pot Final Design Report," September 24, 1979
50. N099TR413014, Rev. NC, "Core Component Pot Siphon Test Report," June 21, 1977
51. N099TR413012, Rev. B, "EVTM Grapple Chain/Tape Test Report," May 6, 1977

52. ESG-D0E-13298, "LMFBR Inflatable Elastomeric Seals Development and Test History," Revised October 13, 1980
53. N099TR000007, Rev. NC, "Qualification Tests--Fuel Handling Inflatable Seals," August 10, 1983
54. N099ER444021, "System 44-4 Remote Viewing System - Articulating Column and Ancillary Equipment Conceptual Design Report," December 9, 1982
55. N099R006679, Rev. 0, "Articulating Column - CRBRP Remote Viewing System Conceptual Design Layout," dated December 8, 1982
56. N099R006683, Rev. 0, "Transporter and Handling Machine - Articulating Column Conceptual Design Layout," dated December 8, 1982
57. N099R006685, Rev. 0, "PSP Port Adapter - Articulating Column Conceptual Design Layout," dated December 8, 1982
58. N099R007022, Rev. 0, "Floor Valve, Gas Lock and Bags - Articulating Column Conceptual Design Layout," dated December 8, 1982
59. N099006092, Rev. 2, "TV Control and Monitoring Console Drawing," dated November 29, 1982
60. N099006094, Rev. 2, "Remote Viewing EI&C Block Diagram," dated November 29, 1982
61. N099007048, Rev. 0, "Articulating CCTV Column - Remote Viewing System P&I Diagram," dated December 8, 1982
62. N099TSR444002, "Trade Study to Establish Design Bases for Conceptual Design of Articulated Viewing System," September 29, 1981

63. N099TI444028, "Articulated Viewing System Geometry Study," October 6, 1981
64. N099TI444029, "Remote Viewing CCTV Concept Study for ISI of the Welds in the R.V. Downcomers, IHX and PSP," February 22, 1982
65. N099TI411-061, "Source of Requirements in SDD-41," January 7, 1977
66. ICD CA53299, "System 41 (Reactor Refueling)/31 (Reactor) Interface Requirements," May 25, 1983
67. ICD CA53300, "System 41 (Reactor Refueling)/32 (Reactor Enclosure) Interface Requirements," June 29, 1983
68. ICD CA53309, "System 41 Interface Requirements with Systems 12, 21A, and 27," October 29, 1982
69. ICD CA53088, "IVTM In-Vessel Section--Reactor Interface," March 25, 1980

6.0 PRIMARY HEAT TRANSPORT SYSTEM

This chapter provides a description of the piping arrangement guidelines and other considerations used in the design and analysis of the in-containment portion of the Heat Transport System (HTS) piping (Section 6.1) and also a detailed discussion of the design, analysis, and test considerations embodied into the large diameter HTS pipe clamps (Section 6.2).

6.1 Piping Arrangement Guidelines

6.1.1 Introduction

The Primary Heat Transport System (PHTS) and the Intermediate Heat Transport System (IHTS) sodium piping within the Clinch River Breeder Reactor Plant (CRBRP) Reactor Containment Building (RCB) must withstand high temperatures for long periods of time, in the presence of flowing sodium, and with a variety of thermal transients occurring during the normal operation of the plant. The reactor primary outlet piping (hot leg) operates at temperatures as high as 1015°F for a significant portion of the 30 year operating life of the plant. The reactor primary inlet piping (cold leg) operates at a temperature of 750°F, but for some transients, can reach as high as 1000°F.

The mechanical design process for the PHTS and IHTS piping within the RCB was influenced by elevated temperature considerations. Since the piping operates at temperatures greater than 800°F, time-dependent material failure modes as well as time-independent material failure modes must be accounted for in the design. Also, since the rapid temperature transients which occur during CRBRP operation may induce through-wall stresses in excess of the piping material yield strength, local plastic straining which can contribute to thermal ratchetting must be investigated.

The following sections describe the considerations and design philosophy embodied into the design and analysis of the CRBRP PHTS and IHTS large diameter piping within the RCB for elevated temperature service. In addition, discussions on snubber minimization, seismic margin assessments, and inelastic analysis utilization are provided.

6.1.2 System Description

The CRBRP heat transport system consists of three almost identical cooling circuits. Each heat transport loop is subdivided into two systems: a primary heat transport loop and an intermediate heat transport loop. These two loops are thermally coupled by an intermediate heat exchanger (IHX). Radioactive sodium is transported by the primary loop from the reactor vessel (RV) to the IHX (hot leg) and then back to the reactor vessel (cold leg). Non-radioactive sodium is transported by the intermediate loop from the IHX through the RCB penetration to the steam generators and then back to the IHX.

The three primary circuits have common flow paths through the reactor vessel but are otherwise independent in operation. The three intermediate loops, discussed in Section 7.1, are completely independent and have no common flow path.

The three HTS loops, as shown in Figure 6-1, are contained in separate inerted cells which are arranged about the centrally located reactor vessel. Atmospheric separation (required for maintenance activities) and radiation attenuation between the heat transport cells and the reactor cavity are provided by bellows seals and concrete shielding.

Each cell contains a primary loop consisting of a 36 inch diameter hot leg pipe, a primary sodium pump (PSP), a 24 inch diameter hot leg cross-over pipe, an IHX, and a 24 inch diameter cold leg pipe with a check valve (CLCV). Isolation valves are not required due to the elevated loop concept which precludes siphoning of reactor coolant in the event of a sodium leak. In addition, each cell contains a 24 inch diameter hot leg and a 24 inch diameter cold leg intermediate line. Both lines connect the intermediate side of the IHX with 24 inch penetrations at the RCB boundary. Flow control of sodium in the PHTS and IHTS is achieved by varying the pump speed. Figure 6-2 shows the arrangement of the Loop 1 HTS within the RCB.

To maintain a minimum sodium level in the reactor vessel in the unlikely event of a sodium leak in the primary boundary, all components and piping below a given elevation are contained within guard vessels.

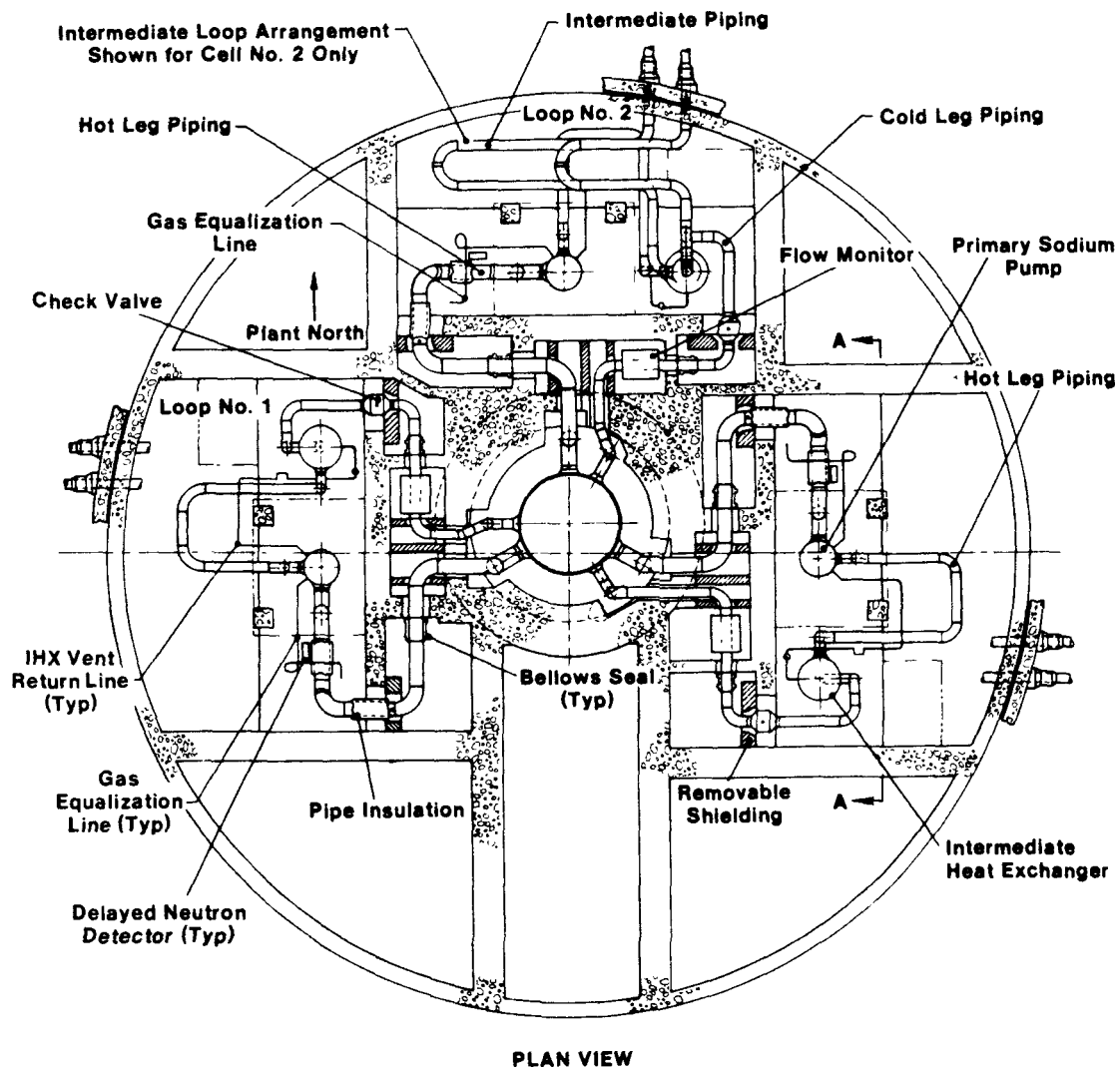


Figure 6-1 General Arrangement of CRBRP Heat Transport System

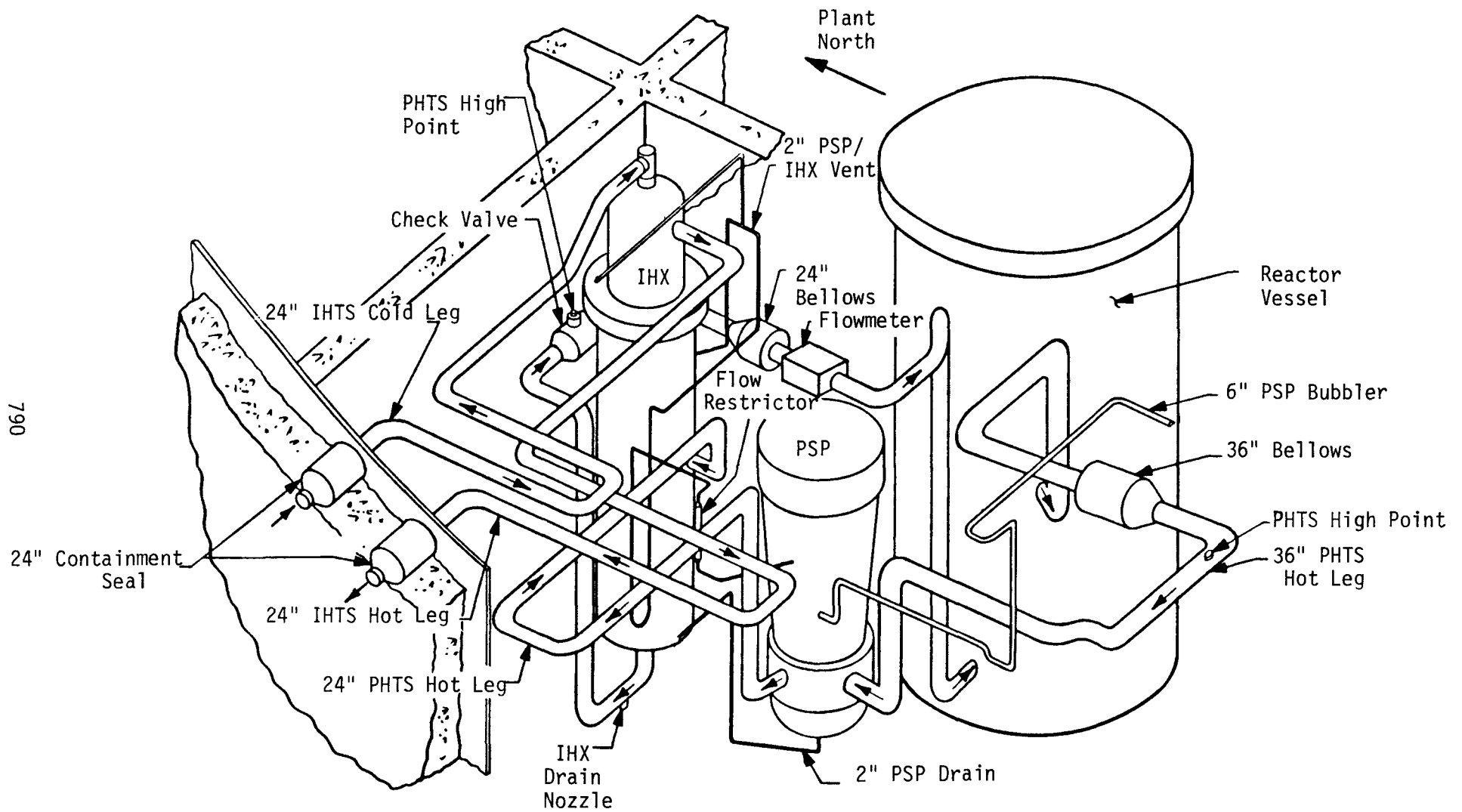


Figure 6-2

Loop 1 of CRBRP Heat Transport System Piping

Table 6-1 presents the fifteen large diameter HTS piping systems within the RCB and Table 6-2 provides additional information regarding sizes, materials, temperatures, pressures, etc.

6.1.3 Design Requirements, Guidelines, and Methods

6.1.3.1 General Requirements

The specific design requirements for the HTS piping systems are stipulated in the CRBRP piping design specification [1]. These requirements were established to be consistent with the performance of specific functions by the piping system during reactor operation and with quality and safety standards developed by regulatory and industry bodies for the design and fabrication of nuclear components. The design specification provides the applied loadings that are used in the CRBRP piping design and evaluation process. These loadings include such effects as internal pressure, deadweight, anchor motions, thermal expansion, thermal transients, and seismic excitations. In line with the design requirements, the CRBRP heat transport piping system was designed and analyzed as an ASME Class 1, Seismic Category I [2] nuclear component in accordance with the requirements of the ASME B&PV Code, Section III [3] as supplemented by special provisions given in Code Case 1592-7 [4] and certain RDT Standards [5,6] for nuclear components in elevated temperature service. The consideration of thermal creep effects sets the elevated temperature rules of Code Case 1592-7 apart from the ASME B&PV Code Section III Subsection NB rules. Unlike Subsection NB design rules which basically guard against time-independent failure modes, the elevated temperature rules are applicable for service conditions where creep and relaxation effects are significant.

6.1.3.2 Arrangement Guidelines

From the operational, functional, and structural requirements listed in the design specification, specific guidelines were established for use in the development of the piping system arrangement. These guidelines were established on the basis of radiation protection, maintenance, in-service inspection, safety, high temperature considerations, and from the general

TABLE 6-1
HTS PIPING WITHIN THE REACTOR CONTAINMENT BUILDING

<u>System</u>	<u>Loop</u>	<u>Pipe Size (in.)</u>	<u>Arrangement</u>	<u>Isometric Drawing</u>	<u>Latest Revision</u>
PHTS	1	36	RV Outlet to PSP Inlet	1453F40	6
PHTS	2	36	RV Outlet to PSP Inlet	1453F45	6
PHTS	3	36	RV Outlet to PSP Inlet	1453F49	6
PHTS	1	24	PSP Outlet to IHX Inlet	1454F71	3
PHTS	2	24	PSP Outlet to IHX Inlet	1454F72	3
PHTS	3	24	PSP Outlet to IHX Inlet	1453F41	3
PHTS	1	24	IHX Outlet to RV Inlet	1453F42	6
PHTS	2	24	IHX Outlet to RV Inlet	1453F46	6
PHTS	3	24	IHX Outlet to RV Inlet	1453F50	6
IHTS	1	24	IHX Outlet to RCB Seal	1453F38	5
IHTS	2	24	IHX Outlet to RCB Seal	1453F43	4
IHTS	3	24	IHX Outlet to RCB Seal	1453F47	4
IHTS	1	24	RCB Seal to IHX Inlet	1453F39	3
IHTS	2	24	RCB Seal to IHX Inlet	1453F44	3
IHTS	3	24	RCB Seal to IHX Inlet	1453F48	3

PHTS: Primary Heat Transport System

IHTS: Intermediate Heat Transport System

RV: Reactor Vessel

PSP: Primary Sodium Pump

IHX: Intermediate Heat Exchanger

RCB: Reactor Containment Building

TABLE 6-2
DESIGN DATA FOR HTS PIPING WITHIN THE REACTOR CONTAINMENT BUILDING

<u>System</u>	<u>Description</u>	<u>Outside Diameter (in.)</u>	<u>Wall^a Thickness (in.)</u>	<u>Material^b</u>	<u>Design Temperature (°F)</u>	<u>Design Pressure (psi)</u>	<u>Insulation^c Thickness (in.)</u>
PHTS	Hot Leg	36.0	0.5	Type 316	1015	30	12.0
PHTS	Crossover	24.0	0.5	Type 316	1015	200	11.0
PHTS	Cold Leg	24.0	0.5	Type 304	775	200	8.0
IHTS	Hot Leg	24.0	0.5	Type 316	965	325	11.0
IHTS	Cold Leg	24.0	0.5	Type 304	775	325	7.0

^aNominal

^bSA-358 (pipe) and SA-403 (fittings) stainless steel

^cAlumina silica

characteristics of the CRBRP RCB. The guidelines, described below, coupled with the screening rules described in the subsequent section, provided the basis for assessing the relative merit of alternate piping sizes, arrangements, and pipe support configurations of the HTS piping systems.

In accordance with the design requirements, the following loop arrangement guidelines were developed to promote safety of the plant:

- A. Piping must be elevated or contained within guard vessels.
- B. Pipe routing must preclude neutron shine on components, particularly on the IHX.
- C. Ordinary concrete to be permitted for radiation shielding.
- D. All loops to be atmospherically separative (required for maintenance activities).

In addition to meeting the design requirements for safety considerations, piping arrangement guidelines involving good design practice and cost considerations were developed as follows:

- A. The piping should be arranged in essentially three identical loops.
- B. High point vents are to be provided in both the hot leg and cold leg piping.
- C. The piping must be drainable.
- D. The piping systems must be trace-heated and insulated with the piping providing the support for these items.
- E. The piping loops are to accommodate the system normal thermal expansion including transient events.
- F. All connections between piping components must be welded.

- G. Bellows seals attached to the piping should be provided between the reactor cavity and heat transport system cells.
- H. Rigid seals (flued head penetrations) attached to the piping must be provided at the reactor containment building boundary.
- I. Stresses in the piping should be limited to the established screening limits.
- J. Welded connections to elbows should be avoided.
- K. Elbow-to-elbow welds should be avoided.
- L. Bend radii for elbows should be limited to 1.5 pipe diameter.
- M. High points in both the hot leg and cold leg must be provided at controlled differential elevations.
- N. The piping must accommodate the motions of cantilever-supported major components.
- O. The piping must meet the loading limitations at component nozzles.
- P. The arrangement should minimize the containment diameter.

Additional guidelines in consideration of pipe supports, maintenance and erection, and in-service inspection were also developed [7].

6.1.3.3 Pipe Screening Rules

Screening rules that account for elevated temperature considerations were established for assessing the acceptability of alternative arrangements of the CRBRP piping and support systems [8]. The objective of the screening rules was to limit the stress induced by each type of loading condition such that, in the final detailed stress analysis of the HTS piping, the ASME B&PV Code design limits for combined loadings were satisfied.

It is advantageous if the piping analyses for deadweight, thermal expansion, and seismic loads are carried out independently of each other during the design process. The allowable stress in the piping for each different event is apportioned so that the combined stresses do not exceed the allowable ASME B&PV Code stress limits. With separate and distinct load and stress limits for each of the major load events, each analysis can be carried out independently. This will assure that if the stress limits are satisfied for the individual loadings, they will also be satisfied for the combined loadings.

To minimize the size (diameter) of the RCB, it is essential that the heat transport piping system loops be short for a compact arrangement. Therefore, the pipe routing rules established for the HTS piping set relatively high allowable stress limits, recognizing that the final ASME B&PV Code compliance evaluations could require complex inelastic analyses. The pipe routing rules were established by apportioning the allowable stress to the different types of loading events such as pressure, deadweight, seismic, and thermal expansion stresses.

In conjunction with the screening rules, methods for determining component nozzle loads must be established to assure that the piping design does not give reactions greater than the allowable nozzle loads. Therefore, a method was developed to obtain component nozzle loads that gave a practical bound on the expected pipe reactions based upon the limit loads for the piping adjacent to a component nozzle.

For the heat transport system cold leg piping that is normally operating at temperatures of less than 800°F, the pipe routing rules were based upon satisfaction of the primary-plus-secondary stress intensity range limits, i.e., Equation 10 of Subsection NB-3600 of the ASME B&PV Code [3]. Specifically, to establish the individual stress intensity limits for each type of loading for routing the cold leg piping, the primary-plus-secondary intensity range for the selected "worst-case" cycle under upset loading conditions had to be satisfied. The governing relation is defined as:

$$[S(P_L + P_b)] + S(Q) \leq 3 S_m$$

where

$[S(P_L + P_b)]$ = the value of the primary membrane-plus-bending stress intensity range that occurs during the operating cycle being evaluated.

$S(Q)$ = the value of the secondary stress intensity range that occurs during the operating cycle being evaluated.

S_m = maximum allowable stress for the material from Table I-1.0 of the ASME B&PV Code, Section III [3].

The above equation had to be satisfied for a reference cycle which consisted of ranges of pressure, seismic, thermal transient, and thermal expansion loading conditions. Based upon the evaluation of this reference cycle, the pipe routing rules were obtained as a set of allowable stress intensity limits for each of the types of loading considered.

For the heat transport system hot leg piping that is normally operating at temperatures greater than 800°F and designed in accordance with Code Case 1592-7 [4], the elastic stress limits to preclude strain ratchetting are usually the most difficult to satisfy. If elastic analysis is used, ratchetting can be precluded by limiting the primary-plus-secondary stress intensity induced under operating conditions involving combined weight, pressure, seismic, thermal expansion, and thermal transient loadings. For the purpose of establishing individual stress intensity limits for routing the hot leg piping, the same basic approach used for the cold leg piping was used.

To establish the stress limit for thermal expansion stresses, the primary-plus-secondary stress intensity limits to satisfy Equation 1, Paragraph T-1322 of Code Case 1592-7 (based on elastically calculated stresses), are used. The operating cycle was defined as the ambient heatup-to-normal operating cycle. In equation form, the limit as prescribed by Code Case 1592-7 can be written as:

$$[S(P_L + P_b)]_{\max} + [S(Q)]_{\text{range}} \leq 0.90 (S_a)$$

where

$[S(P_L + P_b)]_{\max}$ = the maximum value of the primary membrane-plus-bending stress intensity during the cycle.

$[S(Q)]_{\text{range}}$ = the maximum value of the secondary stress intensity range during the cycle.

S_a = S_a is equal to S_y if the average wall temperature at one of the stress extremes defining the secondary stress range is below the creep regime. Otherwise, S_a is defined as the lesser of S_y of $1.25 S_t$ taken at the highest average wall temperature occurring during the cycle at 10^4 hours.

S_y = the average of the minimum specified yield strength values at the maximum and minimum wall averaged temperatures of the cycle under consideration.

S_t = time dependent material allowable defined in Code Case 1592-7 [4].

In establishing the thermal expansion stress limit, it should be noted that the loading cycle being evaluated in accordance with the above equation does not give the maximum primary-plus-secondary stresses which occurs during all normal and upset operating conditions. The Operating Basis Earthquake (OBE) seismic upset cycle would give the maximum stresses. The use of a cycle which gives less than the maximum stress is based upon, firstly, the judgment that if the highest stressed cycle was used, the space requirements for the piping loops would be unacceptable from the standpoint of reactor containment building diameter. Secondly, the piping stresses that are calculated during the routing studies are based on the conservative, simplified piping analysis approach as defined in Subsection NB-3650. By shifting to detailed elastic analysis as defined in Subsection NB-3200, satisfaction of the Code Case 1592-7 elastic limits for all load cycles (in most cases) should be satisfied. Thirdly, a 90 percent factor is applied to the S_a allowable for

added assurance that the ASME B&PV Code limits can be met (in most cases) with elastic analysis. Finally, if inelastic analyses are required, it is expected that the degree of utilization will be limited.

Pressure, deadweight, thermal expansion, and seismic analyses were carried out independently for many design alternatives and configurations to determine stresses for comparison with the established screening rule limits. When the guidelines discussed in the previous section and the screening limits were satisfied, the piping and support design was acceptable and the baseline design established.

6.1.3.4 Design Evaluation

Once the baseline design of the CRBRP piping had been established, the detailed structural evaluations were made in accordance with the methods outlined in recognized nuclear industry codes and standards, the ASME B&PV Code and the additional, supplemental requirements of the RDT Standards. In addition, a comprehensive and detailed assessment of the integrity of the in-containment piping system was made to assure that there was a negligible likelihood of a large break [9]. The evaluation of the heat transport system piping included flexibility (static and dynamic), stress, and heat transfer analyses. The flexibility analyses were carried out on a linear elastic basis and provided forces and moments at selected points along the pipeline. The heat transfer analyses provided the through-the-pipe wall temperature gradients and average pipe temperatures at structural discontinuities. With these analysis data and a prescribed plant duty cycle or loading history, stress evaluations of the piping at selected locations were carried out with the ELTEMP computer program [10] at elbow midpoints and circumferential weld joints between an elbow and a straight pipe section.

At a given location in the piping system, stresses from the various loadings were combined to determine the total primary or load-controlled stresses at the given time in the load history and to find secondary or peak stress (strain-controlled) ranges between times in the histogram. The primary stresses were checked against a restrictive allowable that prevents gross

yielding through the pipe wall. Next, the possibility of gross distortion and creep-fatigue were checked using the elastically calculated primary stresses and ranges of secondary and peak stresses.

If the strain-controlled limits are not satisfied on an elastic basis, either redesign is necessary or inelastic analysis may be used to show that inelastic strains in the piping were not excessive. If the reason for non-compliance of strain-controlled limits is judged to result from excessive through-the-wall thermal transient stresses, the analysis procedure can be modified to use elastic flexibility analysis (forces and moments) with an inelastic analysis of the pipe component (elbow, reducer, etc.) to determine total strains. If the reason for non-compliance is judged to result from excessive flexibility forces due to thermal expansion, the procedure can be modified to use an elastic flexibility analysis and an elastic or inelastic piping component analysis. An inelastic combined flexibility and component analysis (as discussed in Section 6.1.6) is to be used as a last resort due to cost and effort considerations.

6.1.4 Snubber Minimization

6.1.4.1 Background

The number of snubbers used in nuclear power plants has continued to grow during the past decade. In particular, the installation of mechanical snubbers has increased significantly, partly in response to problems identified with hydraulic snubbers in the early to mid-1970's. Due to their safety-related functions and widespread use, snubbers are important components in nuclear power plant design.

Snubbers are used primarily to restrain movement of piping systems and components during low probability dynamic events. While restraining dynamic motion, snubbers allow movement with minimal resistance due to thermal expansions and contractions.

The use of snubbers to restrain piping systems and components affects the cost of nuclear power plants. The most obvious cost is related to the required hardware. This includes the snubbers, pipe clamps, concrete anchors and

embedments, and related connection hardware. Installation of snubbers is also a significant cost, particularly when including the costs associated with the embedments and auxiliary steel. Finally, there are costs related to qualification, maintenance and in-service inspection. In-service inspection activities also include requirements for operability testing during plant service.

Improper operation of snubbers can directly affect the overall reliability of nuclear power plants. Therefore, it is desirable to minimize the quantity of snubbers used to restrain piping systems and components to reduce plant costs and improve system reliability and safety.

6.1.4.2 CRBRP Approach

The CRBRP HTS piping was designed with a relatively thin wall thickness and was routed with large thermal expansion loops due to the rapid temperature transients in the duty cycle and high operating temperatures. This resulted in a very flexible piping arrangement. However, to limit the dynamic response to within allowable levels, a significant number of seismic restraints (snubbers) were required.

The snubbers were located on the HTS piping systems using dynamic analysis computer programs, simplifying assumptions, and good engineering judgement. However, the resulting support arrangements were shown to be sensitive to changes in snubber modeling parameters [11]. In light of this fact, the CRBRP Project initiated a development program (DRS 51.44) which had the objective to confirm and refine the analytical methods and models used for the design/analyses of the HTS piping.

The comprehensive test and analysis program was completed in September 1983 [12]. The correlation analyses have shown conservatism in the predicted responses of a typical Liquid Metal Fast Breeder Reactor (LMFBR) piping system when compared to actual test measurements. One of the major sources of conservatism is that LMFBR piping systems have significantly greater damping than currently allowed for dynamic analysis as specified by the U.S. Nuclear Regulatory Commission [13]. The use of this increased damping would result in

the application of fewer seismic restraints. This would result in a cost savings by reducing hardware, installation, maintenance, and inspection costs while maintaining the same level of structural integrity.

Studies were also performed to investigate postulated snubber failures outside the CRBRP duty cycle events [14]. Even though the results of the study indicate a high tolerance for snubber failure on the CRBRP HTS piping, the fact remains that numerous snubber failures do occur in operating plants. It is therefore advantageous to minimize the quantity installed on piping systems and components.

6.1.4.3 Future in Snubber Minimization Techniques

Based upon the results of the test and analysis program described previously and related activities in the United States and abroad, a potential exists to reduce the quantity of snubbers installed in nuclear power plants. The state-of-the-art is advancing rapidly due to the problems and inherent costs associated with over specification and improper operation of snubbers. Examples of analytical and physical refinements now being used or developed to minimize the application of snubbers on piping systems and components are:

- A. Use of increased damping in dynamic analysis calculations.
- B. Application of refined snubber models, analysis methods, and load combinations.
- C. Substitution of rigid struts by re-evaluation of thermal expansion stresses.
- D. Formal analytical optimization to eliminate unnecessary snubbers using available computer codes.
- E. Design of monoplanar piping systems and the use of fixed length pivotal links (in lieu of snubbers).

- F. Investigation of piping fragility levels to permit higher dynamic loadings than presently allowed.

It is strongly believed that the quantity of snubbers specified for the CRBRP HTS piping could have been minimized if the refinements discussed above, and specifically the use of increased damping, were utilized in the design and analysis process.

6.1.5 Seismic Margin Assessments

6.1.5.1 Background

Nuclear piping systems designed and analyzed in accordance with the ASME B&PV Code and NRC Regulations will, in general, have available inherent reserve capacity to withstand excitations beyond the Safe Shutdown Earthquake (SSE) level. Usually, the design of a typical Seismic Category I piping system is controlled by the Operating Basis Earthquake (OBE) level and not the SSE. The reserve seismic capacities can be attributed to:

- A. Conservative predictions of building and equipment seismic response.
- B. Conservative definition of structural and functional performance limits.
- C. Reserve seismic capabilities incorporated by means of designer actions, i.e., safety factors.

The regulatory agencies, during the licensing process, required that the seismic margin for CRBRP systems and components be quantified since there was a concern of exceeding the design basis SSE value. Therefore, detailed evaluations were performed to demonstrate that positive seismic margins existed.

6.1.5.2 Seismic Margin Approach

The reserve seismic capacity for the HTS piping systems was determined from the product of the structural strength reserve capacity and the piping system seismic response conservatism. The structural strength reserve capacity accounts for the strength of piping components, elbows, clamps, snubbers, and support structures. The piping system seismic response conservatism results from system damping assumptions, development of the ground accelerogram, reduction of floor response spectra due to building inelastic actions, development of design floor response spectra, and the development of design time histories.

The approach taken to calculate the reserve seismic margin for the CRBRP HTS piping systems was based upon methods developed by Rodabaugh and Desai [15] for the Nuclear Regulatory Commission and modified for CRBRP applications by Mello and Chandra [16] and Morrone [17].

6.1.5.3 Piping System Seismic Margins

The reserve seismic margin for the HTS piping systems was determined by the design margin for the SSE event and the nominal margin (between ASME B&PV Code allowable and ultimate failure) as defined below:

$$\text{Design Margin (DM)} = \frac{\text{Allowable Stress}}{\text{Calculated Stress}} = \frac{S_A}{\sigma_C}$$

$$\text{Nominal Margin (NM)} = \frac{\text{Ultimate Stress}}{\text{Allowable Stress}} = \frac{S_U}{S_A}$$

Therefore, the actual margin is:

$$\text{Actual Margin (AM)} = \text{DM} \times \text{NM} = \frac{S_U}{\sigma_C}$$

For the HTS piping systems, the seismic only margin (SOM) was defined as shown below:

$$SOM = \frac{(NM \times DM - 1)}{k} + 1.0$$

where

k = Ratio of seismic only stress (or load) to the total calculated stress (σ_c)

Based upon the relation above, the following minimum reserve strength capacities were determined for the HTS large and small diameter piping systems:

A. Piping Components = 1.67 (small diameter piping)
2.71 (large diameter piping)

B. Clamps = 3.37

C. Snubbers = 1.86 (small diameter piping)
1.89 (large diameter piping)

D. Embedments = 4.0

The conservatism introduced by the piping seismic response prediction (damping, ground acceleration, inelastic actions, etc.) was determined to be a factor of 1.45 by Morrone in an evaluation of the seismic margin of CRBRP systems and components. Therefore, combining the results for the minimum reserve strengths and the response prediction conservatism results in an overall reserve seismic capacity of 2.42 for the small diameter piping and 2.74 for the large diameter piping. This translates into a minimum reserve margin earthquake of 0.605 g for the CRBRP HTS piping, which is determined by the product of the minimum seismic capacity (2.42) and the design SSE maximum ground acceleration for CRBRP (0.25 g).

As a result of the seismic margin assessments, it was demonstrated that a substantial margin exists for the CRBRP large and small diameter piping systems. Therefore, the concern for the "seismic fragility" of the thin-walled piping systems was resolved.

6.1.6 Inelastic Analysis Utilization

6.1.6.1 Background

The first interim stress report for the PHTS and IHTS piping within the RCB was completed in 1975 and supported the baseline arrangement of the piping, including material selection, pipe wall thickness, size of expansion loops, and the location of pipe supports [18]. The second interim stress report, completed in 1977, extended the structural evaluation of the HTS piping to a level that established the adequacy of the design to initiate procurement of materials and fabrication of piping components [19]. Finally, the draft ASME B&PV Code Section III Stress Analysis Report (exclusive of as-built conditions) was completed in October 1979 [20]. However, for a few locations on the PHTS 24 inch cross-over (hot leg) and 36 inch hot leg piping systems, the ratchetting and creep-fatigue evaluations based upon elastic analyses showed that the ASME B&PV Code Case 1592-7 [4] limits were not satisfied. Therefore, to demonstrate compliance with Code Case 1592-7, the inelastic response of the 24 inch cross-over piping system was evaluated [21].

6.1.6.2 Methods

Detailed inelastic stress analyses of piping systems are significantly more complex and costly than linear type elastic analyses. However, piping system problems involving complex geometry, material creep, and plasticity effects are more realistically treated. This is compatible with the ASME B&PV Code screening philosophy in which the simpler method is purposely devised to be the less precise, but the more conservative method.

Since the inelastic methods are inherently costly, there is a necessity for limiting either the scope of the structure to which it is applied, or else to limit the resolution of stress or strain in the model. For example, modern

finite element methods can be applied to the time-dependent stress analysis of an extended piping system. But, in such an extended model, high resolution of stress within individual components (elbows, reducers, tees, etc.) can only be obtained at a very large cost. Conversely, when a critically-stressed component is modeled along, very detailed resolution of stress and strain variation with time can generally be obtained. In that case, the component loads might be typically represented by the internal reactions and the displacements found from an elastic system analysis. In this second case, some loss of resolution in the load history is accepted by the analyst in exchange for better resolution of stress.

The fundamental purpose of detailed inelastic analysis, as applied in CRBRP piping systems analysis, was to demonstrate ASME B&PV Code compliance for critically-stressed "special problem" components. A special problem is one for which the methods previously discussed were either inapplicable or were found insufficient to demonstrate compliance. This was the case for the 24 inch PHTS cross-over piping.

6.1.6.3 Inelastic Analysis Results

Two detailed inelastic analyses were performed for the 24 inch PHTS cross-over piping loop [21]. The analyses considered deadweight, pressure, seismic, and thermal transient loads as defined in the piping design specification [1].

The first analysis excluded the seismic effects; however, the primary load-controlled seismic load-up was simulated by assuming three times the normal operating pressure and deadweight loading. The second analysis included the seismic loads as equivalent static loads; hence, the normal operating pressure loading was used. The seismically equivalent static loading procedure results in conservative moments at the piping system elbows and accurately represents the distribution of piping system internal forces and moments.

The results of the inelastic analyses at the various critical locations on the 24 inch PHTS cross-over piping system showed compliance with Code Case 1592-7 inelastic creep-fatigue damage and strain accumulation limits as documented in

Reference 21. Therefore, the 24 inch PHTS cross-over piping leg was certified as a Class 1 nuclear piping system in accordance with the ASME B&PV Code. Similar evaluations were to be performed for the 36 inch PHTS hot leg, but were not completed at the time of CRBRP Project termination in November 1983. However, it is believed that the 36 inch PHTS hot leg analyses would have produced a certified design.

6.1.7 References

1. "CRBRP Heat Transport System Piping Design Specification," E-953089, Revision 37, June 1983.
2. "Seismic Requirements for Design of Nuclear Power Plants and Test Facilities," RDT Standard F9-2T, U.S. Department of Energy, Washington, DC.
3. ASME Boiler and Pressure Vessel Code, ANSI/ASME BPV-III-NCA, Division 1, "Rules for Construction of Nuclear Power Plant Components," 1974 Edition with addenda through Summer 1975, American Society of Mechanical Engineers, New York, 1974.
4. "Class 1 Components in Elevated Temperature Service, Section III," Code Case 1592-7, ASME Boiler and Pressure Vessel Code, December 1975, American Society of Mechanical Engineers, New York.
5. "Class 1 Nuclear Components (Supplement to ASME Boiler and Pressure Vessel Code, Section III, Subsections NCA and NB)," RDT Standard E15-2NB-T, U.S. Department of Energy, Washington, DC.
6. "Requirements for Construction of Class 1 Elevated Temperature Nuclear System Components (Supplement to ASME Code Cases 1592, 1593, 1594, 1595 and 1596)," RDT Standard F9-4T, Amendment 4, April 1977, U.S. Department of Energy, Washington, D.C.
7. Pollono, L. P. and Mello, R. M., "Design Considerations for CRBRP Heat Transport System Piping Operating at Elevated Temperatures," 79-NE-5, American Society of Mechanical Engineers, New York, 1979.
8. "Structural Evaluation Plan (SEP) for the HTS Piping System," WARD-D-0087, Rev. 1, November 1980.
9. Mallett, R. H. and Nair, B. R., "CRBRP Integrity of Primary and Intermediate Heat Transport System Piping In-Containment," CRBRP-ARD-0185, Volumes 1 and 2, October 1977.
10. Thomas, K., "ELTEMP Revision 3: Modifications, Additions and Corrections," ES-LPD-79-004, October 1979.

11. Mello, R. M. and Pollono, L. P., "Evaluation of the Influence of Seismic Restraint Characteristics on Breeder Reactor Piping Systems," in Piping Restraint Effects on Piping Integrity, PVP-37, pp. 63-77, ASME, New York, NY, June 1979.
12. Hulbert, G. M., Schott G. A. and Heberling, C. F. II, "Correlation Results from Dynamic Testing of an Eight Inch Diameter Piping System", WAESD-ST-94000-7, September 1983.
13. "Damping Values for Seismic Design of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Regulatory Guide 1.61, October 1973, Washington, D.C.
14. Mallett, R. H., Nair, B. R., and Mello, R. M., "CRBRP Piping Integrity: An Evaluation of Postulated Support Failures," CRBRP-ARD-0183, September 1977.
15. Rodabaugh, E. C. and Desai, K. D., "Realistic Seismic Design Margins of Pumps, Valves, and Piping," NUREG/CR-2137, June 1981, U.S. Nuclear Regulatory Commission, Washington, DC.
16. Mello, R. M. and Chandra, U., "CRBRP Heat Transport System Incontainment Piping Reserve Seismic Margins," ES-LPD-83-001, Revision 1, May 1983.
17. Morrone, A., "CRBRP Reserve Seismic Margins," WARD-D-0332, November 1983.
18. "CRBRP; First Interim Stress Report for Heat Transport System (SDD-51) Piping," WARD-D-0120, November 1975.
19. "CRBRP; Second Interim Stress Report for Heat Transport System (SDD-51) Piping," ES-LPD-77-001, July 1977.
20. "ASME B&PV Code Section III Stress Analysis Report (Exclusive of As-Built Conditions) for the CRBRP Heat Transport System Large Diameter Piping Within Containment," ES-LPD-79-002, Sections I-VII, Draft, October 1979.
21. Dhalla, A. K. and Mallett, R. H., "Inelastic Evaluation for the PHTS 24-Inch Hot Leg Piping," ES-LPD-79-002, Section VII-A, February 1982.

6.2 Pipe Clamp Design

6.2.1 Introduction

The design of pipe clamps for Liquid Metal Fast Breeder Reactor (LMFBR) piping systems presents a departure from conventional Light Water Reactor (LWR) designs. This is due to the nature of the liquid metal sodium, which is the heat transport system coolant carried by the primary and intermediate piping systems. Since LMFBRs reach much higher temperatures, experience more severe temperature transients, and operate at substantially lower pressures than LWRs, the Heat Transport System (HTS) piping is designed with a relatively thin wall thickness to reduce the severity of the radial thermal gradients in the piping system. These temperature effects cause high bending stresses in the pipe wall. In addition, a metal-to-metal clamp directly applied to the piping outside diameter or a welded-on lug (attachment) will effectively increase the thickness of the piping locally and also provide a local thermal heat sink. This condition represents a gross material and thermal discontinuity which will further increase the bending stresses in the pipe wall under thermal transient conditions. Also, high local bending stresses will also be produced in the clamp body itself. Another area for concern is the potential for local buckling of the pipe wall if welded-on lugs are used due to the effect of a concentrated (point) load applied on a relatively small section of thin-walled pipe. To avoid these problems, non-integral insulated clamps were designed for CRBRP HTS piping applications.

The following sections describe the design, analysis, and test considerations embodied into the large diameter HTS pipe clamps for sodium piping systems within the CRBRP Reactor Containment Building (RCB).

6.2.1.1 Design Evolution

Historically, pipe clamps used for LMFBR piping systems have been primarily of the "cold clamp" design (i.e., clamp body insulated from the pipe wall). However, "hot clamps" (i.e., metal-to-metal) have been successfully utilized in various liquid metal test loops. Foreign liquid metal piping experience also indicates that the use of the "hot clamp" design may be feasible,

especially for smaller diameter piping. For thick-walled LWR piping systems, the "hot clamp" design is used exclusively because the clamp-induced effects are known to be less critical.

Thermal studies have shown that standard, non-insulated ("hot clamp") pipe clamps act as heat sinks and cause severe local discontinuity stresses in large diameter thin-walled piping during the large and rapid thermal transients that occur during reactor scrams and other upset events. Also it has been shown that the differential metal temperatures between the pipe wall and the clamp could significantly reduce the thermal fatigue life cycle capabilities of the pipe. Although there is an economic incentive to use the "hot clamp" design, an insulated "cold clamp" design was developed for use on LMFBR piping systems.

The initial design of the CRBRP HTS clamps, shown in Figure 6-3, was based upon the concept used for the high temperature piping in the Fast Flux Test Facility (FFTF). However, for certain load configurations, finite element analyses have shown that the "flexible" clamp design was not structurally acceptable for CRBRP applications [1]. This design produced high localized clamp/pipe interface loads and unacceptable pipe wall bending stresses. In addition, it was discovered that the combined clamp/pipe stiffness was lower than desired for certain applications. This condition resulted in less than optimum conditions in terms of attenuating piping system dynamic response. As a result of these concerns, the clamps were redesigned to produce a stiffer clamp band and a more favorable stress condition in the HTS piping [2].

6.2.1.2 Design Requirements

The pipe clamps provide attachment points on the HTS piping for the connection of supports and restraints in order to minimize piping stress and loads on connecting equipment. The HTS pipe clamps were designed [3,4], tested [4], and analyzed [1,2,5] in accordance with ASME B&PV Code [6,7] and the following requirements [8]:

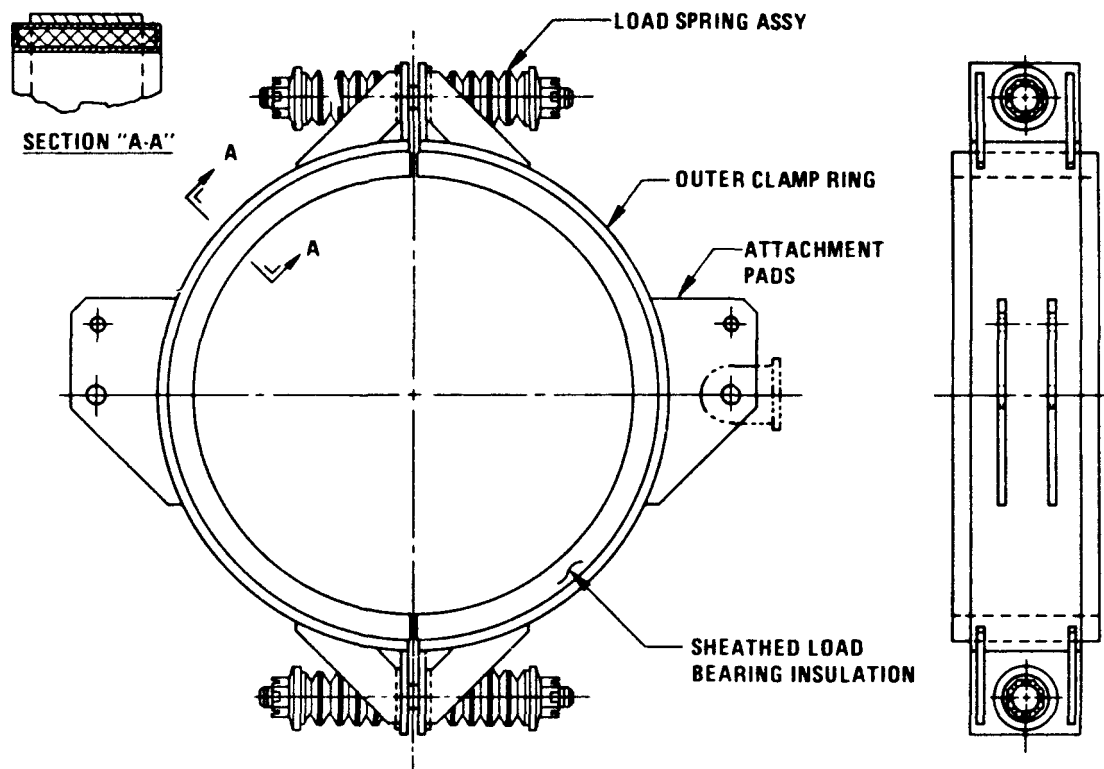


Figure 6-3 Initial HTS Flexible Clamp Design

- A. The clamps shall accommodate the piping diametral thermal variations without excessive stress on the pipe wall.
- B. The clamps shall be insulated from the pipe wall to minimize thermal discontinuity stresses.
- C. The clamps shall accommodate fabrication tolerances in the pipe outside diameter.
- D. The clamps shall provide for gaseous leak detection and passage for liquid sodium leakage.
- E. The clamps shall be capable of carrying any combination of maximum design loads (thermal expansion, deadweight, seismic, etc.).

The requirements for the fabrication, inspection, testing, cleaning, packaging, Quality Assurance, and shipping of the large diameter HTS clamps within the RCB are delineated in the equipment specification [9].

6.2.1.3 Design Features

The design of a typical pipe clamp assembly used on the horizontal sections of the CRBRP HTS piping is shown in Figure 6-4. The complete restraint assembly consists of the non-integral insulated clamp, seismic snubbers, pipe hangers, and miscellaneous hardware as shown in Figure 6-5. The clamp assembly consists of two semi-circular carbon steel bands (with attachment gussets) which are held together by a system of Belleville springs to accommodate variations in pipe diameter due to temperature changes. For the large diameter HTS piping, the width of the bands ranged from 6 to 12 inches, depending upon the required load rating capability of the clamp design. The Belleville spring stackups accommodated the pipe thermal expansion and contractions without inducing large changes in the bolt/spring loads or the contact pressure at the pipe/clamp interface. This was accomplished by designing the Belleville spring stackup such that its load-deflection curve

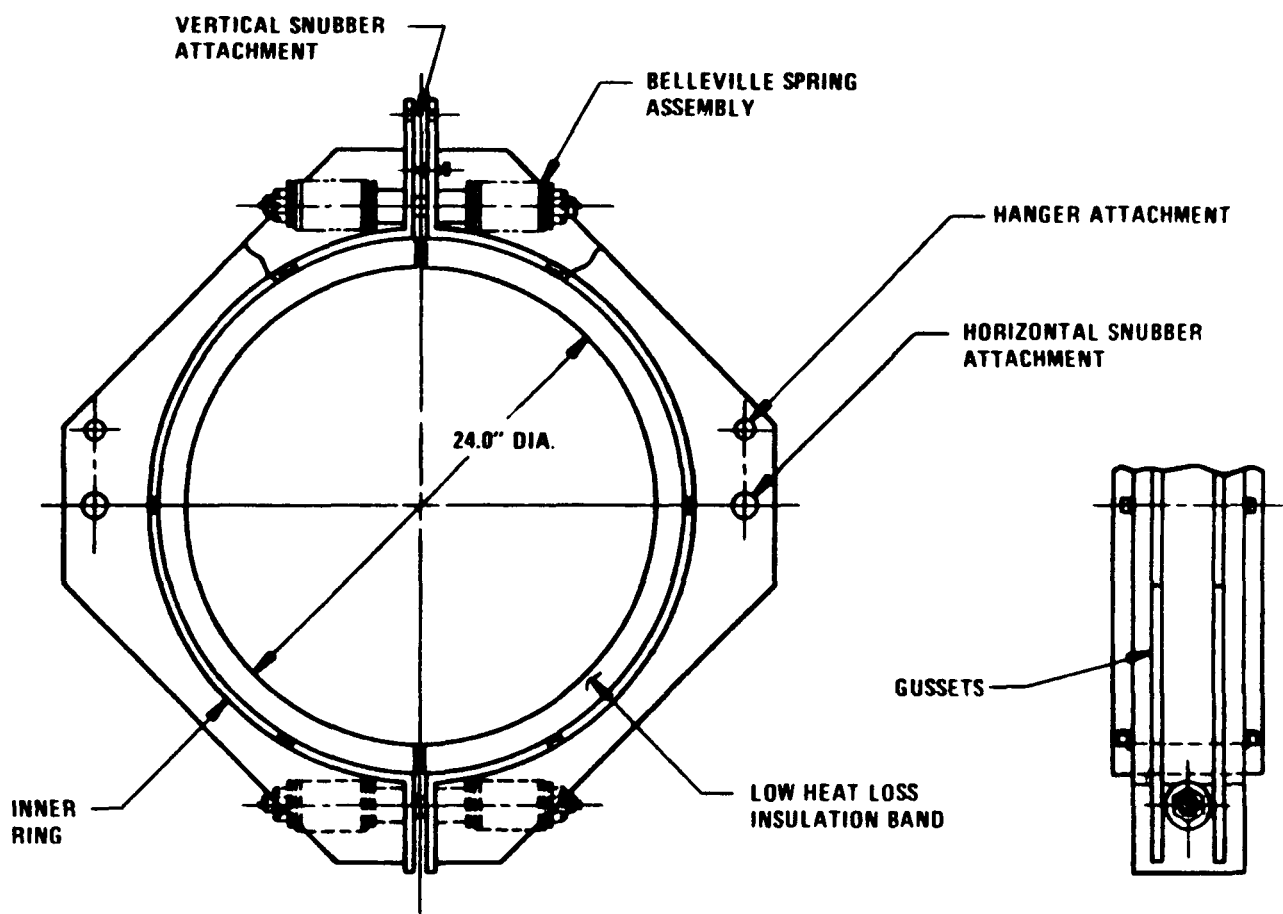


Figure 6-4 Typical HTS Horizontal Pipe Clamp Assembly (Type B)

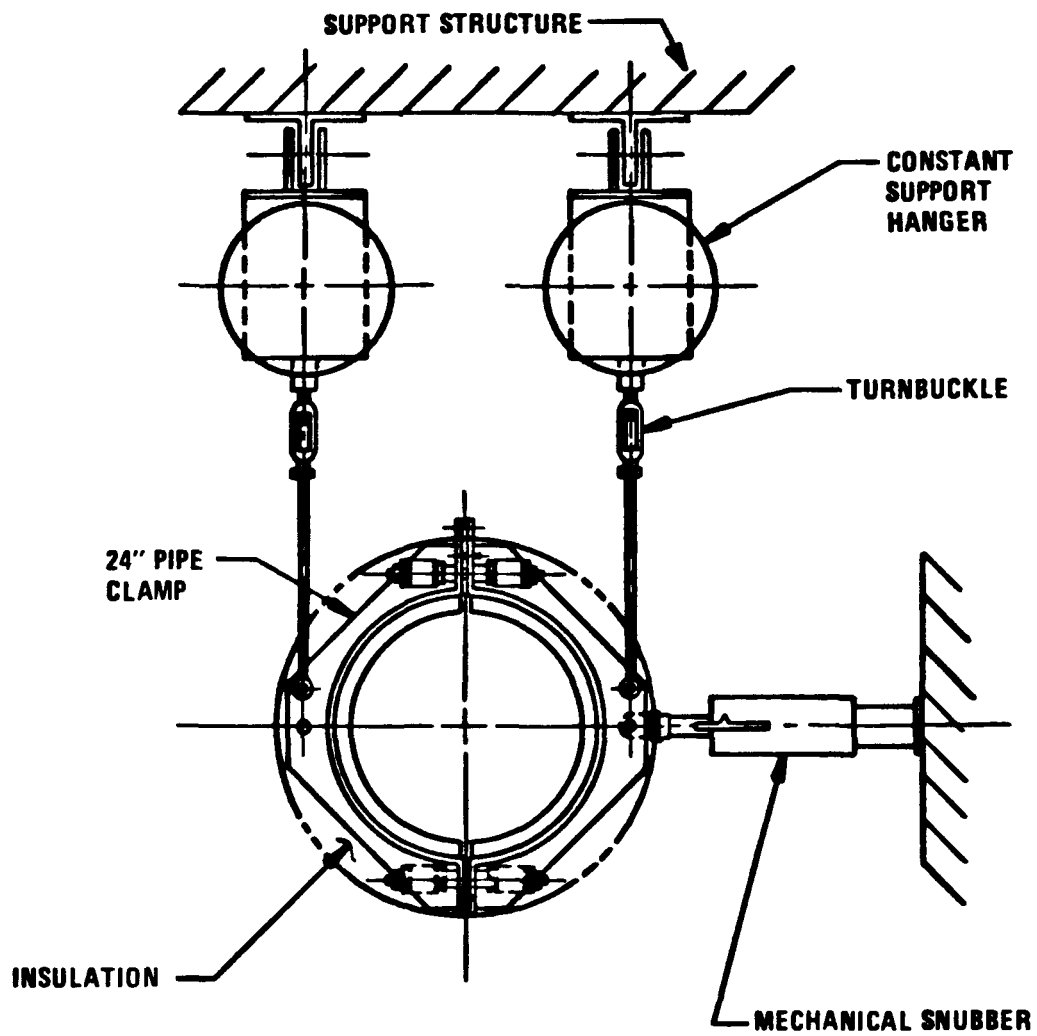


Figure 6-5 Typical Horizontal Restraint Assembly for HTS Piping

had a relatively flat response in the vicinity of the selected clamp preload. The preloaded springs prevented the clamp halves from lifting off the pipe wall when loads were applied at the attachment points.

Two semi-circular insulation bands are sandwiched between the clamp bands and outer pipe wall to minimize the thermal transient stresses. These insulation bands, which are 1.5 inches thick (for all clamp sizes), contain segmented sections of load bearing insulation (Marinite XL). The insulation blocks are encased by stainless steel sheathing which encapsulate the pipe wall over the width of the clamp (axial to pipe centerline) to provide a positive compressive load area. Similar design non-integral clamps have been used for the liquid metal piping systems in the Fast Flux Test Facility (FFTF) and in the Sodium Pump Test Facility (SPTF).

For the long vertical sections of HTS piping within the guard vessels, a special non-integral vertical clamp assembly was designed as shown in Figure 6-6. The vertical pipe clamp consists of three major components: the outer clamp ring, the steel-sheathed load bearing insulation inner bands, and the pipe transition section (adaptor) which includes an axial support ledge. The vertical and horizontal clamp design principles are similar with the exception of the required pipe transition section which formed an integral part of the HTS piping pressure boundary.

In total, there were thirteen unique clamp sizes designed for the large diameter HTS piping within the RCB as shown in Table 6-3. The only distinction between the Type A and Type B designs is that the Type B clamps have an extended bolt flange at the top of the clamp which permits attachment of a vertical seismic restraint.

6.2.2 Analysis Methods and Modeling Techniques

The analytical methods and modeling techniques used for the design and analysis of both the stiff and flexible clamps were essentially the same [1,2]. Since the clamp assembly is slightly larger in diameter than the outside diameter of the HTS pipe and because the pipe wall and the portion of

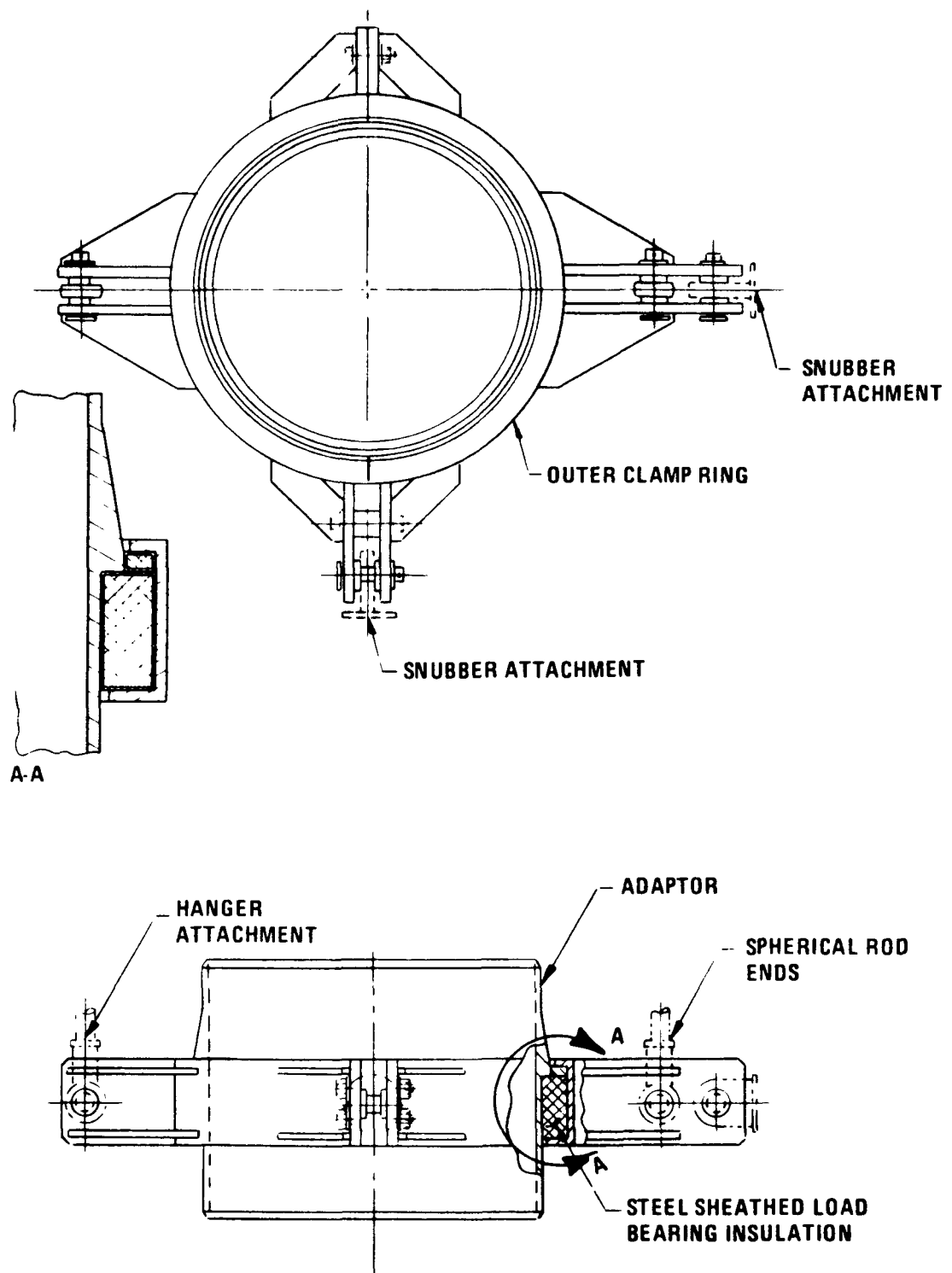


Figure 6-6 Typical HTS Vertical Pipe Clamp Assembly

TABLE 6-3
CRBRP HTS LARGE DIAMETER CLAMP DRAWINGS

<u>Clamp Size Diameter x Width (inches)</u>	<u>Type^a</u>	<u>Assembly Drawing Number</u>	<u>Latest Revision</u>
24 x 6	A	1185E72	4
24 x 6	B	1185E73	4
24 x 8	A	1497E27	4
24 x 8	B	1497E28	4
24 x 12	A	1185E76	4
24 x 12	B	1497E31	4
25 x 12	B	1622E99	0
36 x 8	A	1185E78	3
36 x 8	B	1185E79	3
36 x 12	A	1497E29	3
36 x 12	B	1497E30	4
24	Vertical	1499E52	0
24	Primary Cold Leg	1538E29	1

^aRefers to different support arrangement capabilities:

Type A - Deadweight and horizontal (load ear) seismic supports only.

Type B - Deadweight and both horizontal (load ear) and vertical (split line) seismic supports.

the clamp band between the load lugs and the bolt flange gussets are relatively flexible, the distribution of contact pressure acting on the clamp/pipe interface is complex and is a nonlinear function of preload, differential thermal expansion, and externally applied loads. Therefore, a two-dimensional (2-D) finite element model was developed to determine the circumferential pipe-to-clamp (pipe/clamp) interface load distribution due to preloads, external loads, and thermal expansion.

This model utilized beam elements to model the clamp band and the pipe wall, and 2-D constant strain elements to model the gussets and bolt flanges. The insulation bands were modeled with a series of spring-gap elements. The resulting interface load distributions were then applied to a three-dimensional (3-D) shell model of the pipe in a uniform manner over the width of the clamp. The purpose of the 3-D model was to determine the local state-of-stress in the pipe wall. To further refine and verify the analysis methods, a 3-D clamp/pipe interaction model was also developed.

All analyses were performed using the WECAN finite element computer program [10]. Table 6-4 provides a listing of all structural analyses and calculations performed on the HTS pipe clamps during the design process.

6.2.2.1 2-D Clamp/Pipe Interaction Model

As stated previously, the purpose of the 2-D models was to determine the circumferential interface load distribution between the pipe and the clamp. A typical model is shown in Figure 6-7. The pipe was represented by beam elements which covered a small arc (5°) with exception of the elements adjacent to the clamp split line (which used a finer arc). Since this was a 2-D model, the bending stiffness of the beam elements were modified to reflect the difference in stiffnesses between the equivalent ring (composed of beam elements) and the actual stiffness of the pipe. Matching the stiffness of the equivalent ring with that of the pipe was accomplished by an iterative process.

An effort was made to qualify the 2-D interaction model and assumptions by comparison of results with available data from thermal testing of a prototypic clamp design [1,5]. A comparison of both experimentally derived and

TABLE 6-4
STRUCTURAL ANALYSES FOR CRBRP HTS CLAMPS

<u>Calculation Number</u>	<u>Title</u>
LPE-PD-76-005	HTS Piping/36 inch Clamp Thermal Analysis
LPE-PD-76-020	PHTS Piping Pipe Clamp Design
LPE-PD-76-021	Pipe Clamp Design Bolt Sizes and Washers
LPE-PD-76-022	Pipe Clamp Design Spring Plate Thickness
LPE-PD-76-023	Pipe Clamp Design Rod End Pin Diameter
LPE-PD-77-041	Bolt Preload Requirements for Vertical Pipe Clamps
LPE-PD-77-049	24 inch Horizontal Clamp Stop Pin
LPE-PD-78-007	Pipe Clamp Design
LPE-PD-78-022	Summary of Existing Analyses and Test Results for CRBR Pipe Clamps and Recommended Pipe/Clamp Stiffnesses
LPE-PD-79-001	Review of Existing Documentation on Pipe/Clamp Structural Analyses and Testing, and Recommendations for Subsequent Analytical Models and Load Cases to be considered
LPE-PD-79-008	Structural Analysis of Pipe Clamp Effects on the PHTS and In-Containment IHTS Piping Due to Pipe/Clamp Interaction
LPE-PD-79-019	Pipe Clamp Stiffness and Rotational Clamp Slip Tests and Analysis for 24 inch x 6 inch Type A Pipe Clamp
LPE-PD-79-020	Stress Analysis of Pipe Clamp Friction Test Rig
LPE-PD-80-001	Structural Analysis of the Clamp Effects for Revised Large Diameter Insulated Pipe Clamps on PHTS and In-Containment IHTS Piping
LPE-PD-80-002	Interface Loads for Vertical Clamps
LPE-PD-80-010	Structural Analysis of Revised 24 inch x 12 inch Insulated Pipe Clamps
LPE-PD-80-021	3-D Interaction Model of 24 inch x 12 inch Clamp and Pipe for Inelastic and Equivalent Elastic Analysis for Stress Classification
LPE-PD-80-023	Structural Analysis of PHTS Cold Leg Vertical Clamp

TABLE 6-4 (Continued)

<u>Calculation Number</u>	<u>Title</u>
LPE-PD-80-030	Parametric Studies of Tolerance Stackups for Horizontal Pipe Clamp and Effect on Pipe Stresses
LPE-PD-80-031	Stiff Clamp Design - Sizing of Outer Segment ID
LPE-PD-80-034	Analysis of PHTS Cold Leg Seismic Tie Clamp
LPE-PD-80-038	Analysis of Reactor Vessel Downcomer Clamp
LPE-PD-80-039	Detailed Dimensions for PHTS Cold Leg Restraint Design
LPE-PD-81-004	Reprocessed Stiff Clamp Design Pipe Stress Files Utilizing the Updated Stress Classification Procedure from Calculation (SARB) LPE-PD-80-021
LPE-PD-81-015	Analytical Studies of a Special Horizontal Clamp with a 1.0 inch Thick-Walled Transition Piece for IHTS SWR Transient Loads
LPE-PD-81-019	Equivalent Elastic Analysis for Pipe Stress Classification Due to External Loads Using a 3-D Interaction Model of 24 inch x 12 inch Clamp and Pipe
LPE-PD-81-024	Additional Analytical Studies on a Special 24 inch x 12 inch Horizontal Clamp with a 1.0 inch Thick Pipe Transition Piece
LPE-PD-81-025	Investigation of Maximum SMBDB Loads Applied to a 24 inch x 12 inch Clamp
LPE-PD-82-008	Reprocessed Pipe Stresses Due to Clamp Effects Using a Revised Stress Classification Procedure
LPE-PD-82-011	NF Analysis of Horizontal Pipe Clamps
LPE-PD-82-014	NF Analysis of Modified Horizontal Pipe Clamps
LPE-PD-82-015	NF Analysis of Special 25 inch x 12 inch Horizontal Clamp
LPE-PD-83-005	NF Analysis of IHX Seismic Tie Clamp
LPD-PD-83-006	Reassessment of Clamp Induced Pipe Stresses with respect to Proposed Code Allowables for 316 SS Welded with 316 SS Weld Wire
LPE-PD-83-008	NF Analysis of IHX Downcomer Clamp
LPE-PD-83-011	NF Analysis of Reactor Vessel Downcomer Clamp

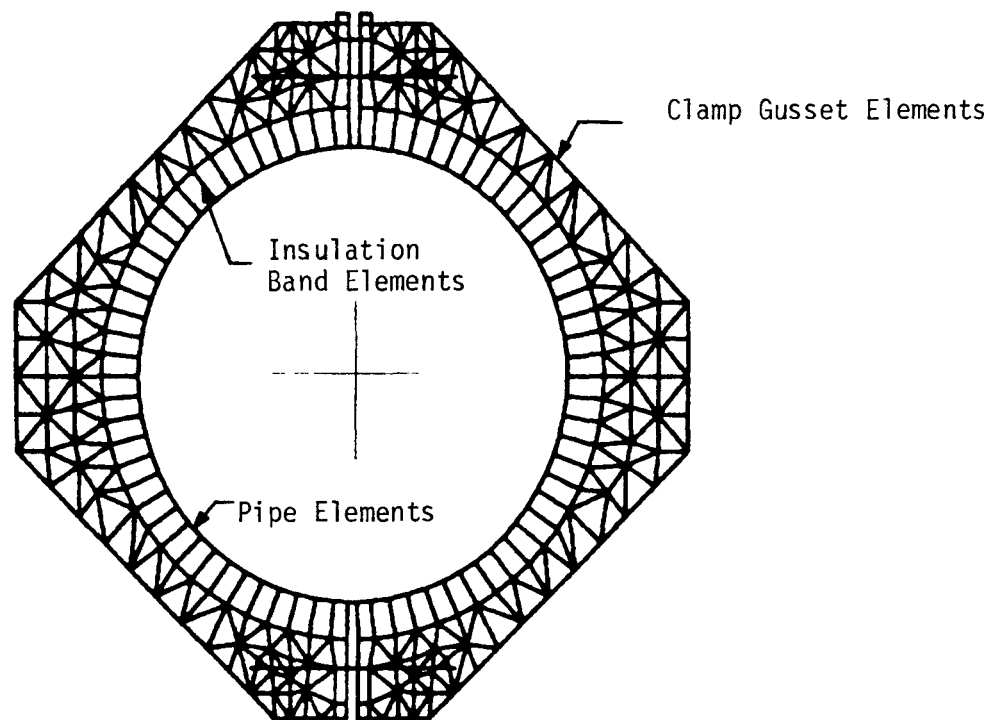


Figure 6-7 2-D Clamp/Pipe Interaction Model

analytically predicted radial deflections had demonstrated that the two deformed shapes were virtually identical. Additional qualifications using the 3-D shell model of the pipe have also validated the 2-D interaction model approach.

The clamp band was also modeled with beam elements having the same arc lengths as the corresponding pipe elements. The attachment pads, bolt flanges, and flange gussets were modeled with shell elements. Actual cross section properties were used with the exception of doubling the attachment pad and bolt flange gusset thickness to account for the presence of two flanges and two load ears at each location.

The clamp band and the pipe beam elements were connected by several dynamic spring-gap elements. The spring represented the equivalent load-bearing insulation (Marinite XL) stiffness, and the gaps were given a distribution which corresponded to the worst case mismatch between the maximum/minimum radii of the pipe OD, the insulation band ID/OD and the clamp band ID. To simplify the analysis, the gap distribution between the insulation band OD and the clamp band ID was simply added to the gap distribution due to the difference in radii between the pipe OD and the insulation band ID.

The Belleville spring stackups and the stop bolt between the upper flanges were modeled with dynamic spring-gap elements also. The dynamic element allowed a negative spring constant to be used for the Belleville spring stackups so that a decrease in the total spring constant could be modeled. The clamp preload was induced in the interaction model by using negative gaps. The stop bolt stiffness was based on an axial loading acting on a cross-sectional area defined by the root diameter of the bolt threads. Since this bolt is to be installed after the clamp is preloaded, the initial gap for analysis was set equal to the differential deflection between the upper two bolt flanges for the preload only case. This means that the stop bolt gap should be zero under the preload only conditions but may open or closed under thermal expansion or externally applied loads.

The 2-D interaction model did not account for the 3-D effects of the clamp or the pipe. These 3-D effects are caused by the fact that there are two separated gussets rather than one solid piece as modeled. Also, the deflections of the pipe wall under the clamp band would be a function of the axial location with respect to the edges of the clamp band. However, a finite element model which would determine both the axial and circumferential interface load distributions would be large and unwieldy. The axial effect discussed above was not considered to be significant, since all of the analyses have shown that the hoop stress is the critical stress component in every load case analyzed. Because the hoop stress component is primarily a function of the circumferential distribution, it should not be greatly affected by the axial variations. This assumption was verified by the development and analysis of a 3-D interaction model discussed in Section 6.2.2.3.

6.2.2.2 3-D Pipe Shell Model

The interface loads obtained from the 2-D interaction model described previously were applied to a 3-D shell element model of the pipe. The 3-D model was used for all load cases which produced symmetric interface load distributions such as preload, deadweight, thermal expansion, vertical seismic loads only, and horizontal seismic loads only.

The interface loads given by the interaction model were actually concentrated nodal loads equal to the integrated pressure loading over the width of the clamp and the average arc length on each side of each spring-gap element. To apply these loads to the shell model properly, the interface loads were concentrated as nodal loads spread uniformly over the width of the clamp band.

The stresses obtained by this procedure are due to the clamp effects only. Therefore, these stress components were combined with the global pipe stress state due to other effects such as deadweight, thermal expansion, internal pressure, through-the-wall thermal gradients, and seismic loads in order to obtain the total stress state in the pipe. Before this was done, the clamp-induced stress components were divided into appropriate categories

(primary membrane, primary bending, and secondary). Then, the local clamp-induced stress components were combined with the global pipe stresses in a manner consistent with the ASME B&PV Code [6,7] as modified by RDT Standard F9-4T [11].

6.2.2.3 3-D Clamp/Pipe Interaction Model

In order to determine the effect of the actual axial interface load distribution on the resultant pipe wall stress distribution, a 3-D model of the clamp/pipe was developed [2]. The finite element model of the three major components for a particular clamp size is shown in Figure 6-8. Modeling considerations are given in Reference 2. Since the large 3-D interaction model was relatively expensive to run, the initial interest was to examine the preload case only. However, additional analyses were performed to confirm the clamp/pipe stiffnesses determined using the 2-D interaction model approach.

As a result of the analyses, it was observed that the axial interface load distribution was not uniform across the width of the clamp as assumed in the previous combined 2-D interaction model/3-D shell model approach. However, examination of the maximum stress locations revealed that the 3-D interaction model, which did include the axial effect, produced significantly lower pipe wall stresses. Thus, the previous combined 2-D/3-D approach was verified and shown to be conservative.

External load cases were performed on the 3-D interaction model to verify the combined clamp/pipe stiffnesses calculated from the 2-D models. As a result, the stiffness values were within 15 percent and therefore validated the analytical methods.

In order to perform inelastic analyses using the 3-D clamp/pipe interaction model and to evaluate stress classification considerations, the linear portions of the model were developed into super elements (substructures) [10].

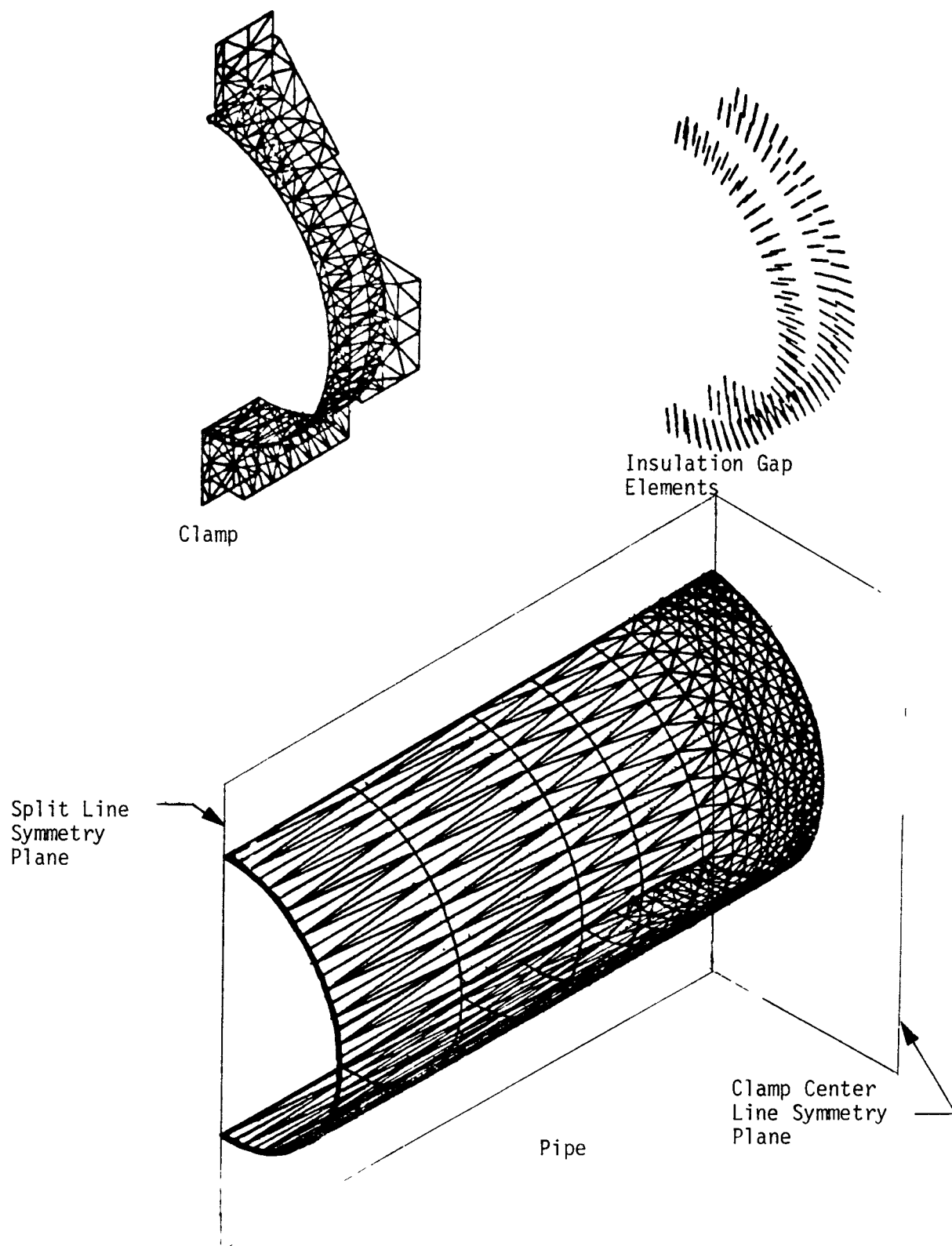


Figure 6-8 3-D Clamp/Pipe Interaction Model

Two typical super elements are shown in Figure 6-9. The use of the super element model offered a considerable cost savings in computer charges without significant loss of modeling detail. The results of the inelastic analyses using the super element model are discussed in Section 6.2.5.

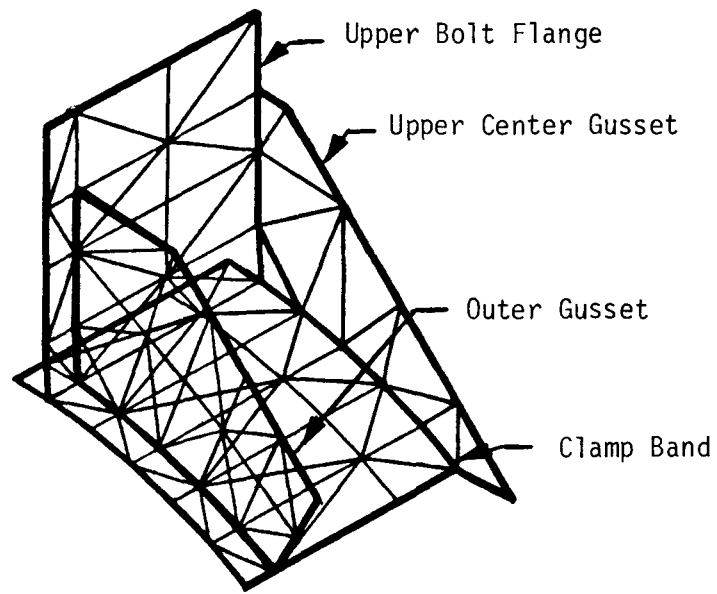
6.2.3 Clamp/Pipe Stiffness and Performance Data

The 2-D interaction models were used to calculate a combined clamp/pipe stiffness using the bolt flange and the load ear deflections from the external load cases. The stiffness values, shown in Table 6-5 for each clamp size, were calculated by dividing the applied load in each case by the appropriate deflection from the 2-D interaction model analysis. Also, as mentioned previously, the clamp/pipe stiffnesses calculated from the 2-D interaction models were successfully verified by comparison with the results from the 3-D interaction models. It should be noted that biaxial variation and the directional dependence relative to clamp stiffness was not as pronounced as observed previously with the flexible clamp design. This is a more desirable effect, especially for seismic considerations.

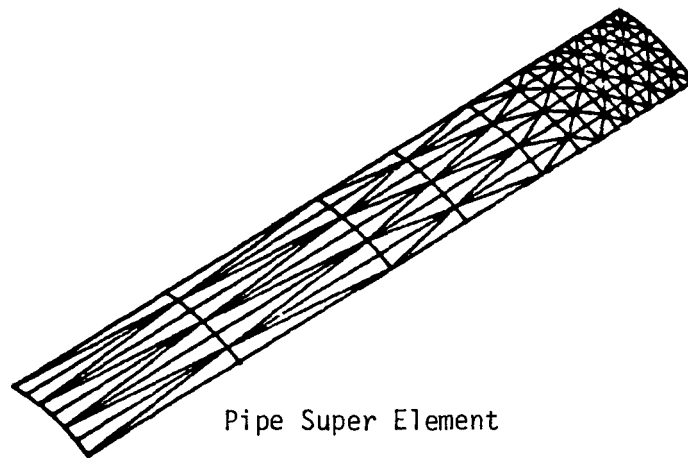
The clamp/pipe stiffnesses were used in the seismic analyses of the Primary Heat Transport System (PHTS) and Intermediate Heat Transport System (IHTS) in-containment piping loops. The stiffnesses were combined in series with the snubber (or rigid strut) and structural steel stiffnesses to obtain an equivalent support stiffness of the support train [12].

The load rating for each HTS clamp size is shown in Table 6-6. It should be noted that these values are dependent upon the CRBRP pipe geometry and not necessarily applicable if these clamps are used on piping with different diameter-to-thickness (D/t) ratios.

There are essentially an infinite number of possible load combinations that each clamp could experience. To select a reasonable number of load cases for analysis, all specified load combinations were examined using the 2-D interaction model for one clamp size (24 inch x 12 inch clamp). Circumferential plots of the interface load distribution for each unique load case were made and compared to determine the bounding load cases (which



Clamp Super Element



Pipe Super Element

Figure 6-9 Typical Clamp and Pipe Super Elements

TABLE 6-5
CRBRP CLAMP/PIPE STIFFNESSES FOR 24 INCH AND 36 INCH HTS CLAMPS

<u>Clamp Size Diameter x Width (inches)</u>	<u>Load Condition</u>	<u>Combined Stiffness (lb/in.)</u>
24 x 6	20K, ^a Compressive, ^b Split Line Load ^c	3.4×10^5
	20K, Tensile, Split Line Load	2.0×10^5
	20K, Compressive, Load Ear Load	1.4×10^6
	20K, Tensile, Load Ear Load	7.5×10^5
24 x 8	26.6K, Compressive, Split Line Load	4.3×10^5
	26.6K, Tensile, Split Line Load	2.5×10^5
	26.6K, Compressive, Load Ear Load	1.6×10^6
	26.6K, Tensile, Load Ear Load	1.1×10^6
24 x 12	40K, Compressive, Split Line Load	5.4×10^5
	40K, Tensile, Split Line Load	3.1×10^5
	40K, Compressive, Load Ear Load	1.6×10^6
	40K, Tensile, Load Ear Load	1.5×10^6
36 x 8	28.8K, Compressive, Split Line Load	2.6×10^5
	28.8K, Tensile, Split Line Load	1.8×10^5
	28.8K, Compressive, Load Ear Load	9.6×10^5
	28.8K, Tensile, Load Ear Load	9.5×10^5
36 x 12	43.2K, Compressive, Split Line Load	3.0×10^5
	43.2K, Tensile, Split Line Load	1.9×10^5
	43.2K, Compressive, Load Ear Load	9.9×10^5
	43.2K, Tensile, Load Ear Load	9.0×10^5

^a Corresponds to clamp preload condition (kips).

^b Relative to clamp/pipe.

^c Attachment location.

TABLE 6-6
LOAD RATINGS FOR CRBRP IN-CONTAINMENT HTS PIPE CLAMPS^a

Clamp Size Diameter x Width (inches)	<u>Load Conditions^{b,c}</u>			
	<u>Preload (lb)</u>	<u>Normal & Upset Loads (lb)</u>	<u>Emergency Loads (lb)</u>	<u>Faulted Loads (lb)</u>
24 x 6	20,000	11,100	13,300	20,000
24 x 8	26,600	14,800	17,700	26,600
24 x 12	40,000	22,200	26,600	40,000
36 x 8	28,800	16,000	19,200	28,800
36 x 12	43,200	24,000	28,800	43,200
25 x 12	65,000	36,100	43,200	65,000

^aThe load ratings are a function of the pipe geometry for thin walled piping.

^bThe loads listed are the maximums for a single load applied either parallel to the clamp split line (split line load) or perpendicular to the clamp split line (load ear load).

^cFor combined loadings, the magnitude of the resultant load vector should not exceed the values listed above.

totalled seven). The interface loads obtained from the seven bounding load cases were applied to the 3-D pipe shell model to determine the critical pipe stresses. As a result, the individual load ratings were determined for each clamp size. The exact procedure is described in detail in Reference 1.

6.2.4 Test Programs

The primary purpose of the tests performed on the CRBRP HTS clamps was to demonstrate structural adequacy of the clamps with particular emphasis on the structural integrity of the load bearing insulation (Marinite XL) [4]. Tests were conducted on the flexible clamp design under combined deadweight and seismic loading conditions at expected CRBRP operating temperatures.

Overall, the testing performed on the clamps included the following considerations:

- A. Mechanical and thermal evaluations
- B. Structural integrity tests
- C. Temperature distribution and thermal cycling tests
- D. Clamp/pipe stiffness tests
- E. Clamp friction and stiffness tests

The following sections highlight the tests performed on the HTS clamps. In summary, the completed testing verified the adequacy of the horizontal and vertical insulated clamps for use on the high temperature, thin-walled HTS piping.

6.2.4.1 Mechanical and Thermal Evaluations

The initial test carried out on the HTS pipe clamps was a mechanical and thermal characterization test of a prototypic horizontal pipe clamp (24 inch x 6 inch) [13]. The clamp assembly was mounted at mid-span on a 6 foot section of 24 inch diameter by 0.375 inch wall stainless steel pipe. The pipe was instrumented with heaters on the inside surface, and thermocouples and strain gages on the pipe and clamp to measure temperature profiles and

strains for various clamping forces and temperatures. The test assembly was insulated with 10 inches of insulation (alumina silica) and suspended from a test structure frame.

The objective of the test was to qualify the load carrying capability of the clamp and to provide clamp heat loss and clamping force data for various pipe temperatures in support of the configuration final design.

The test program also included cold tests and low temperature cyclic tests. For the cold tests, dimensional and strain characterizations of the clamp assembly and the pipe section were recorded initially and at incremental clamp bolt torque (preload) values.

In summary, the above testing demonstrated that clamp spring compression and the change in clamp gap showed linear relationships to bolt torque. Also, temperatures measured on the clamp body and spring assembly did not exceed design values. Upon completion of the tests, visual inspections of the clamp revealed no damage to any of the clamp components from the cold or hot torque tests. The dimensional changes and strain changes resulting from the cyclic tests were also acceptably small.

6.2.4.2 Structural Integrity Tests

The purpose of these tests was to demonstrate the structural adequacy of both the horizontal and vertical pipe clamps when subjected to conditions of physical assembly, temperature cycling, and dynamic loadings which closely simulated CRBRP plant conditions [14]. Particular attention was again given to the structural adequacy of the load bearing insulation (Marinite XL) contained in the insulation bands under the clamp. Separate load-deflection tests were also performed later on Marinite XL as documented in Reference 15.

The horizontal pipe clamp was mounted at mid-span on a section of stainless steel pipe (24 inch diameter by 0.5 inch wall) which measured approximately 6 feet long. The pipe section was secured in a test rig fitted with mounting pads which were used as attachment points for the mechanical snubbers and constant load hangers.

Three dynamic thrusters (actuators) were connected to both ends of the pipe section through a structural beam. The thrusters were used to exert dynamic loads on the pipe clamp. The pipe was insulated in a configuration similar to the CRBRP plant piping and was trace heated on the inner surface to achieve the pipe design temperature of 1015°F (for horizontal clamp test). Thermocouples, strain gages, load cells, and accelerometers were used to measure test conditions and responses.

The vertical clamp assembly inner transition section, which actually forms an integral part of the piping boundary, was welded to two sections of 24 inch diameter x 0.5 inch wall pipe, forming one continuous pipe section approximately 9.5 feet long. The test frame was designed to accommodate two horizontal snubbers (at a 90° angle) and two dynamic thrusters which were connected to both ends of the pipe.

The vertical pipe section was insulated and trace heated internally to achieve a pipe temperature of 775°F (for vertical clamp test). Thermocouples, strain gages, load cells, and accelerometers were installed on the clamp and pipe sections to measure and record test conditions and responses.

The test setup for both clamps were designed to simulate CRBRP operating plant conditions. The test pipe section diameter, wall thickness, and material were identical to the large sodium piping systems. The constant load hangers, attachment hardware, and the snubbers were of conventional designs. The insulation thickness, material, and general design reflected the plant insulation design.

Loadings used in the testing for deadweight and seismic conditions were taken to equal or exceed the design loadings for each clamp. The test sequence for each clamp consisted of the following steps:

- A. Application of deadweight vertical load;
- B. Heat-up of the pipe to the operating temperature (1015°F or 775°F);

- C. Dynamic application of 50 cycles of the Operating Basis Earthquake (OBE) design load;
- D. Dynamic application of 10 cycles of the Safe Shutdown Earthquake (SSE) design load which was 2.0 times the OBE load; and
- E. Shutdown and visually inspect the clamps.

For the horizontal clamp, the visual examination following testing did not show any evidence of damage to the outer clamp bands other than some discoloration of the metal. No cracks or breaks occurred at any of the clamp welds. The canned insulation showed some slightly damaged areas. A localized depression of approximately 0.030 inch deep was seen at the extreme end of the insulation band. The cause for this indentation was the "digging in" of the ends of the outer bands as a result of the preload of the spring plates. The only other observed effect was the failure of a few spot welds along the edges of the bands. These observations concerning the insulation bands did not affect the structural adequacy of the clamp.

For the vertical clamp, visual examination of the outer ring and the inner pipe transition section did not show any effects other than some surface discoloration. No cracks or breaks occurred at any of the welds in these metal parts. Examination of the inner canned insulation bands indicated a raised area on the load bearing side of 0.060 inch high by 9.5 inches long located approximately at the midpoint of the band. This raised section was observed on both bands. The probable cause for this effect was the strong "rocking" motion of the test pipe observed during the dynamic load tests. None of the spot welds on the vertical clamp insulation bands were broken.

6.2.4.3 Temperature Distribution and Thermal Cycling Tests

Thermal cycling and temperature distribution tests were conducted on both the horizontal and the vertical pipe clamps [14]. The purpose was to investigate: (1) the actual surface temperature of both clamps (e.g., the spring plate temperature of the horizontal clamp), and (2) the reaction of the

horizontal clamp to temperature swings while under design loading conditions. Both clamps were set up in their respective test rigs as discussed in Section 6.2.4.2. Static loads of 7,000 lb and 11,400 lb were applied and maintained during the tests on the horizontal and vertical clamps, respectively, with the constant load hangers.

For the horizontal clamp, the temperature of the pipe was increased to 400°F at a rate of approximately 40°F per hour and held constant until the temperatures on the clamp body stabilized. The temperature was then increased in steps to 775°F and finally to the design temperature of 1015°F at a rate of 40°F per hour. The pipe temperature was then reduced to ambient by means of forced draft cooling through the inside surface of the pipe. These three temperature plateaus, which simulated sodium fill temperature, hot functional test temperature and operating temperature, were repeated ten times.

For the vertical clamp, the temperature of the pipe was increased to 400°F at a rate of approximately 40°F per hour and the clamp temperature allowed to stabilize. The temperature was then increased to the design temperature of 775°F and cooled to ambient. This temperature cycle was repeated ten times as with the case for the horizontal clamp test. Throughout the temperature test, visual inspections were performed, and strain gage and temperature readings were monitored and recorded.

The clamp surface temperatures obtained from the tests were in close agreement with predicted temperatures. Thermal growth of the pipe was shared equally by each set of springs on the horizontal clamp. Stress levels, as measured by the strain gages, were in general agreement with predicted values. Visual examination of both clamps after disassembly did not reveal any structural damage.

The only unexpected result occurred during thermal cycling of the horizontal clamp. During the cool-down phase of the temperature cycling, it was observed that the two clamp halves shifted around the pipe in the direction of the load. The shift came in incremental steps and was accompanied by a "snapping"

sound. The tendency of the clamp to shift was alleviated by the addition of a stop screw located on the top spring plate of the outer band. The stop screw was added to all clamp sizes.

6.2.4.4 Clamp/Pipe Stiffness Tests

The purpose of these tests was to verify the clamp/pipe stiffness characteristics obtained from analytical models described in Section 6.2.2 [16,17,18,19,20]. As before, a prototypical 24 inch x 6 inch HTS clamp was mounted on a straight piece of pipe approximately 4 feet in length and placed in a standard MTS test machine. Loads were applied to the clamp (up to the clamp rated load value) and displacement measurements were taken.

From the load-displacement data, the clamp/pipe stiffness was calculated. Under tensile loads, it was observed that significant hysteresis was present due to the preloaded Belleville springs. The tests also confirmed the expected bi-directional stiffness effect and approximately verified the predicted stiffness values obtained from the analytical models described in Section 6.2.2.

6.2.4.5 Clamp Friction and Stiffness Tests

The ability of a friction type clamp to function under seismic loads without rotating about the pipe depends on the magnitude of two opposing forces. One is the overturning force or moment exerted upon the clamp as a result of a misalignment between the clamp and pipe center line and the center line of the seismic restraint/embedment. The other is the resisting friction force or moment generated by the clamp preload force. As long as the clamp friction forces are greater than the vertical component of horizontal seismic force, the clamp will not slip (rotate).

The purpose of the clamp/pipe friction tests were twofold: first, to demonstrate that an overturning vertical load (equivalent to the vertical component of the 7° permissible seismic force offset) could be resisted by the clamp friction force; and secondly, to evaluate the effect of horizontal clamp loading on the friction forces [21].

The pipe/clamp was mounted in a test fixture made up of a section of 24 inch pipe and measured approximately 6 feet long. The test fixture was designed so that deadweights could be suspended from one ear of the pipe clamp while a pre-determined horizontal load could be applied to the same clamp ear.

To demonstrate resistance to slip under a 7° mismatch, a vertical load was incrementally added to the load ear until a load of 2,465 lb (the capacity of the test jig) was reached. The clamp did not slip at this load. The 2,465 lb load represented a 20,230 lb horizontal applied force offset 7° from the horizontal.

To demonstrate the effect of horizontal side loading on the clamp friction force, the clamp was set up with a horizontal tensile load applied at one load ear. In this case, the preload on the clamp ears was reduced, and an approximately equal horizontal load was imposed on the clamp ear. The reduced preload (and hence the horizontal load because of the preload-to-horizontal-load relation of the clamp) was considered because of the anticipated high friction loads. Under these conditions, a vertical load was incrementally applied until a rotational slip was observed. Clamp slip was accompanied by an audible "pop" from the test assembly. A vertical load of 2,316 lb was required to produce a rotational slip of the clamp on the pipe.

Additional stiffness tests were performed by applying a horizontal load to the clamp ear and measuring displacements in the friction test rig. The resulting stiffness data was considered a more representative model of the expected pipe/clamp behavior than obtained from previous stiffness testing. This data compared favorably to the stiffnesses obtained from the analytical models described in Section 6.2.2.

6.2.5 Primary/Secondary Stress Classification

The design of reasonably conservative piping systems requires careful classification of clamp-induced stresses into primary (P) and secondary (Q) stress categories [22]. The basic characteristic of a primary stress is that it is not self-limiting (see Subsection NB-3213.8 of [6]). Primary stress is

not redistributed or relieved by inelastic deformation. The behavior is equilibrium-controlled or load-controlled and primary stresses may be alternatively described as equilibrium-controlled or load-controlled stresses. The basic characteristic of a secondary stress is that it is self-limiting (see Subsection NB-3213.9 of [6]). Secondary stress is caused by displacement constraints and is redistributed or relieved by inelastic deformation. This behavior is compatibility-controlled or deformation-controlled and secondary stresses may be alternatively described as compatibility-controlled or deformation-controlled stresses.

The importance attached to the classification of induced stresses derives from their relationship to modes of failure. Primary stresses which can result in failure from a single application of load are limited to relatively low allowable values. Secondary stresses which can result in failure only from repeated load cycles are less severely limited. It is the range and number of cycles of secondary stress (rather than the level of stress) that are of particular importance in preventing failure by exhaustion of ductility or gross distortion.

In the FFTF, the clamp induced pipe stresses were conservatively designated as primary. For CRBRP, the clamp preloads as well as the hanger and snubber loads were significantly higher than those in FFTF. Consequently, it did not appear to be possible to assume that the local clamp-induced pipe stresses are primary and still comply with the ASME B&PV Code [6] primary stress limits.

Initially in CRBRP, the stresses induced in the pipe by loads such as preload, deadweight, and seismic were considered to be largely primary, and the stresses due to thermal expansion were secondary. However, since the clamp restricted the pipe deformation greatly, any localized yielding of the pipe wall would be partially alleviated by the ability of the clamp to redistribute the interface loads around the circumference of the pipe. This implied that some portion of the clamp-induced stress state was secondary in nature. Since the exact amount of secondary stress could not be initially determined and because it was not conservative to consider all of the clamp-induced stress as secondary, simplified procedures were used to separate pipe stresses into primary and secondary components [1,2,5,22].

To resolve the stress classification question for CRBRP HTS clamps, both inelastic analysis and nonlinear elastic parametric studies were required. A 3-D interaction model with linear portions represented by super elements was developed (as described in Section 6.2.2.3).

Because it was difficult to determine a specific primary to secondary split based on the inelastic analysis and since it was very expensive to perform inelastic parametric studies, a simplified equivalent elastic analysis technique was developed and utilized to simulate the inelastic results as discussed in the following section.

The equivalent elastic analysis was based on the concept that the local material flexibility in the plastic region could be simulated by lowering the elastic modulus of the pipe.

This simplified procedure required local stress predictions from a number of elastic analyses with different (more flexible) elastic moduli to simulate the material nonlinearity in the pipe. The adequacy of this approach was confirmed by comparing the equivalent elastic results with those obtained by detailed inelastic analysis.

The simplified equivalent elastic procedure did not depend upon the specific clamp design or the pipe geometry. The classification of stresses into primary and secondary categories is based upon the ASME B&PV Code definition of the load-controlled and the deformation-controlled quantities. To establish an acceptable but general primary-secondary split of the local stresses required parametric studies with various clamp designs and applied loadings.

Based upon the analytical studies, it was concluded that the membrane portion of the clamp-induced stresses should be treated as a primary stress and the bending portion was essentially secondary. However, to ensure that a moderate amount of conservatism remained in the treatment of the pipe stress due to clamp effects, a judgment was made classifying the bending component as one-third primary and two-thirds secondary. This classification procedure was

utilized when the pipe stress components due to clamp effects were combined with the global pipe stress state due to operating conditions to satisfy the ASME B&PV Code requirements [6,7].

6.2.6 Final Design Status

The design of the large diameter HTS clamps was essentially complete at the time of CRBRP termination in November 1983. All of the drawings (design layout, assembly, and detail) and design related documents were up to date and checked. The ASME B&PV Code analysis reports (Subsection NF) were in process but not completed. However, the basis for these reports were the calculations identified in Table 6-4 and the reports referenced in Section 6.2.7. It is assumed that these NF stress reports could have been successfully completed in order to qualify the HTS clamp designs.

The final design review on the HTS support hardware (which included the large diameter pipe clamps) was scheduled after CRBRP Project termination and therefore did not occur. However, it is believed that the design review would have been successful in verifying the adequacy of the HTS clamp designs to meet the specified requirements and allow release for fabrication of plant units.

6.2.7 References

1. Jones, G. L., "Report on the Stresses Induced in Liquid Metal Piping Due to Insulated Pipe Clamps for the Clinch River Breeder Reactor In-Containment Pipe Loops," ES-LPD-79-003, January 1980.
2. Jones, G. L., "Report on the Stresses Induced in Liquid Metal Piping Due to Insulated Pipe Clamps for the Clinch River Breeder Reactor In-Containment Pipe Loops - Stiff Clamp Design," ES-LPD-80-001, February 1981.
3. Pollono, L. P. and Mello, R. M., "Design Considerations for CRBRP Heat Transport System Piping Operating at Elevated Temperatures," 79-NE-5, ASME, New York, 1979.
4. Pollono, L. P. and Mello, R. M., "Design and Testing of CRBRP Insulated Horizontal and Vertical Pipe Clamps," in Effects of Piping Restraints on Piping Integrity, PVP-40, pp. 177-200, ASME, New York, 1980.

5. Jones, G. L., "An Evaluation of Clamp Effects on LMFBR Piping Systems," in Effects of Piping Restraints on Piping Integrity, PVP-40, pp. 201-218, ASME, New York, 1980.
6. ASME Boiler and Pressure Vessel Code, ANSI/ASME BPV-III-NCA, Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components," 1974 edition with addenda through Summer 1975, ASME, New York, 1974.
7. "Class 1 Components in Elevated Temperature Service, Section III," Code Case 1592-7, ASME Boiler and Pressure Vessel, December 22, 1975.
8. "CRBRP Heat Transport System Piping Design Spectification," E-953089, Revision 37, June 1983.
9. "Pipe Clamps for Sodium Service," E-953132, Revision 5, May 1983.
10. Proprietary
11. "Requirements for Construction of Class 1 Elevated Temperature Nuclear System Components (Supplement to ASME Code Cases 1592, 1593, 1594, 1595, and 1596)," RDT Standard F9-4T, Amendment 4, April 1977, Oak Ridge National Laboratory, Reactor Standards Office, Oak Ridge, TN.
12. Mello, R. M. and Pollono, L. P., "Evaluation of the Influence of Seismic Restraint Characteristics on Breeder Reactor Piping Systems," in Piping Restraint Effects on Piping Integrity, PVP-37, pp. 63-78, ASME, New York, 1979.
13. Letter T-SCT-DKS-78-3, D. K. Schmidt to L. P. Pollono, et al., "Test Report, Pipe Clamp Test," dated January 6, 1977.
14. Gaydos, R. A., "Seismic Test of Clinch River Breeder Reactor Program Primary Horizontal and Vertical Pipe Restraint Systems," EL:785, July 1978.
15. Letter LPD-79-133, L. P. Pollono to R. H. Mallett, "Load-Deflection Test Results of Marinite XL Load-Bearing Insulation," dated July 19, 1979.
16. Letter LPD-78-017, R. H. Mallett to R. W. Devlin, "Preliminary Clamp/Pipe Stiffness Test," dated January 30, 1978.
17. Letter ARD-ME-BG-78-14, B. Gowda to L. P. Pollono, "CRBRP Pipe/Clamp Tests, W.O. 30958, dated January 4, 1978," dated May 15, 1978.
18. Letter LPD-78-164, Rev. 1, G. L. Jones to R. H. Mallett, "Effective Support Stiffness for CRBR Piping," dated December 11, 1978.
19. Letter LPD-79-227, G. L. Jones to R. H. Mallett, "Combined Clamp/Pipe Stiffnesses for CRBRP In-Containment Piping," dated December 7, 1979.

20. Letter LPD-79-054, G. L. Jones to R. H. Mallett, "CRBRP Pipe Clamp," dated March 29, 1979.
21. Letter LPD-79-220, G. M. Hulbert to R. H. Mallett, "Pipe Clamp Friction and Stiffness Tests," dated November 27, 1979.
22. Jones, G. L. and Dhalla, A. K., "Classification of Clamp Induced Stresses in Thin-Walled Pipe," 81-PVP-17, ASME, New York, 1981.

7.0 INTERMEDIATE HEAT TRANSPORT SYSTEM

7.1 Intermediate Heat Transport System and Piping Design

The Intermediate Heat Transport System (IHTS) transports reactor generated heat from the Primary Heat Transport System (PHTS) to the Steam Generator System (SGS) for production of superheated steam at the temperature and pressure required for turbine operations. During shutdown the IHTS transports reactor decay heat to the SGS for subsequent rejection in the Steam Generator Auxiliary Heat Removal System (SGAHRs) or in the main turbine condenser.

7.1.1 Summary of Functions and Design Description

7.1.1.1 Functions

The functions of the IHTS are as follows:

- A. Transports reactor generated heat (975MWt at full reactor thermal power) from the PHTS to the SGS while maintaining an adequate flow rate for controlling reactor temperature conditions within limits which prevent damage to the reactor vessel, fuel, and reactor internals.
- B. Provides sodium flow, in response to plant process control, over the operating power range of 40 to 100 percent reactor thermal power.
- C. Transfers decay heat from the PHTS to the SGS under all normal and off-normal conditions including failure of a heat transport system component loop.
- D. Provides containment of sodium coolant by providing a boundary for IHTS coolant confinement.

E. Provides capability for rapid drainage of the sodium coolant to the sodium dump tank (provided by the SGS).

F. Provides an extension of the reactor containment boundary.

7.1.1.2 Design Description

The IHTS consists of piping and components required to transport 975MWt of reactor generated heat from the PHTS to the SGS. A detailed description of the IHTS design and operation is given in Reference 1. The IHTS is designed to support operation over a range of 40 to 100% rated power and to support decay heat removal during off-normal operating or shutdown conditions. The IHTS is comprised of three independent piped loops, each thermally coupled to the PHTS by an Intermediate Heat Exchanger (IHX) located in the Reactor Containment Building (RCB) and to the SGS superheater and evaporators located in the Steam Generator Building (SGB). As shown in Figure 7-1, non-radioactive sodium coolant is circulated by a sodium pump located in the cold leg of each loop through the tube side of the IHX and the shell side of the superheater and evaporators. A permanent magnet (PM) flowmeter, located in the cold leg, is used to control pump speed for the purpose of balancing heat delivered to each IHTS loop. The IHTS is a closed loop system with an expansion tank and argon cover gas to accommodate thermally induced system volume changes. The arrangement and relative elevation of the IHTS piping and components are designed to promote natural circulation for decay heat removal.

The three IHTS loops are completely independent with no direct inter-connections. Separate cells are provided for each loop both within and outside of containment. The IHX tube walls provide a passive barrier for isolation of the activated primary sodium from the intermediate sodium. As a precaution against the unlikely event of an IHX tube leak, the IHTS pressure is maintained at least 10 psi greater than the PHTS within the IHX. Also radiation detectors monitor the IHTS flow from the IHX for any contamination by radioactive primary sodium.

The in-containment portions of the IHTS consist of main loop hot leg and cold piping between the IHX and the RCB wall penetration. The IHTS shares these nitrogen inerted cells with PHTS equipment. These RCB cells are steel-lined, concrete, shielded structures.

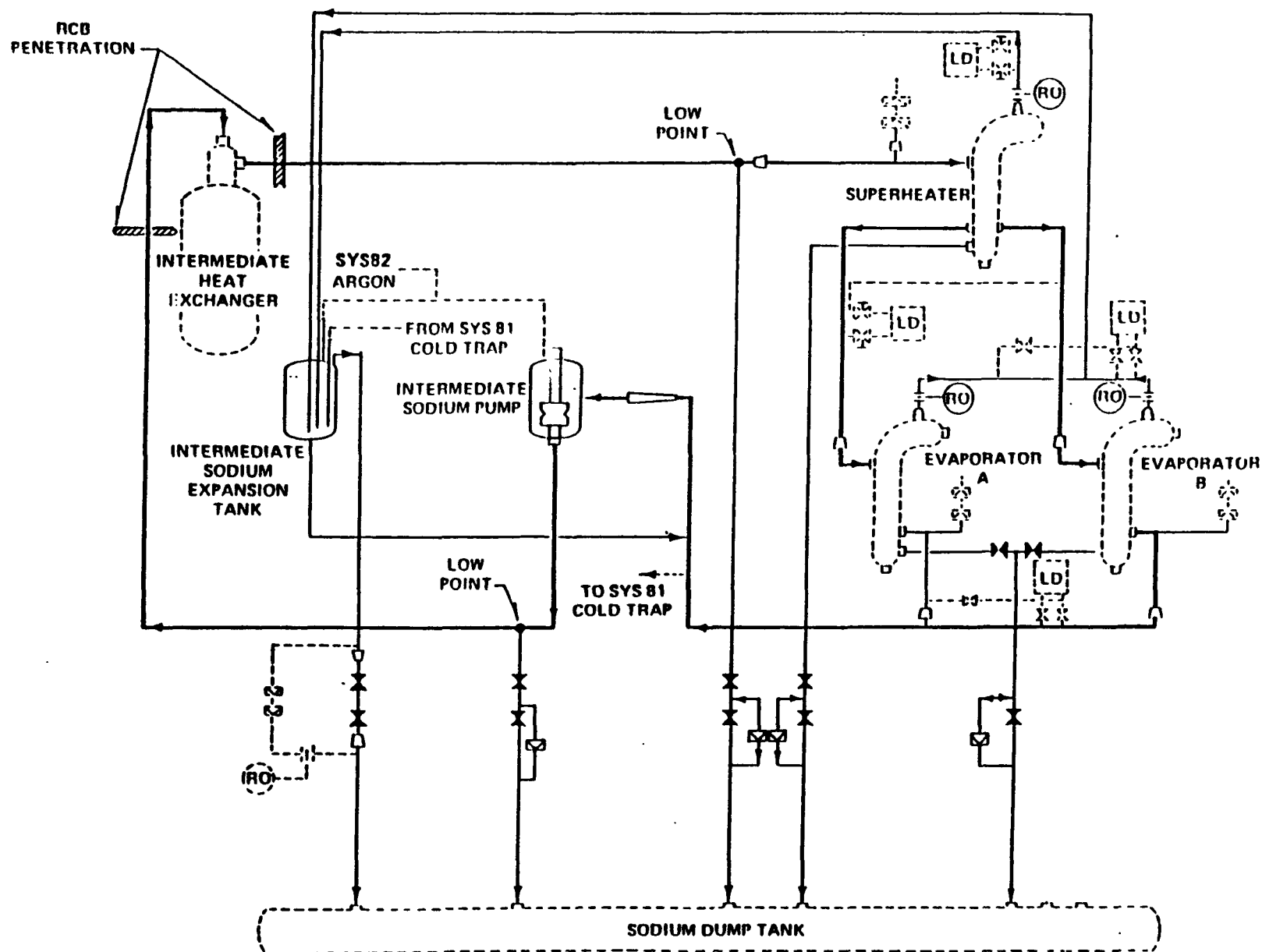


Figure 7-1 CRBRP INTERMEDIATE HEAT TRANSPORT SYSTEM SCHEMATIC

The ex-containment portions of the IHTS are housed in the Intermediate Bay and Steam Generator Bay of the Steam Generator Building (SGB). The IHTS shares the Steam Generator Bay with SGS equipment. The air-filled SGB cells are reinforced concrete structures with thermally insulated, steel catch pans on the floors to protect against sodium spills. The ex-containment IHTS main loop pressure boundary, including passive barriers at the steam generator modules and rupture discs and isolation valves on connecting auxiliary equipment, is considered an extension of reactor containment.

The ex-containment IHTS consists of the main loop hot and cold leg piping from the containment penetrations to the steam generator modules with a sodium pump located in the cold leg. The expansion tank is located off the cold leg line upstream of the sodium pump. The component arrangements and pipe routings in the Steam Generator Bay are essentially identical for the three loops. The IHTS heated sodium exits the IHX, passes through the containment penetration, and flows through several thermal expansion loops to the superheater inlet. The partly cooled sodium is divided as it leaves the superheater and flows to the two evaporator inlets. The sodium from each evaporator is combined through a mixing tee and then enters the intermediate coolant pump suction. The sodium from the pump discharge returns through the other containment penetration and completes the circuit through the IHX. The piping between the superheater and evaporators and between the evaporators and mixing tee are symmetrical for flow equalization to minimize thermal transients.

The main loop employs 24-inch OD hot leg piping between the IHX and superheater, 18-inch OD piping between superheater and evaporators and between evaporators and the mixing tee and 24-inch OD piping between mixing tee and IHX. Expansion loops provide the flexibility to accommodate thermal expansion. Penetrations through the RCB are made with rigid flued heads which are an integral part of the piping welded to the containment structure for containment sealing. The IHTS piping is supported and restrained by constant load hangers, mechanical snubbers, rigid restraints, axial restraint fittings and specially designed spring-loaded clamps to meet piping stress criteria and interface connecting equipment loads. The clamp design incorporates a canned, load-bearing insulation between the clamp and the pipe to reduce stresses in the pipe wall during temperature transients by isolating the large clamp thermal mass from the relatively small pipe thermal mass.

Other major components in the IHTS are the counterflow shell-and-tube IHX (provided as part of the PHTS), the hockey stick type shell-and-tube steam generator modules (provided as part of the SGS), cylindrical sodium expansion tank and the centrifugal pump driven by variable speed drive. All of these components are vertically oriented. The pump is described in Section 7.3 of this report.

Auxiliary systems connected to the IHTS main loop include a sodium dump/drain system with dump tank, sodium/water reaction pressure relief subsystem, steam/water-to-sodium leak detection subsystem, recirculating cold trap system, trace heating system, sodium-to-gas leak detection system, sodium pump drive system, and instrumentation.

All materials for the IHTS piping and components are specified to minimize corrosion and erosion and ensure compatibility with the environment. Hot leg piping and fittings are Type 316H SS; cold leg piping and fittings are Type 304H SS. Piping between the superheater and evaporators is 2 1/4 Cr-1Mo ferritic steel. Trimetallic joints are used to make a transition from ferritic steel components to austenitic stainless steel piping with a short alloy 800H (Ni-Fe-Cr) pipe section used as an intermediate coupling. The IHTS Expansion Tank is constructed of Type 304H SS. Parts of the IHTS pump in contact with the sodium are Type 304, Type 304H, and Type 316H SS except for bearings and special parts. The sodium valves in the IHTS are constructed primarily of Type 316H SS. Other materials in contact with the sodium are special materials such as the Stellite valve seats.

7.1.2 Evolution of Design

7.1.2.1 System Design Evolution

The basic IHTS system concept was established in 1974 as described in Reference 2. Subsequently the IHTS system design was baselined in December 1976 [3]. Thereafter, the system evolved to its final state as thermal/hydraulic analyses were completed, sodium fire analyses were performed to provide information for the SGB design and final operating procedures and maintenance procedures were finalized. The system design and operating/maintenance procedures were adjusted in response to various IHTS system design reviews, key system reviews

and Alara reviews. The final IHTS system design is described in Reference 1.

The Final Working System Design Review (FWSDR) for the IHTS was conducted in December 1982 [4] [5]. All comments from the FWSDR were resolved, principally through the subsequent revision of the System Design Description (SDD) draft Sections 3, 4, 5 and 6. These sections were submitted in 1983 to Westinghouse for approval to baseline [6] [7] [8] [9].

7.1.2.2 Piping and Component Design Evolution

The design requirements for the IHTS piping were mutually established with the PHTS piping and were contained in the Heat Transport System Piping Specification [10].

The design layout of the piping system was initially established to satisfy flexibility requirements for accommodating elevated temperature expansions of both the piping and the supporting components. Rod and constant load spring hangers were provided to provide support for the weight of the piping system including pipe, clamps, fluids, insulation, heaters, etc. Coupled with fluid pressure requirements this initial phase of the design satisfied load controlled requirements of the ASME Code Section III Class 1 for Elevated Temperature Components (Code Case 1592) with anticipated reserve for short term dynamic loadings and strain controlled requirements for accumulated creep and fatigue damage.

Design evolution from this point was essentially the result of development of the support system for the piping for dynamic load events. The dynamic load events that dictated the design of the supports were seismic, sodium/water reaction (SWR) including pressure relief from the sodium water reaction pressure relief subsystem (SWRPRS) and steam generator response to steam/water pipe break loads. This evolution was multifaceted in that iteration and progressive revisions to the baselined design by way of Engineering Change Proposals resulted from:

- o Revision of design basis OBE and SSE seismic spectra reflecting both increased ground accelerations and building/basemat interaction [11].

- o Revision of SWR design basis leak (DBL) from 7 tube instantaneous rupture to 3 tube mechanistic rupture [12].
- o Implementation of dynamic coupling in analytical modeling and analysis to satisfy NRC Reg Guides as incorporated in the seismic criteria document WARD-D-0037 [13]. The need for extensive coupling became evident as interface information on the frequency, mass and local stiffness of supporting components and building structure became available.
- o Iteration of interface requirements between piping and components, piping supports and building auxiliary structure.

The major revisions to the piping and support arrangement baselines were accomplished through ECPs G0613 [14] for the large diameter (8-inch and larger) piping and G0555 [15] for the small piping. The piping arrangements established by these ECPs were the bases for performing the load controlled stress analysis which formed the basis for the fabrication release of the piping system. Final design reviews were held for the main loop piping [16] and for the auxiliary piping [17].

Concurrent with the evolution of the piping system design was the development of specific unique piping components which were an integral part of the system.

These included:

- o Trimetallic transition joints for connecting austenitic stainless steel piping to 2-1/4 CR-1Mo steam generators and carbon steel dump tanks.
- o Cold stand-off pipe clamps utilizing load bearing insulation for connection of snubbers, hangers and rods to provide piping system support.
- o In-line axial forgings (restraint fittings) for connection of paired snubbers to react high SWR axial thrust loads.
- o Mixing tee to combine the sodium flows from the two evaporators and provide a uniform stream to the pump.

7.1.3 Key Problems and Resolutions

The key piping design problems were associated with providing support for dynamic loadings. The key system problems involved an intermediate side vent for the IHX, margin for the main loop rupture disks during an SSE and the "hot tee" condition created by the SWR DBL.

7.1.3.1 Accommodation of Sodium-Water Reaction Loads

Among the requirements for the CRBRP IHTS piping design is a set of dynamic loads generated by the SWR associated with the postulated tube ruptures occurring in the steam generator (i.e., SWR DBL). Even though the rupture disks and associated pressure relief system limit the consequences of an SWR, the presence of acoustic pressure pulses propagating throughout the system causes the SWR event to become a design limiting event for the IHTS piping. Furthermore, the design characteristics necessary to deal with SWR loading are not necessarily the same as those design characteristics necessary to deal with seismic loading.

The SWR is propagated rapidly through the sodium system in the form of a pressure pulse that produces a dynamic reaction in the IHTS piping and support structure. An additional consequent load pattern is produced by the rapid formation of the SWR gas bubble causing the expulsion of sodium and reaction products out through one or more of the main IHTS rupture disks and into the SWRPRS. The structural forces generated in the IHTS by the SWR were initially considered using a seismically optimized configuration and shown to yield significantly higher than allowable piping stresses, equipment nozzle loads and restraint forces when compared to the appropriate ASME Nuclear Code criteria.

Modifications were made to the IHTS piping design to account for the SWR generated loads. It should be noted that the total SWR generated energy quantity remains the same, and that all modifications are limited to optimization of the structural system characteristics and refinement of the analytic procedures. The combined effect of analysis and structural modifications indicate significantly reduced loads at selected critical points in the IHTS piping system.

The analysis of the complex interaction of forces initiated by the SWR was accomplished through use of the hydrodynamic transient analysis program "TRANSWRAP II" [18]. The TRANSWRAP II program was developed as an analytical tool specifically to investigate and identify the SWR load patterns. The GSAP4 structural dynamic computer program was utilized with TRANSWRAP generated SWR load histories to calculate the structural response. The GSAP4 program is a special General Electric proprietary program for use in piping structural analysis.

The intense SWR pressure pulse and resultant fluid momentum loads create relatively high differential axial load components throughout the IHTS. The unique nature and magnitude of the axial loads introduces unacceptably high load stresses in the pipe when an attempt is made to react it out using only lateral-clamp restraints. Additionally, the lateral clamp assemblies commonly utilized in high temperature sodium systems consist of a load bearing insulation clamp which results in a "soft" spring rate relative to that obtainable with non-insulated designs. In order to limit and balance the effects of these SWR induced forces, several measures were employed in the piping system. Chief among these measures was the utilization of in-line axial pipe restraints in combination with the usual lateral pipe clamp attachments. The inclusion of axial restraints restricts motion in that direction, thereby absorbing the axial pressure and momentum load component at the source. The increased stiffness characteristics exhibited by the relatively solid axial connection also serves to reduce the load effect of dynamic excitation produced by seismic response spectra input. The axial restraint stiffness rate used for the current analysis is an order of magnitude greater than the equivalent stiffness obtained through use of the lateral clamp arrangement.

The SWR loads imposed on IHTS equipment nozzles can be particularly high on the steam generators in which the tube break is postulated to occur. A significant factor in consideration of nozzle loads is axial motion of the steam generator. This cyclic motion is caused by the initial pressure peak impacting and reflecting within the vessel. The relatively high translational movement at these points require a sufficiently flexible length of pipe adjacent to the nozzle. This means restraints had to be located remote from the nozzle in order to provide this flexibility.

An additional load mitigating effect was obtained by the optimization of straight pipe run lengths. The longer the length of a pipe run between elbows, the further apart the peak pressures of the dynamic response in that pipe leg. The resultant magnitude of the differential pressure history curve tends to increase as the transit times increase. Thus, it is advantageous to avoid inordinately long runs between elbows.

Utilizing the procedures noted in this section, a significantly improved balance of SWR loads can be achieved. The system loads are significantly lowered in the critical nozzle regions at the expense of only moderate increase in an isolated number of piping components and restraints. The overall impact is a general transference of load from more critical to less critical regions in the piping system.

The SWRPRS pressure relief lines exist as a safeguard to overpressure in the IHTS. The IHTS pressure boundary is maintained by duplex rupture disk assemblies which provide the overpressure protection. The disk rupture occurs through buckling of the disk which then impacts a cutting edge and opens to flow. The structural response to an SWR event is dependent on the rupture disk behavior. How the rupture disk is modeled in analyses will affect the determined load response. For CRBRP analyses, the TRANSWRAP II code used a theoretical model which has been modified using Large Leak Test Rig (LLTR) Series II Test program data to arrive at a validated model [19].

7.1.3.2 Transition Weld Joints for Joining Stainless Steel-to-Ferritic Steel

The transition weld joints (TWJs) are located in the CRBRP IHTS where the material transitions from stainless steel to 2-1/4 Cr-1Mo or from stainless steel to carbon steel. The two critical locations are at the superheater inlet and the superheater vents.

The first interim stress report was completed for the IHTS TWJs, and a preliminary design review was conducted. The design review determined that additional analysis was required prior to proceeding into fabrication. The fabrication equipment specification was released.

The major change with the TWJs occurred in the design integration with the piping system application. The initial arrangement had larger steam generator nozzles than piping and used reducers between the steam generator and piping. The TWJs were on the small diameter side. In this position they could not be shown to meet load controlled stress requirements. The TWJs were then moved to the larger diameter side of the reducer and were able to show acceptable margins.

The inelastic analysis for the superheater inlet TWJ determined that a conservatively estimated life would be at least 15 years. Further refinement of the analysis may increase this value. The elastic analysis of all of the plant TWJs was performed under a subcontract to Foster Wheeler Energy Applications. Completion of the elastic analysis provided the basis to release the transition joints for Project use in September 1981 [20].

Evaluation of the Transition Joint Life Test (TJLT) [21] was completed and the results demonstrated that the ASME Code method of design has a safety factor of at least two on failure. From these results, the 26-inch diameter TWJ and the 3-inch superheater vent TWJ will be ASME Code stamped for a design life of 15 years. All others were shown to be good for a 30 year life. Plans were underway to establish methods for ultrasonic inspection of these joints during the plant life.

7.1.3.3 Large Pipe Clamps

Pipe clamps are non-integral attachments to the pressure-containing boundary, to be used to transmit the loadings from the effects of piping dead weight, thermal expansion, seismic events, sodium/water reactions, and other dynamic loads through piping restraints to the building structural steel.

Clamps were designed for use throughout all IHTS pipe lines except the 36-inch line from the diffuser to the sodium pump.

The design concept was originally developed for FFTF and was modified by Westinghouse into a "compliant" cold stand-off clamp. The load ratings of those clamps were too low for the sodium/water reaction loads found in the IHTS piping system, so further modifications were made by General Electric to produce a clamp design that would not transmit loads through the "split line" attachment. Due to the high temperature and rapid temperature transients of the IHTS main-loop piping, a cold stand-off type pipe clamp was designed to meet the service conditions. The clamp has two half rings held in place by a preloaded belleville spring assembly. It is insulated between pipe and clamp rings by encapsulated load-bearing insulation to minimize direct temperature effects from piping. It has uniform stiffness and produces relatively low local stresses in the pipe.

Another feature of the General Electric design was the need for many more multiple load pickup points on each clamp than the Westinghouse design provided. This removed the need for closely spaced clamps.

A three-dimensional substructured nonlinear analytical method was developed to analyze and size clamp under any loading combination. A preliminary stress analysis was completed for a nominal 24-inch pipe and clamp. The results of the three-dimensional analysis showed local pipe stress, when combined with maximum stresses at clamp location from piping system loads are within acceptable limits allowed by the ASME code. Clamp stresses showed overstressed areas in five components of the clamp requiring design modifications.

Reanalysis subsequently performed on the 24-inch clamp disclosed two overstressed conditions. A material change with a minimum S_m requirement of 25,000 psi and a reduction of the preload eliminated both overstressed conditions. This analysis also concluded that the remaining size clamps (26-inch, 18-inch and 8-inch) would also satisfy their design loads.

Design drawings for all clamp sizes were issued as draft for use in design and analysis. Preliminary stress analysis of two sizes of clamps was performed using the three-dimensional clamp and pipe model. Based on this preliminary analysis, a method was developed to evaluate each clamp location on all large pipe. The results were used to support the IHTS main loop piping design review [16].

The clamp intervening element was tested [22] to establish material properties and failure criteria for use in the clamp analysis. Preliminary analysis of the large clamp was completed [23] and the large clamp design was released for procurement following a design review [24].

7.1.3.4 Axial Restraint Fitting

The axial restraint fitting is a special design, forged pressure-containing component. It has a thicker wall section than the pipe to provide additional strength and integral restraint attachments diametrically arranged to provide restraint attaching points on the pressure-containing boundary. The fitting is to be welded to the IHTS main-loop piping at locations where axial restraining effects are needed to limit pipe movement resulting from high axial sodium/water reaction loads.

The design of the fitting was analyzed using three-dimensional finite-element linear analysis and was shown to provide adequate restraining effect under the representative support load without detrimentally affecting the integrity of the piping system. A preliminary stress report was prepared. The report shows that, for the representative support loads used in the analysis, the piping stresses resulting from various loading effects are within acceptable screening limits. Final inelastic analysis with final piping reaction loads will be required to show the axial restraint fitting design acceptable.

The axial design was analyzed using screening rules derived from detailed finite element modeling and loads from the piping system analysis computer runs for the lines where these axials are located. The detailed finite element study of the axial design identified critical locations. The screening equations in the computer code, ASEP, evaluate the stress state at these locations under various load and temperature conditions.

The analysis conclusion is that IHTS lines D and E contain no axial restraint fitting incapable of passing the current pressure boundary screening criteria. This should remain true if Line E dynamic loads do not increase.

The axial restraint fittings from Line A that were evaluated for this report satisfied all but the following criteria:

- (1) Secondary limits for ΔOBE and $\Delta SAMS(HOT)$.
- (2) Secondary limits for ΔP and ΔSWR .
- (3) Primary + secondary range limits for ΔP and ΔSWR .

Screening criteria in computer code ASEP show that inelastic analysis will show the present axial restraint fitting design to meet these criteria; thus showing the design acceptable.

A post-processing computer program, capable of evaluating the stresses in the pipe resulting from the combined effects of the restraint load and piping system loads, was developed. It would have been incorporated into the main piping stress analysis program.

The detail design of the axial restraint fitting and the Fabrication Stress Report [25] were completed. The axial restraint fittings were released for project use following approval of the design layout drawings. Fabrication of these components was initiated.

7.1.3.5 Mixing Tee

The IHTS mixing tee combines the sodium flows from the two evaporators and provides a uniform stream to the pump. The design conditions are relatively severe due to large differences in evaporator outlet temperatures that can occur during certain transients and the complex fluid hydraulic behavior at the mixing tee.

The mixing tee design concepts were developed and tested by using the results of a research program on mixing components at Argonne National Laboratory (ANL). Those two concepts included a standard tee and a tee with insert plates to divide the flow into smaller streams to promote mixing within the tee body and thereby reduce the temperature oscillations at the tee wall surfaces. The basic design requirements were established and additional testing requirements were identified to obtain the data required to perform a complete analysis of the two designs. To provide that data, a development test program conducted at

ANL included testing for temperature fluctuations throughout the mixing tee and vibration effects due to flow and mixing.

The experimental data obtained from the development program conducted at ANL was used to convert the CRBRP imbalance transients into fluctuating transients, which were incorporated into the detail fatigue analysis of the mixing tee. A preliminary design review was held [26], and the standard tee (without insert plates) was selected as the inserts presented inspection problems and difficult design problems under certain transient events.

Preliminary analysis performed by General Electric on the 24-inch Main Mixing Tee indicated that the wall thickness of the tee must be increased from 0.5-inch to 1.0-inch in order to accommodate the primary loadings. A detailed second interim stress analysis included all of the primary and secondary loading in addition to the fatigue loading. The results showed that the standard tee should provide a 12-year service life under the most severe fatigue conditions when evaluated to the 1974 Code. To mitigate the problem an ASME Code change was initiated to allow the use of the 1977 Code through Summer 1979 addenda in the fatigue analysis of the mixing tee. This newer Code edition reduces excess conservatism as compared with the 1974 Code, which had been used previously in the tee analysis. This proposal was submitted to the RDT Standard F9-4T Task Force for evaluation and recommendation. The Task Force recommended use of the new Code edition, which will extend the maximum allowable service life of the mixing tee. The proposed code change was approved by the CRBRP Project Office. The mixing tee was design-reviewed and released as part of a main-loop piping spool with the spool drawings and parts lists.

The mixing tee final stress analysis was initiated at Foster Wheeler Energy Applications. This work included the 24-inch, 8-inch and 3-inch mixing tees. This analysis and final stress report was to be completed in March 1984.

7.1.3.6 IHTS Piping Support Loading Under Rapid Drain Conditions

Evaluation of the IHTS piping support arrangements showed that, when the sodium mass is removed from the piping, equilibrium forces from the constant load spring hangers resulted in excessive pipe stresses at a number of locations. For planned maintenance events, where sufficient time is available, an acceptable solution could consider installation of mechanical travel stops, temporary blocking, etc. However, the need to accommodate the relatively rapid (<1 hour) dump of the sodium that can result from plant events such as large SWRs, pipe leaks, etc., led to proposing an engineered modification to the constant support hanger design.

A modified hanger system was devised to eliminate the overload condition. At selected locations the constant hanger was to be replaced by a combination of constant hanger and variable (spring type) hanger, each of which would have a smaller displacement and cost than the replaced hanger. This modified hanger system was analyzed and found to significantly reduce the empty weight loading on piping and components. This modified hanger system was proposed to other Project participants by General Electric in September 1983 [27].

In October 1983 Westinghouse proposed consideration of two other alternatives to solve the empty weight problem [28]:

- A. Use travel stops and pins in lieu of additional variable support hangers.
- B. Use rigid restraints in lieu of constant support hangers at selected locations.

These alternate solutions had been analyzed and evaluated by General Electric prior to submittal of the proposed modified hanger system. Response to the Westinghouse comments, including further justification of the proposed system, was underway at the time of Project termination in November 1983.

7.1.3.7 IHX Intermediate Side Vent

During the course of the IHTS system design, several proposals for venting the IHTS side high points of the IHX were considered. A venting subsystem was required to facilitate IHTS fill and drain; prevent gas accumulation which might inhibit pony motor flow or natural circulation flow and provide means for isolating the IHTS from PHTS in the event of an IHX tube leak.

The final venting subsystem selected [29] consisted of the following elements:

- o A 2-inch sodium piping continuous flow running vent connected between the IHX intermediate side outlet high point vent nozzle and the main loop hot leg piping.
- o A 2-inch sodium piping interconnecting line, with an orifice and reach rod operated (active) manual valve, between the running vent line and IHTS cold leg main loop piping.
- o A freeze vent (non-active) connected to the IHX IHTS inlet piping for fill and drain.
- o Portable argon gas bottle connection duplex rupture disk piping assembly connected to both the continuous hot leg running vent line and IHTS cold leg piping at IHX inlet to provide for introduction of gas to effect a siphon break, or provide for isolation of the IHTS from the PHTS at the IHX.

Trace heating and PM flowmeter on the running vent would not be safety-related. The flowmeter would include a low flow alarm to warn the operator of loss or reduction of vent line flow.

The duplex rupture disk assembly would be located as close to the connecting flowing sodium line as possible to minimize gas pockets or stagnant sodium. The duplex rupture disk assembly would contain a sodium detector between the disks. The assembly would be tested in both directions to verify that it will take normal IHTS system pressure/temperature and then burst in the reverse direction on application of the bottled gas supply.

The IHX vent subsystem was being implemented into the IHTS system design and supporting systems designs at the time of CRBRP termination. Design and manufacture of the duplex rupture disk assemblies would have been initiated in 1984 following establishment of final interface piping loads.

7.1.3.8 Superheater Rupture Disk Margin During Safe Shutdown Earthquake

The upstream membranes in the main IHTS duplex rupture disk assemblies would be subjected to degradation effects of creep, corrosion, stress relief and fatigue during their service lives. The design service lives for the superheater and evaporator rupture disks are 5 years and 10 years respectively [30]. The above degradation effects all tend to decrease the rupture disk buckling pressure. It was necessary to determine that adequate margin existed above the burst pressure to avoid common mode failure of the rupture disks due to seismic loading during the Safe Shutdown Earthquake (SSE) at the end of the superheater rupture disk life.

The above effects occur principally with the superheater rupture disks operating at the elevated temperatures; only minor effects occur with the evaporator disks operating at low temperatures. The creep, corrosion and fatigue effects were determined analytically and the stress relief effect was determined experimentally [31]. The fatigue effects were found to be negligible. The rupture disk rating was found to decrease by about 2 psid due to creep, by 11 psid due to corrosion and 21 psid due to stress relief for an overall degradation effect of 34 psid over the 5 year superheater rupture disk life.

Conservative analysis [31] showed that the minimum end of life margin for the main IHTS superheater rupture disk under SSE conditions would be 28 psid. This analysis evaluated IHTS loop 3 which contain the longest piping runs and therefore the highest seismic loads. Additional margin would be provided for rupture disks in the other two IHTS loops which contain the shorter piping runs. At time of CRBRP termination, the SSE imposed loads on the rupture disks were being reevaluated with the current IHTS piping arrangement; additional margin was expected because additional pipe supports had been added subsequent to the original loop 3 seismic analysis performed in 1978.

7.1.3.9 "Hot Tee" Evaluation

In the baselined design of the SGS, flow detectors were provided downstream of the main IHTS rupture disks assemblies. These detectors were part of the Plant Safety System and would have initiated automatic plant shutdown if the disk(s) burst due to a large SWR in one of the evaporators or superheaters. Automatic plant shutdown would have included trip of the IHTS sodium pumps and SGS recirculation water pumps and isolation and blowdown of the evaporators and superheater in the affected loop. In the baselined design, the SWRPRS bus, the SGS safety relief valve operators and water dump valve operators were non-1E because they were not involved with decay heat removal. In addition, only a single division of power was provided to each SGS loop.

The OPDD-10, Section 7, Criteria 26 [32] requires consideration of loss of off-site power and an additional single failure (failure of division of power) in conjunction with a design basis event such as the SWR DBL. Under these guidelines it had to be assumed, for the SWR DBL with loss of a single division of power to the affected SGS loop, that the SGS would not be depressurized through the power relief valves nor would the water be dumped through the water dump valves. Thus the SGS water/steam side would only be depressurized through the failed tubes of the affected steam generator module (SGM). The water/steam would continue to flow into the shell of the affected SGM and then into the connecting sodium line. A stationary SWR reaction interface potentially could exist at the tee, connecting the affected SGM, IHTS pipe and rupture disk.

Another potential problem existed in that multiple tube failures could occur in the affected SGM over this long SGS depressurization period, which could result in overpressurization of the IHTS. During a SWR DBL most of the sodium in the affected SGM would be ejected, leaving behind a heel of sodium in the lower region of the SGM. Steam from the DBL would continue to flow over this stagnant heel; the accompanying sodium/water reaction would raise the temperature of this region to about 1700-2200°F resulting in multiple tube failures. Because of the finite flow capacity of the SWRPRS relief lines, the design pressure (425 psig under this emergency condition) of the IHTS, including IHX tubes, could be exceeded by the steam pressure produced from the multiple tube failures.

To correct these conditions, General Electric proposed to make the SWRPRS Bus Class 1E; provide two divisions of power to each loop; make safety relief valves and water dump valves Class 1E and add air accumulators for these valves [33]. By proper assignment of SGS valves to either one of two divisions of power supplied to each loop, the evaporators and superheater in each loop could be isolated and depressurized rapidly (~ 10 seconds for superheaters and 30 seconds for evaporators) under all conditions. These proposed revisions would have ensured that excessive temperatures of the steam generator module tubes and IHTS piping would not occur during a SWR DBL event.

Extensive analysis, along with stress rupture testing of 2-1/4 Cr-1Mo tubing at elevated temperature were performed during 1982-1983 to quantitatively demonstrate the need for the above proposed system changes. The analysis included evaluating water side blowdown times for all configurations and failure modes, calculation of sodium side dynamics (pressure flow) as a function of time, calculation of tube wall temperatures for comparison to tube material properties and selection of the optimum configuration with respect of Division of Power assignment to the SGS valves [34].

This analysis work [34] showed that with the present baseline system (i.e., non-1E SWRPRS Bus, non-1E SGS valves and single division of power to an SGS loop) a SWR DBL in the superheater would result in superheater tube temperatures in the stagnant pool region in the range of 2000-1800°F for a period of about 35 seconds. Stress-rupture test data indicated that the superheater tubes could rupture in this time period. If all superheater tubes were to rupture, the steam introduced into the IHTS would produce an IHTS pressure of about 1050 psig (in excess of the 450 psig emergency pressure design condition for the IHTS). High thermal loading on the "Hot Tee" and high mechanical loading on the SWRPRS piping from the prolonged high temperatures would result from the delayed SGS water side depressurization.

A DBL in the evaporator with the present baselined system would also produce elevated superheater tube temperatures; however, these temperatures would not be high enough to rupture the superheater tubes based on the stress rupture test data. However, the evaporator shell side temperature of 1700°F would exist for about 15 minutes during which time caustic corrosion would penetrate the evaporator tubes; failure of numerous evaporator tubes could be expected.

Prior to Project termination, the CRBRP-PO approved the concept of making the SWRPRS Bus 1E and authorized the relief valves and dump valves to be qualified 1E by the manufacturers. Further analyses of the consequences of the SWR DBL were planned using tubes failing at a rate based on empirical data from the ETEC SWR tests to justify the proposed system changes described above.

7.1.4 Final System Status

7.1.4.1 Status of Design/Analysis

SDD 51B, Sections 1 and 2 were baselined and updated as the system design evolved [1] [3]. The three remaining SDD sections covering system limitations, set points and precautions (Section 3); operations (Section 4); casualty events and recovery procedures (Section 5) and maintenance and in-service inspection (Section 6) were updated and transmitted to Westinghouse for approval to baseline [8] [9] [10] [11]. Pre-operation Test Specifications and Construction Test Requirements (CTRs) had been prepared. The CTRs were revised in early November 1983 [35] to reflect FWSDR comments.

Significant reliability evaluation work had been performed in support of the IHTS design. The final Reliability Design Support Document (RDSD) for the IHTS was provided in March 1983 [36]. An independent design review of the IHTS system design was performed and the design review report was issued in November 1983 [37].

IHTS design personnel participated in the Sodium Fire Mitigation Task Force to evaluate methods of mitigating the effects of a sodium fire in the SGB [38]. In addition, IHTS design personnel provided pre-test analyses [39] [40] for the sodium fire test program.

Pipe support ICDs and isometric drawings of the IHTS piping were released for Project use. Following the main loop piping final design review [16], the Fabrication Stress Report was issued and pipe spool drawings and ordering data for pipe supports were released for Project use. Final stress analyses of the main loop and auxiliary IHTS piping was underway at the time of CRBRP termination. A modified hanger system to reduce the dead-weight loading in the

piping system under empty conditions was receiving project review at time of CRBRP termination [27].

The sodium dump tanks had been designed, manufactured and delivered to the storage site near CRBRP. The sodium expansion tanks were designed and subjected to a final design review in January 1983 [41]. In February 1983, the expansion tanks were released for fabrication, which was underway at time of Project termination. The status of engineering work on the sodium pumps is covered in Section 7.3.

7.1.4.2 Outstanding Activities to Complete Design

The following engineering change proposals (ECP) covering IHTS design or in-service inspection were being reviewed by Project participants at the time of Project termination.

- A. G1284 - ECP baselines sodium purity and cleanup requirements.
- B. G1027 - ECP baselines sodium pump combustibles.
- C. G1403 - ECP requires IHTS sodium dump valve operator as "failed closed". This resolves one of the FWSDR concerns.
- D. G1498 - ECP establishes in-service inspection (ISI) requirements for TWJ and identifies equipment required to perform ISI at 400°F during refueling.
- E. G1436 - ECP provides overpressure protection from control valve/regulator failures in Argon cover gas system resulting from seismic events.
- F. G1387 - ECP revises main loop piping drawings to incorporate some thicker wall fittings as determined to be required from stress analyses.
- G. G1451 - ECP revises main loop piping support arrangement from constant support system to a combination of constant and variable hangers. This hanger change is needed to resolve high upward loadings resulting from dump of sodium at elevated temperatures.

The principal remaining work in the IHTS systems and piping design covered the following areas:

- A. Completing the final thermal/hydraulics analysis report.
- B. Preparation of the IHTS overpressure protection report.
- C. Completing final stress reports for main loop piping, auxiliary piping, piping clamps, transition joints, venturi flowmeters, flow restrictors, mixing tees and axial restraint forgings.
- D. Completing verification of computer codes used in the final piping and system designs.
- E. Issuing Installation and Operating Manuals for large and small pipe clamps.
- F. Performing off-normal evaluation of IHTS piping for failed (locked) snubbers.
- G. Performing off-normal evaluation of IHTS piping for hot cell (fire) conditions.
- H. Completing detailed design of the IHX IHTS side vent system including design of duplex rupture disks.
- I. Preparing maintenance platform ICD

7.1.5 References

- 1. "System Design Description, Intermediate Heat Transport System, System 51B," General Electric Company Specification 23A2173, Revision 74, June 29, 1983.
- 2. "Clinch River Breeder Reactor Plant Reference Design Report," Westinghouse Electric Corporation, June 1974.
- 3. "System Design Description, Intermediate Heat Transport System, System 51B," General Electric Company Specification 23A2173, Revision 0, December 1976.
- 4. Letter XL-590-20401, F.P. Crimi (General Electric Company) to W.J. Purcell (Westinghouse Electric Corporation), "CRBRP; System 51B Intermediate Heat Transport System Final Working System Design Review (FWSDR) Package Submittal," September 20, 1982.

5. Letter CR-747:FN:82-420, C.H. Fox, Jr. (PO) to W.J. Purcell (W-OR), "CRBRP: Final Working System Design Review (FWSDR) on the Intermediate Heat Transport System (IHTS)," December 16, 1982.
6. Letter XL-590-830400, F.P. Crimi (GE) to W.J. Purcell (W-OR), "CRBRP; System 51B; Transmittal of Draft Section 3 of SDD-51B for Approval to Baseline," July 15, 1983.
7. Letter XL-590-830445, F.P. Crimi (GE) to W.J. Purcell (W-OR), "CRBRP; System 51B; Transmittal of Draft Section 4 of SDD-51B for Approval to Baseline," August 3, 1983.
8. Letter XL-590-830527, F.P. Crimi (GE) to W.J. Purcell (W-OR), "CRBRP; System 51B; Transmittal of Draft Sections 5 and 6 of SDD-51B for Approval to Baseline," September 29, 1983.
9. Letter XL-590-830561, F.P. Crimi (GE) to W.J. Purcell (W-OR), "CRBRP; System 51B; Transmittal of Draft Appendix I of SDD-51B for Approval to Baseline," October 21, 1983.
10. CRBRP Equipment Specification, "CRBRP Heat Transport System Piping Design Specification," E953089, Westinghouse Electric Corporation, December 1975.
11. Engineering Change Proposal (ECP) L00-001 "Seismic Response Spectra," Westinghouse Electric Corporation, Approved March 20, 1976.
12. ECP L10-137, "Redefinition of the Design Basis Sodium/Water Reaction Event," Westinghouse Electric Corporation, Approved July 14, 1980.
13. WARD D-0037, "Seismic Design Criteria for the Clinch River E breeder Reactor Plant," Westinghouse Electric Corporation, May 25, 1977.
14. ECP G0613, "System 51B Main Loop Piping and Support Arrangement," General Electric Company, Approved December 12, 1980.
15. ECP G0555, Revision 4, "System 51B Dump, Drain and Fill Piping Update," General Electric Company, Approved May 29, 1981.

16. Letter XL-590-10325, F.P. Crimi (GE) to W.J. Purcell (W-OR), "CRBRP; IHTS Main Loop Piping Design Review," August 14, 1981.
17. Letter XL-590-20454, F.P. Crimi (GE) to W.J. Purcell (W-OR), "CRBRP; IHTS Auxiliary Piping Design Review Report and Design Release," October 27, 1982.
18. J.O. Sane, et al, "Selection of the Sodium/Water Reaction Design Basis Event for the Clinch River Breeder Reactor Plant and Analyses of the Effects of the Event of the Intermediate Heat Transport System," CRBRP-GEFR-00593, August 1982 (Revised November 1984).
19. D.E. Knittle and J.C. Amos, "LLTR Series II Test Program Large Leak Tests Final Report," GEFR-00609, October 1982.
20. Letter XL-590-10394, F.P. Crimi (GE) to W.J. Purcell (W-OR), "CRBRP Transition Joint Design Release," September 30, 1981.
21. P.J. Ring, et al, "Transition Joint Development Program - Final Report," GEFR-00645, December 1982.
22. Letter 82ESG-2081, E.F. Weisner (Rockwell International) to W.J. Purcell (W-OR), "Marinite P Insulation Materials Properties Report," March 31, 1983.
23. S.V. Jeffries, et al, "IHTS Pipe Clamp Fabrication Stress Report," GEFR-00677 (Draft), July 21, 1983.
24. Letter YL-595-830641, J.P. Verkamp/E.F. Petersen (GE) to F.P. Crimi (GE), "Design Review Report IHTS Large Pipe Clamps," October 11, 1983.
25. Letter YL-606-20073, S.V. Jeffries (GE) to A.W. Dalcher (GE), "IHTS Axial Restraint Fitting Fabrication Stress Report," May 28, 1982.
26. Letter XL-590-80361, D.H. Krueger (GE) to E. Rodwell (W-OR), "Mixing Tee Design Review Report," October 27, 1978.

27. Letter XL-595-830555, P.E. Bohaboy (GE) to W.W. Dewald (W-OR), "CRBRP; ECP G1451, Rev. D1, 51B (IHTS) Piping Support Update," September 6, 1983.
28. Letter WG830064, C.G. Reese (W-OR) to G.G. Glenn (GE), "CRBRP; ECP G1451, Rev. D1, 51B (IHTS) Piping Support Update," October 6, 1983.
29. Letter LG820538, G.G. Ritter (W-OR) to G.G. Glenn (GE) and R.J. Beely (Rockwell International), "CRBRP; Approval of ECP G1208, Revision 2, System 51B IHX Intermediate Side Vent," December 8, 1982.
30. System Design Description for CRBRP Steam Generator System (System 53), General Electric Specification 22A3525, Revision 101, August 2, 1983.
31. D.F. Casey and J.C. Whipple, "Design Evaluation of CRBRP Rupture Disks," CRBRP-GEFR-00714, October 1984.
32. OPDD-10, "Clinch River Breeder Reactor Plant Overall Plant Design Description," Revision 146, Westinghouse Electric Corporation, August 1, 1983.
33. Letter XL-595-20276, P.E. Bohaboy (GE) to W.J. Purcell (W-OR), "Request for Approval of Expedited Change Upgrading System 53 Evaporator Water Dump Valves and Steam Generator Module Power Relief Valves to Class 1E Active Valves," December 28, 1982.
34. Letter LG830452, W.J. Purcell (W-OR) to G.G. Glenn (GE) and P. Brewington, Jr. (Clinch River Breeder Reactor Plant Project), "CRBRP; System 53, Steam Generator Open Technical Issues," October 10, 1983.
35. Letter XL-590-830568, F.P. Crimi (GE) to W.J. Purcell, (W-OR), "CRBRP; System 51B, Intermediate Heat Transport System Construction Test Requirements," November 10, 1983.
36. Letter XL-599-830039, W.W. Phelan (GE) to W.J. Purcell (W-OR) and W.W. Dewald (W-OR), "CRBRP; SHRS Reliability (WA612) - RDSD (Revision 3) for Intermediate Heat Transport System (System 51B), March 30, 1983.

37. Letter XL-590-830577, F.P. Crimi (GE) to W.J. Purcell (W-OR), "CRBRP; Completion of System 51B Milestone No. 51G1M16, System 51B Design Review Report Release," November 4, 1983.
38. "Task Force II Study of CRBRP Sodium Fires," Westinghouse Electric Corporation, March 1983.
39. Letter XL-590-20109, F.P. Crimi (GE) to W.J. Purcell (W-OR) and R.J. Beeley (Rockwell International), "Pre-Test Analysis of DRS.03 Second Air Benchmark Spray Fire Test," April 6, 1982.
40. Letter XL-590-20303, F.P. Crimi (GE) to W.J. Purcell (W-OR) and R.J. Beeley (Rockwell International), "Pre-Test Prediction of DRS 26.03 First Large Scale Sodium Fire Test Using Modified Spray-3B Code," July 20, 1982.
41. Letter YL-595-830163, J.P. Verkamp/E.F. Petersen (GE) to F.P. Crimi (GE), "Design Review Report, IHTS Sodium Expansion Tank," March 17, 1983.

7.1.6 Major ECPs Affecting IHTS Design*

The following ECPs were the major ECPs which affected the IHTS system and piping baselined designs:

G0452	System 51B Auxiliary Piping and Supports
G0547	System 51B Superheater and Evaporator Vent Piping and Supports
G0555	System 51B Dump, Drain and Fill Piping Update
G0613	System 51B Main Loop Piping and Support Arrangement
G0689	System 51B Expansion Tank Return Line Update
G0971	System 51B Transition Joints Spool Fabrication, Hold Removal
G1093	Instrument and Leak Detection Pipe Transition Joints
G1123	Increase Transition Joint Size
G1208	System 51B IHX Intermediate Side Vent
G1451	System 51B (IHTS) - Lines A & E Piping Support Update (Under review at time of CRBRP termination)
L10-001	Seismic Response Spectra
L10-011	CRBRP Load Combination
L10-137	Redefinition of the Design Basis Sodium/Water Reaction Event

- * ECPs exclude those related to the sodium pumps. These are described in Section 7.3. All ECPs were approved and implemented in Project documentation except as noted.

7.2 Leak Jacket Design

7.2.1 Introduction

A sodium leak in the System 51B Intermediate Heat Transport System (IHTS) piping would result in a violent sodium fire in the air atmosphere of the CRBRP Steam Generator Building (SGB). To reduce the extreme temperatures and pressures associated with a spray type sodium fire, the IHTS insulation system was redesigned with a leak jacket which encapsulated the pipe in order to mitigate the sodium jet potential from this postulated accident event. The key features of the leak jacket concept included:

- A. An inner jacket designed to accept the sodium jet impingement loads and contain the sodium leakage with minimal leakage into the SGB cell environment.
- B. A mechanically sealed interface using Z-rings between the pipe clamp and the insulation inner jacketing to minimize sodium leakage from the interface into the cell environment and to channel any leakage from the pipe surface covered by the pipe clamp into the insulation annulus.
- C. An annulus drain connection sized to channel leaking sodium into a liquid stream with minimum build-up of back pressure in the insulation annulus.

The majority of the sodium leakage would be directed through a dedicated drain system to catch pans and the fire suppression deck system in the SGB. The balance of the sodium leakage, which had the potential for spray formation, had a direct bearing on the SGB cell pressure and temperature transients. Due to the uncertainties relating to the assumptions regarding potential leak path areas, flow conditions in the pipe/inner jacket annulus, etc., experimental verification was considered important. Therefore, to demonstrate the feasibility and effectiveness of the proposed leak jacket design concept, full scale tests of a typical CRBRP 24 inch elbow/pipe section were performed.

The following sections discuss the background of the problem, rationale used for establishing the design basis leak, descriptions of leak jacket concepts and testing, and evaluations and additional consequences of the final CRBRP design.

7.2.2 Background

Based on data obtained from the April 1982 Large Scale Sodium Fires Test (DRS 26.03 [2.5-5]), the predicted spray-burning rates used in the plant sodium spray fire analyses were found to be nonconservative as discussed in Section 2.3.1 of this report. The Spray Pool Combustion Analysis (SPCA) code was subsequently modified to reflect the observed test phenomena and hence provide a more accurate representation of accident conditions in the CRBRP SGB [1]. Calculations for SGB sodium leak accidents, using the modified code, resulted in cell temperatures, pressures, and aerosol releases that exceeded the current design criteria.

In January 1983, two task forces were formed by the CRBRP Project to:

- A. Assess the reliability of the SPCA code predictions (Task Force I),
- B. Make recommendations regarding potential design fixes to mitigate the leak accidents and to make recommendations regarding future actions (Task Force II).

Based upon the findings of Task Forces I and II (documented in References 2 and 3 respectively), the following actions were recommended and subsequently approved by the CRBRP Project [4]:

- A. The piping insulation design for IHTS piping (4 inches and above in diameter) be modified to remain intact in the event of a design basis sodium leak in order to reduce the potential for sodium spray formation.
- B. Tests be performed using water to verify the performance of the modified piping insulation (leak jacket) design.

- C. A test be performed with sodium to demonstrate the effectiveness of the large scale catch pan fire suppression deck test article (DRS 26.03) under realistic thermal loadings (see Section 2.3.1 of this report).
- D. The SPCA code be further modified to enhance its capabilities.
- E. A cell-by-cell analysis be performed to identify additional design fixes as required.

Based upon the above, the IHTS piping system insulation design was modified. The water testing of the modified insulation concept on a 24 inch elbow/pipe section was successfully completed in July 1983 by demonstrating the effectiveness of the design to mitigate sodium spray to a minimum. The large scale sodium fire test was also completed in May 1983 and verified the structural integrity and fire suppression capability of the catch pan-fire suppression deck design. Modifications to the SPCA code were completed in August 1983, and a cell-by-cell analysis of the consequences of a sodium and NaK leak accident was completed in September 1983. Additional changes to the baseline design were identified as necessary to reduce the environmental impact of the accidents to acceptable levels.

An Engineering Change Proposal (ECP A26036) was prepared to baseline the requirements for the modified piping insulation design and the additional requirements derived from the cell-by-cell analysis [5,6]. In addition, the ECP included requirements to reduce the aerosol releases to the atmosphere to a level that would eliminate the need for equipment qualification for the "non-severe" sodium aerosol environment.

Due to Project termination in November 1983, ECP A26036 was never approved by the CRBRP Project.

7.2.3 Design Basis Leak Rationale

The Fast Flux Test Facility (FFTF) demonstrated that the probability of a pipe break in the primary heat transport system (PHTS) piping was so low that it need not even be considered as a Design Basis Accident (DBA) [7]. The lines of defense, built about detectable leak-before-break rationale, were logical and formidable. This same approach was used for the CRBRP PHTS and IHTS in-containment and also extended to the IHTS piping out-of-containment.

The leak-before-break concept was accepted by the Nuclear Regulatory Commission (NRC) in the CRBRP Safety Evaluation Report (SER) [8] for both the PHTS and IHTS. The Design Basis Leak (DBL) selected for the IHTS corresponded to a leak recommended in the NRC Branch Technical Position (BTP) Mechanical Engineering Branch (MEB) 3-1 [9]. This position pertains to piping for moderate energy fluid systems (MEFS) and states:

Section II, Moderate Energy Fluid System Piping; 3(c)2, Through Wall Leakage Cracks:

Fluid flow from a crack should be based on a circular opening of area equal to that of a rectangle $1/2$ pipe diameter in length and $1/2$ pipe wall thickness in width.

BTP MEB 3-1 provides the criteria for establishing the piping locations having relatively higher potential for failure. Piping failures generally occur at high stress and fatigue locations, such as terminal ends, elbows, branch connections, reducers, tees, etc. In accordance with MEB 3-1, breaks and leakage need not be postulated in those portions of a piping system that meet certain stress limits. However, leakage must be postulated at the locations in each piping loop that exceed the prescribed stress limits.

The CRBRP sodium and NaK systems outside containment can be classified as moderate energy fluid systems within the intent of the definitions in MEB 3-1. Since the IHTS piping operates at elevated temperature, the MEB 3-1 criteria were modified to account for the ASME B&PV Code elevated temperature criteria in Code Case N-253.

The establishment of a design basis sodium leak was based on the following rationale for CRBRP:

- A. For piping with stresses exceeding the MEB 3-1 limit and protected by the proposed insulation leak jacket, the DBL was taken as the MEFS size leak and the maximum sodium jet cannot exceed 50 gallons per minute (gpm) from a MEFS leak in the IHTS piping.
- B. For piping with stresses below the MEB 3-1 limit, the piping was not protected by the insulation leak jackets. It was shown by Linear Elastic Fracture Mechanics (LEFM) that existing defects will not propagate based on the total plant duty cycle loading [10]. This includes both the IHTS piping and auxiliary sodium lines. However, it was still postulated that a 50 gpm sodium jet existed for this piping.

The 50 gpm sodium jet corresponded to a maximum limit of 5 percent of the MEFS leak established by MEB 3-1 for the 24 inch CRBRP IHTS piping. This limit would result in a peak accident gas temperature of 500°F and peak pressure of less than 1 psig in the CRBRP SGB during the DBL event. These accident conditions were considered to be within the acceptable limits for the SGB.

7.2.3.1 Criteria for MEFS Leak Locations

The CRBRP IHTS piping was designated American Nuclear Standard (ANS) Safety Class 2 in accordance with the safety requirements and functions of the IHTS piping system. In conformance with Regulatory Guide 1.26 [11], the IHTS piping must be designated ASME B&PV Code Class 2 as a minimum. To assure added quality in the design and fabrication, the CRBRP IHTS piping system was designed, constructed, and stamped to ASME B&PV Code Class 1 requirements.

MEB 3-1 provides different criteria for postulating piping leakage cracks depending on whether the piping is low temperature ASME Class 1 or ASME Class 2 or 3. Since the IHTS piping was designated ANS Safety Class 2, the criteria for ASME Class 2 and 3 were used to satisfy the intent of the NRC Standard Review Plan (SRP) Section 3.6.2 [12].

For moderate energy fluid systems, MEB 3-1 provides the stress limits for when leakage cracks must be postulated. These criteria are based on the low temperature ASME B&PV Code (NC-3600) [13] piping rules and are as described below and in Reference 14:

If the maximum stress ranges as calculated by the sum of Equations (9) and (10) in Paragraph NC-3652 of the ASME B&PV Code are greater than $0.4 (1.2 S_h + S_A)$ considering those loads and events thereof for which Level A (Normal) and Level B (Upset) stress limits have been specified in the piping systems design specification (i.e., sustained loads, occasional loads, and thermal expansion) including an Operating Basis Earthquake (OBE) event, leakage cracks shall be postulated. The S_h and S_A are allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively. In equation form, these criteria can be expressed as follows:

$$S_{OL} = B_1 \left(\frac{P_{\max} D_o}{2t} \right) + B_2 \left(\frac{M_A + M_B}{Z} \right) \quad \text{Equation (9)}$$

$$S_E = i \left(\frac{M_C}{Z} \right) \quad \text{Equation (10)}$$

where

- B_1, B_2 = primary stress indices
- $P_{\max}$ = maximum pressure, psig
- D_o = outside diameter of pipe, in.
- t = nominal wall thickness, in.
- Z = section modulus, in.³
- M_A, M_B, M_C = bending moments, in.-lb
- i = stress intensification factor

Therefore, the postulated leakage criteria is defined as:

$$S_{OL} + S_E > 0.4 (1.2 S_h + S_A)$$

Since the IHTS piping operates at elevated temperatures, the MEB 3-1 criteria were modified to account for the ASME B&PV Code elevated temperature criteria given in Code Case (CC) N-253 [15]. Code Case N-253 modifies the low temperature piping rules as follows:

- A. Extends the check of Equation (10) to Service Level C.
- B. The stress allowable reduction factor (due to fatigue), f , becomes a function of material, temperature, and thermal transient stresses (as measured by ΔT_1 and ΔT_2) on the component. (See Appendix B of CC N-253.)
- C. There is an additional check on thermal expansion stresses (Eq. 10(b) in CC N-253).

Based on the above and MEB 3-1, the following criteria were developed to locate leakage cracks in the IHTS piping:

$$S_{OL} + S_E > 0.4 (1.2 S_h + S_A)$$

and

$$S_{OL} + \frac{\Delta T_1}{\Delta T_e} S_E + \frac{E \alpha \Delta T_1}{4(1-\nu)} > 0.4 [1.2 S_h + (0.75 S_{yc} + 0.25 S_h)]$$

where

$$S_A = f (1.2 S_C + 0.25 S_h)$$

$$f = \text{given by Table B-1 of CC N-253}$$

$$S_{OL} = \text{given by Equation (9) in NC-3652}$$

S_E = given by Equation (10) in NC-3652
 ΔT_i , ΔT_e , etc., as described in CC N-253
 E = modulus of elasticity, psi
 α = coefficient of thermal expansion, in./in./°F
 ν = Poisson's ratio

These criteria were applied on a load cycle-by-load cycle evaluation for all CRBRP Level A, B, and C events.

The considerations and criteria described above were used to identify all piping elbows which required leak jacket protection in the CRBRP IHTS [14].

7.2.3.2 Leakage from IHTS Piping Excluded by MEB 3-1 Requirements

As discussed in Section 7.2.3.1, only pipe locations which exceeded certain structural limits were evaluated for the MEFS leak. The IHTS sodium piping not required to be evaluated for the MEFS leak was assessed by a mechanistic approach, primarily to ensure that even a large defect will not grow to a through-wall crack. The intent was to establish a more prudent basis for evaluation than categorically eliminating a leak location. The approach taken in this evaluation consisted of the several sequenced steps consistent with the approach taken in the CRBRP primary and intermediate system piping integrity report [16] (within containment). First, the largest expected residual defect in the piping after the final inspection was established. This defect was then assessed in terms of crack growth based on the plant duty cycle, using LEFM. The analysis demonstrated that through-wall cracking based on the piping loads from the plant duty cycle, from a large pre-existing defect, was not credible. The assumption of a 50 gpm sodium jet as a design basis leak causing sodium spray was considered to be very conservative at these locations.

7.2.3.3 Leakage from Small Diameter IHTS Sodium Piping

The CRBRP small sodium piping was evaluated for leaks based on the same requirements as the large diameter IHTS piping. Detailed assessments with respect to the requirements of MEB 3-1 are discussed in Reference 14.

Calculations to estimate the leakage from a hypothetical crack in the most highly stressed elbow in the IHTS small diameter piping indicated a leakage rate of less than 1 gpm. Alternatively, a very long crack (approaching 17 inches) was required to produce the 50 gpm leak rate. Therefore, the 50 gpm leakage rate clearly envelopes the leakage from any through-wall crack in the small diameter sodium piping which satisfies the MEB 3-1 requirements.

7.2.4 Leak Jacket Design Requirements and Features

Four conceptual design concepts were initially investigated to mitigate the sodium fire effects in the CRBRP SGB in the event of a MEFS DBL.

Leakage from the pipe was to be controlled so that a jet stream did not impinge on other objects which could create a sodium rainfall in the building. Sodium leakage was minimized by either containing most of the spill volume within the insulation system (presuming the leak rate time history is minimized with active mitigation) or by channeling an unmitigated leak to acceptable catch pans or drain pipes. The concept evaluations, which included estimated costs for design, fabrication, and associated verification test programs, are discussed in Reference 17.

The insulation system design modifications considered the IHTS hot leg piping within cells 244, 245 and 246 above the 806 foot elevation in the CRBRP steam generator building. Specifically, this included the 24 inch OD piping going to the Superheaters and the 18 inch OD piping leading to the Evaporators. The design concepts were applicable to both pipe sizes, but the emphasis on scoping type structural and hydraulic analysis was on the 24 inch OD pipe size for which the MEFS leak rate time histories were available. The adaptability of the IHTS piping insulation design concepts to System 51B small diameter Steam Generator vent piping are also discussed in Reference 17.

7.2.4.1 Initial IHTS Insulation Design Description

The initial design of the IHTS insulation system had at least two major areas of potentially unacceptable leakage by a sodium jet stream or ring spray. The largest region open to a jet stream from an MEFS leak was where the snubbers and pipe hangers attach to the axial restraint fittings on the piping. There was no jacketing to contain a leak close to the axial restraint fittings. The other region was near pipe clamps, where the inner jacketing comes close to but does not touch the Marinite rings under the pipe clamp. Any leak near the pipe clamp could be contained in the inner jacket to pipe annulus between the convection seal and the pipe clamp, thus forcing it to spray from the circumferential annulus between the jacket and pipe clamp. Also, seam leakage at the jacket overlaps could likewise be significant on the initial design configuration. The inner jacket for the elbow insulation was assembled from many pieces which were riveted and banded. Furthermore, the elbow insulation did not have a stainless steel outer jacket. The outer insulation cover on elbow insulation was fiberglass cloth. This same cloth was also used to cover insulation between the inner and outer jackets near the pipe clamps. The initial designs typically used 0.0355 inch thick stainless steel inner jacket material and 0.015 inch thick stainless steel outer jacketing, neither of which was designed to resist the MEFS jet impingement force. Therefore, the initial insulation design was not suitable for mitigation of sodium leaks.

7.2.4.2 Leak Jacket Concepts

As stated previously, four different conceptual designs were considered for sodium leak mitigation. One common feature to all was the method in which the jet stream was mitigated at the axial restraint fittings. This was accomplished by using an inner jacket cover plate that could be seal welded to the lugs on the axial restraint fittings. The cover plate in no way interfered with snubber motion requirements, and it sealed off the large area where sodium otherwise could freely flow. Another common feature was that the pipe clamps were not "housed" in a container of insulation jacketing. One strong reason for not wanting to house the pipe clamps was the potential for overheating the clamps, since they would no longer experience the free convection cooling that otherwise would exist.

The first design concept used the inner jacket to react the jet impingement force and the inner annulus as the flow path to ducting which penetrated through the insulation to sodium drain pipes. The second concept permitted flow to escape the inner annulus, enter the outer annulus occupied by the insulation, and then discharge from the outer jacket to the sodium drain pipes. The jet impingement force was structurally contained by the outer jacket.

The third concept was similar to the second, except a perforated inner jacket was used to diffuse the jet into a spray which was structurally contained by a much thinner outer jacket. The first three concepts were primarily intended to handle leak rate time histories without active flow mitigation. The fourth concept was a variation of the first concept. The inner jacket was designed for the jet force, but the remaining features were designed to hold the sodium with active flow mitigation within the outer jacket annulus until the system fills itself to the highest elevation of piping where there would be a vent drain in the outer jacket which would then lead to sodium drain pipes.

Detailed comparisons were performed on each of the four proposed conceptual designs and are presented in Reference 17. In summary, the evaluations considered the hydraulic, thermal, structural, and cost (design and fabrication) aspects of the proposed leak jacket designs. Based on the conclusions presented in Reference 17 and recommendations resulting from the CRBRP Sodium Fire Task Force, the first conceptual design concept was selected for further evaluation and testing, and is discussed in the following sections.

7.2.4.3 Leak Jacket Design Details

Figure 7-2 depicts the salient features of CRBRP leak jacket design as applied to the region of piping between the Evaporator inlet nozzle and the first pipe clamp. These design features were equally applicable at insulation system interfaces with all pipe clamps and components in the SGB.

Design details for the leak jacket concept are shown in Figures 7-3, 7-4 and 7-5. All circumferential seams in the inner jacketing were full penetration butt-welded. All longitudinal seams would be provided with a 2.25 inch wide

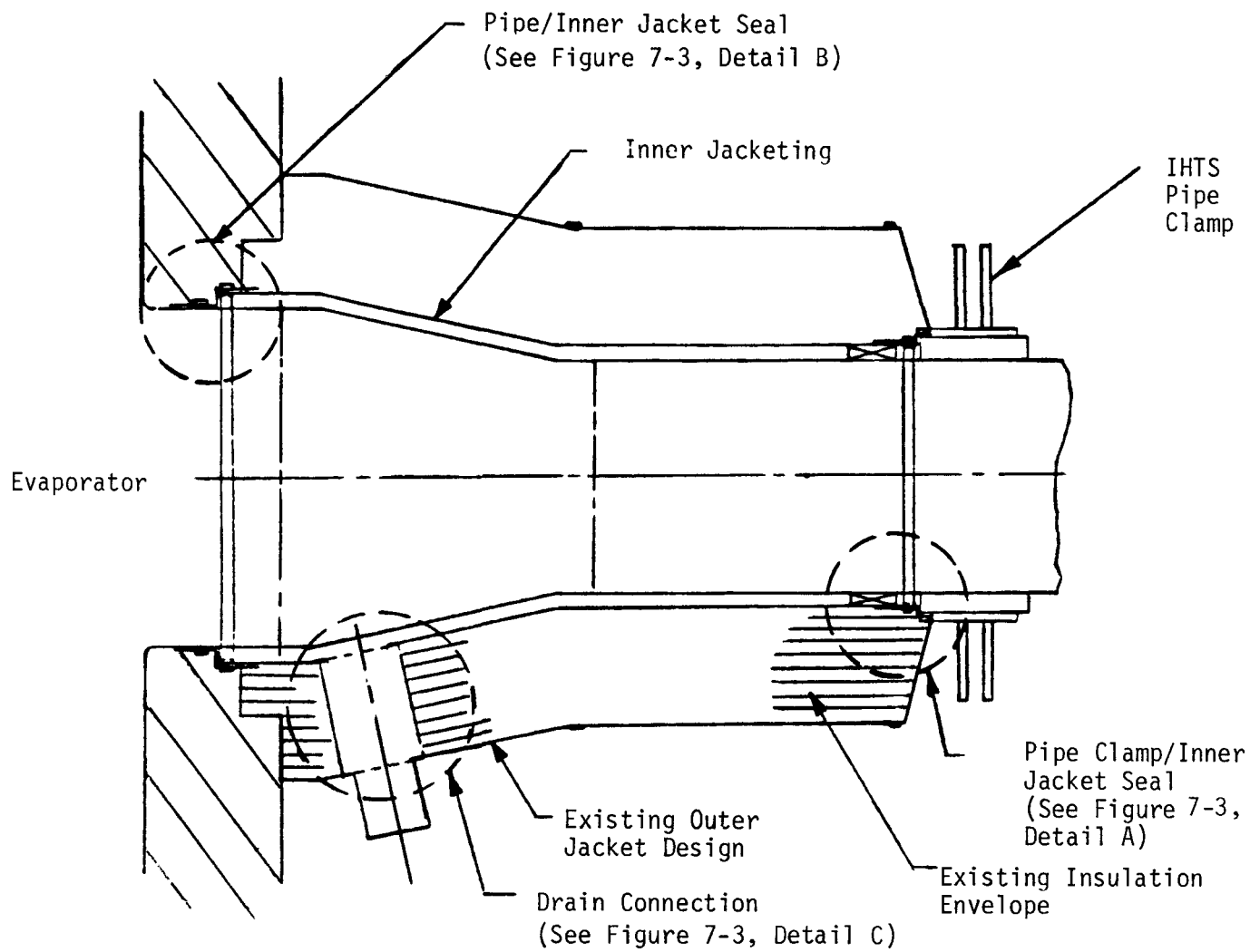


Figure 7-2 IHTS Leak Jacket Design Concept

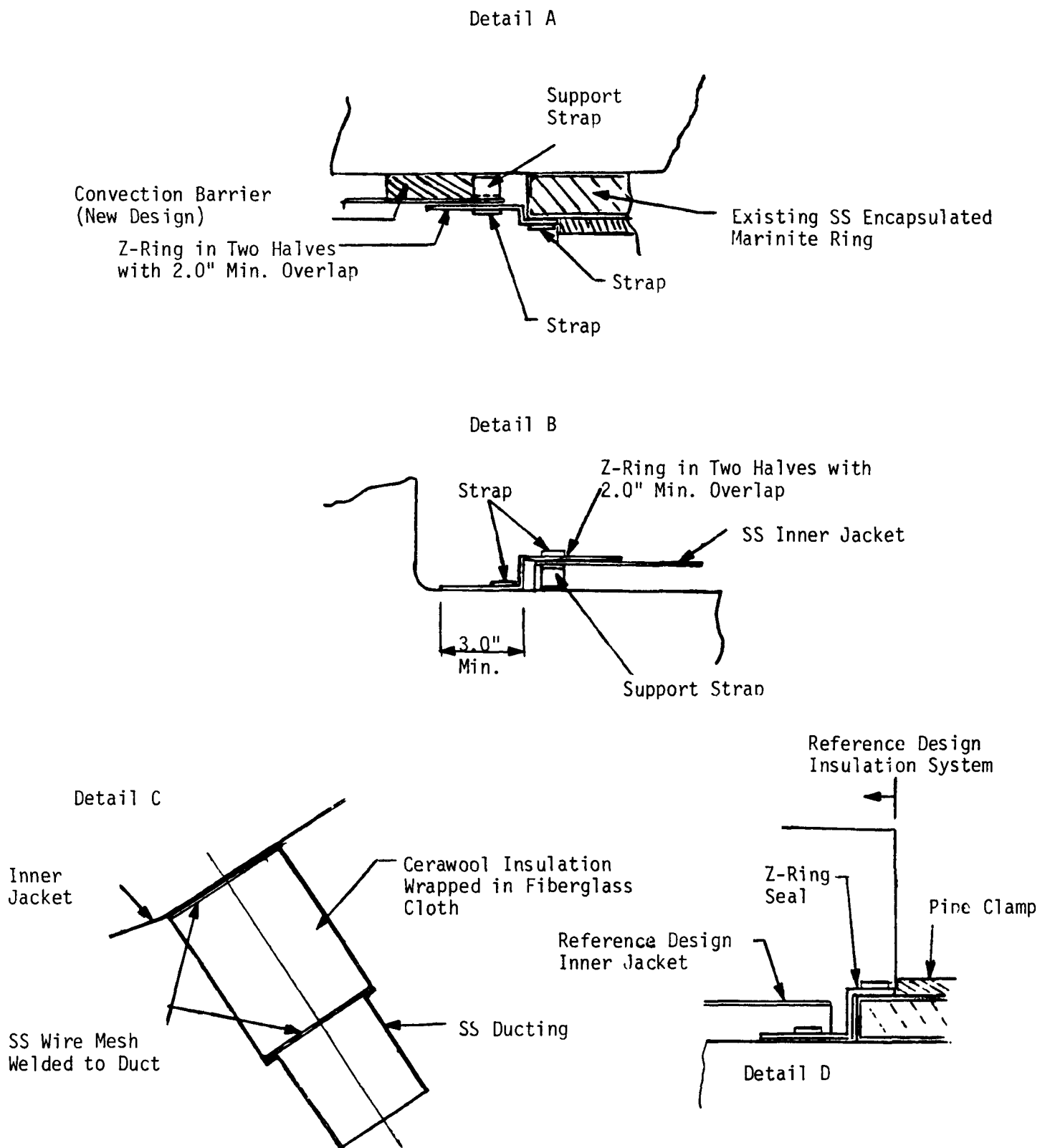
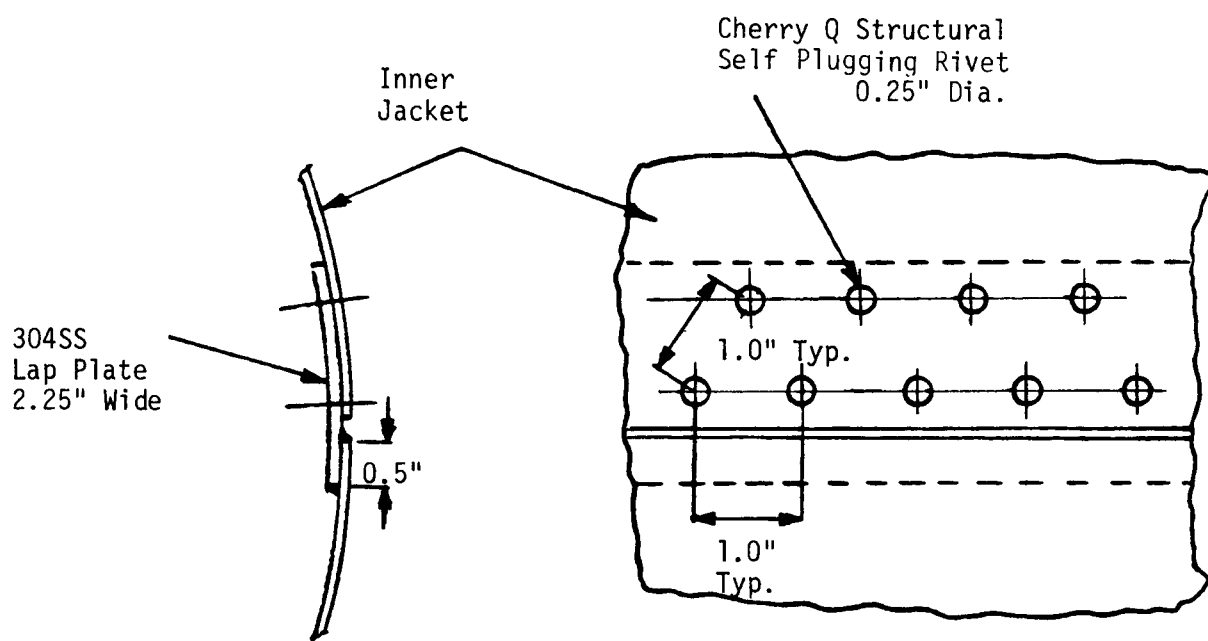


Figure 7-3 IHTS Leak Jacket Details and Interfaces

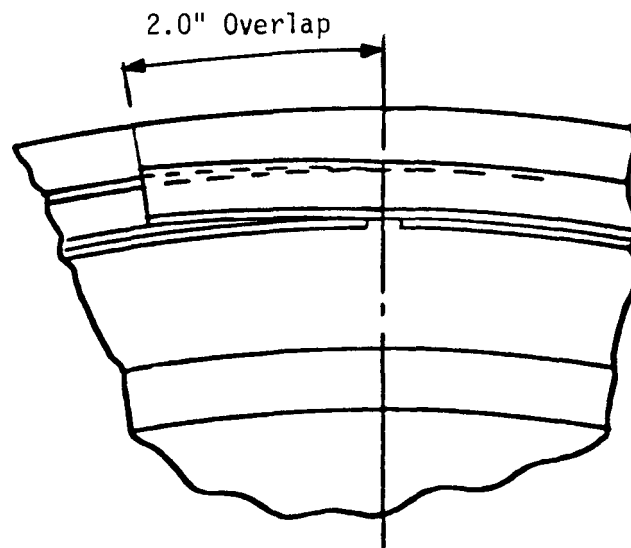
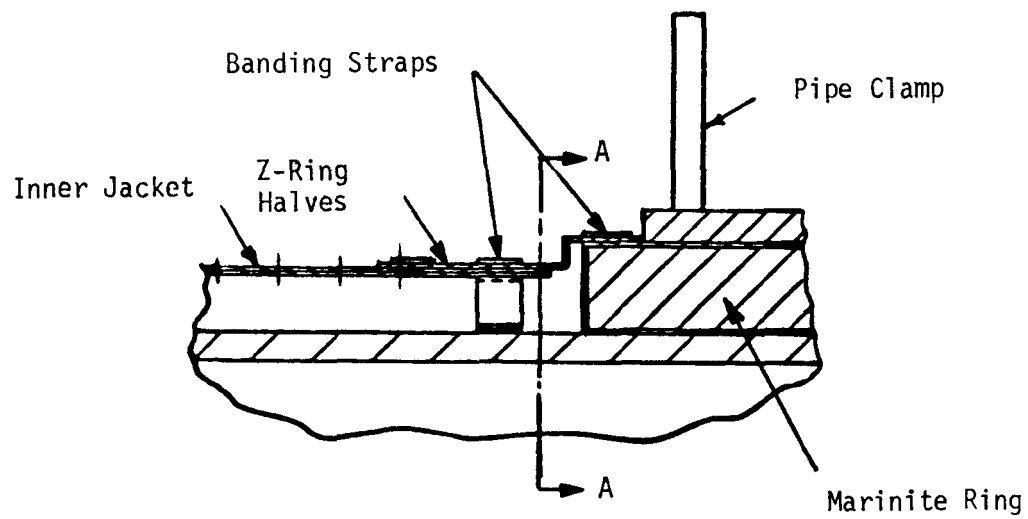


Detail E

Figure 7-4

IHTS Inner Jacket Attachment Detail

Z-Ring Seal



Section A-A

Z-Ring Overlap Showing
Tapered Section

Figure 7-5 IHTS Z-Ring Details at Pipe Clamp

lap plate of the same thickness as the inner jacket welded to one inner jacket panel and pop-riveted to the mating panel (as shown in Figure 7-3, Detail E). Two rows of 0.25 inch diameter rivets, with a 1.0 inch triangular pitch spacing were to be used. Structural self-plugging rivets, Cherry Type Q or equal, were to be specified for the longitudinal seams. These rivets are vibration resistant and have a specially designed mandrel to provide a seal, upon installation, that offers high resistance to leakage.

The inner jacketing for elbows had an 18 segment mitered configuration built in two halves with the axial seams located along the elbow intrados and extrados. Longitudinal seams were provided with lap plates and pop-riveted in the same manner as for straight pipe inner jacketing. At interfaces between elbows and straight piping, the adjacent inner jacket circumferential seams were full penetration butt-welded.

Interfaces with pipe clamps and component nozzles were sealed with a Type 304 stainless steel (SS) Z-ring, made in two halves with a 2.0 inch overlap and clamped around the inner jacketing and the pipe clamp SS encapsulated Marinite ring using banding straps (as shown in Figure 7-2, Details A and B). The Z-ring had the same thickness as the inner jacketing.

Figure 7-2, Detail D shows an adaptation of the Z-ring seal installed at pipe clamps beyond which the leak jacket features are not required for leak mitigation. The Z-ring did not affect the installation of the reference design insulation in this area of the piping.

At least one drain connection was provided for every 10 foot run of piping between a component and a pipe clamp and between two pipe clamps. The drains were sized to limit the back pressure buildup in the inner jacketing/pipe annulus to less than 2 psi.

7.2.4.4 Leak Jacket Tests

Leak rate and drain flow characterization tests were performed, using water, on the insulation system inner jacketing design concept proposed for mitigating sodium leaks in the CRBRP IHTS piping and described previously.

The primary objective of the testing was to evaluate the leak rate performance for 24 inch horizontal piping and to verify that the leakage would be mitigated to less than 5 percent of the design basis leak for the IHTS 24 inch piping in the CRBRP Steam Generator Building. Another objective was to record the drain flow and leakage spray on high speed movie film and video tape for evaluation of various characteristics. Detailed discussions and descriptions of the tests and findings are presented in Reference 18.

The test assembly, shown in Figure 7-6, included the inner jacketing for both elbow and straight pipe sections, Z-ring seals, drain connection, and interfacing pipe clamps mounted on a 24 inch pipe assembly. The test assembly also included penetrations for System 68 heaters and thermocouples and a System 66 leak detector. An insulation plug was inserted in the drain for the first test run. Ambient water was used for the testing with flow and structural design parameters appropriately scaled to simulate sodium leak performance under the worst case design basis leak conditions in the IHTS piping.

Sharp-edged orifices, equal in area to the design basis leak area for IHTS 24 inch piping, were installed in the pipe assembly at locations selected to yield the worst case leakage. These locations, which were subjected to jet impingement forces, included the center of the elbow external radius (at the inner jacket bolted seam) and the inner jacket/Z-ring interface along the inner jacket bolted seam.

Leakage flow from the test assembly and the drain effluent were collected separately for each test run. Also, during two test runs, the leakage flow was collected separately from each inner jacket penetration and from the Z-rings, pipe clamps, and inner jacket seams.

It was determined, following the initial testing, that a modified Z-ring configuration would be required to reduce excessive leakage from the pipe clamp area at the Z-ring interface. The modification consisted of a sheet metal band in two overlapping halves that was installed between the pipe clamp Marinite rings and the pipe clamp as shown in Figure 7-7. The band was seam

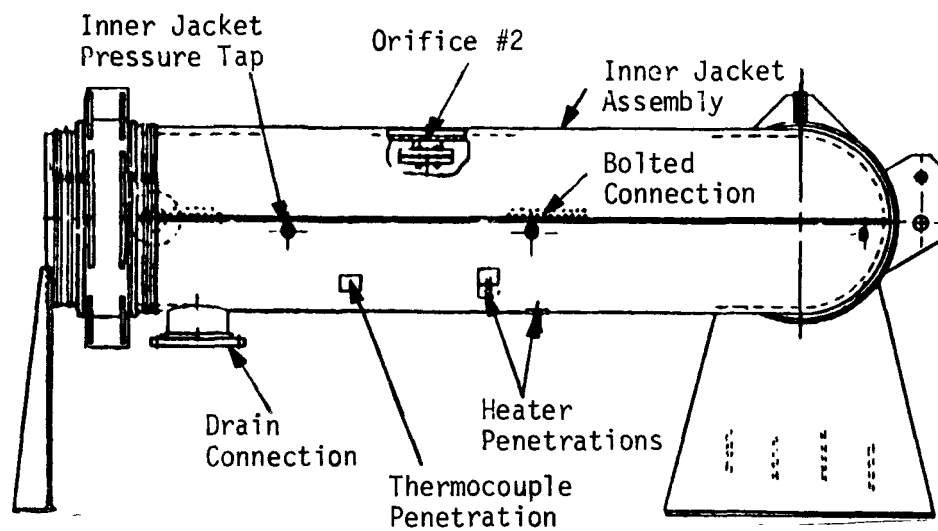
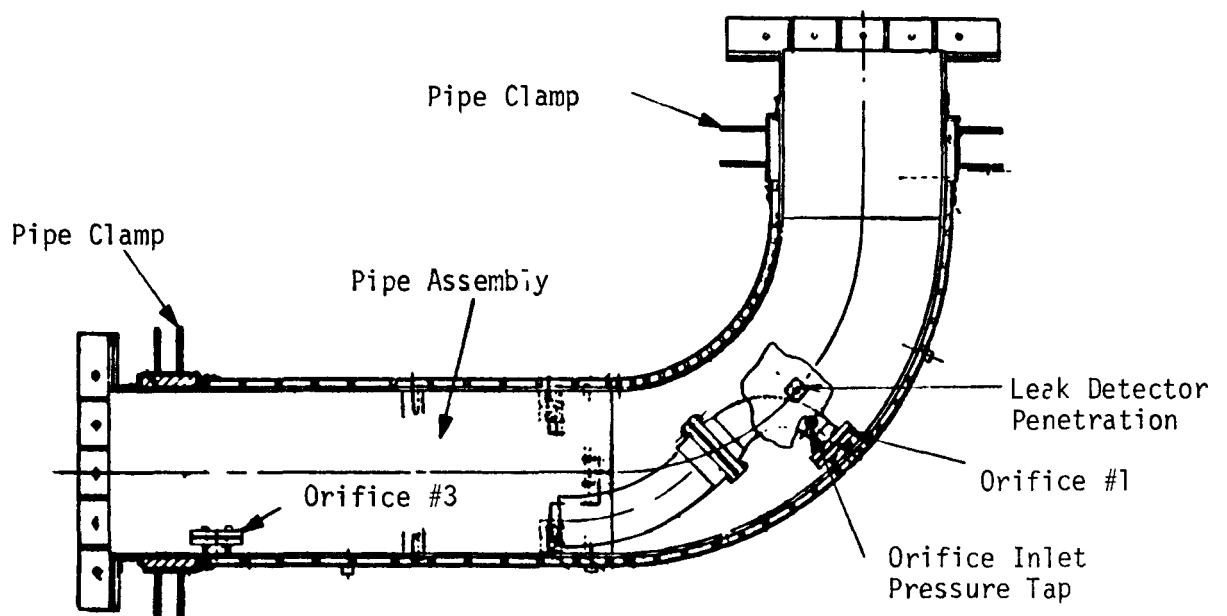


Figure 7-6 IHTS Insulation Leak Test Specimen Assembly

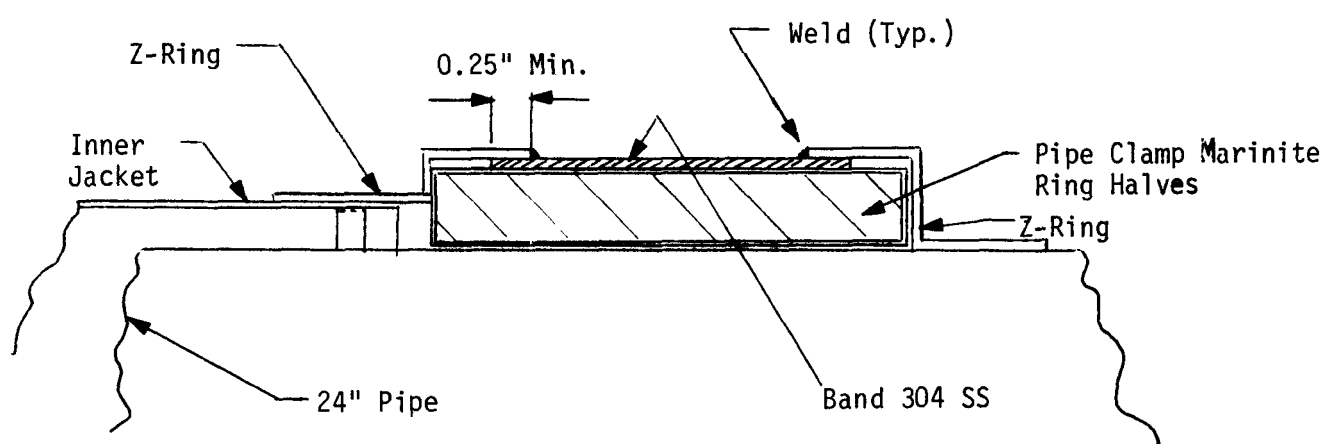


Figure 7-7 IHTS Modified Z-Ring Seal Over Marinite Ring Halves

welded to the Z-rings on both sides of the pipe clamp. Leak testing was repeated with the modified Z-rings and showed a significant improvement in the leakage performance of the test assembly.

Drain flow characterization tests were conducted with the drain effluent impinging on a heavy duty grating installed above a Q-deck to simulate CRBRP plant conditions. Significant flow rotation was observed during the first drain flow test resulting in the drain effluent assuming a conical shape. The vorticity also resulted in annulus pressures that were higher than predicted by pretest evaluations. A cruciform shaped flow straightener was inserted into the drain connection and the drain flow and leak tests repeated for the orifice location yielding the worst leakage. The flow straightener eliminated most of the vorticity and also resulted in lower annulus pressures, which in turn, contributed to lower leakage.

The leakage flow was characterized by focusing on the leak from the one location on the test assembly that produced a jet which transformed into spray. This leak path was on the outboard Z-ring-to-Z-ring interface along the vertical face of the pipe clamp Marinite ring. The leakage from this location was collected and measured in a separate test run.

The test program was successful in meeting all the objectives and demonstrated the feasibility of the leak jacket concept to mitigate sodium spray events. The principal conclusions derived from the testing were:

- A. The computed total sodium leakage based on water testing and scaling was 4.72 percent for a typical IHTS piping application and meets the design objective of 5 percent maximum design basis leak for the Steam Generator Building.
- B. Transformation of a leak jet to a spray occurred principally around the pipe clamp region and amounts to less than 1 percent of the design basis leak.

- C. The spray droplets observed from water testing had an average diameter of 0.25 to 0.35 inch. The droplet sizes should not be significantly different with liquid sodium.
- D. A flow straightener (cruciform) in the drain connection was required to minimize vorticity and reduce annulus pressures.
- E. The drain insulation plug does not offer any restriction to flow and will be blown out by the impinging sodium.

7.2.5 Leak Jacket Design Evaluation

It has been shown that the leak jacket design concept can be applied to the IHTS piping and be effective in mitigating design basis leaks in air-filled cells. However, the design is complex and costly, and the potential for uncertainties in actual plant use is extremely high. This section discusses some of the major problems and concerns envisioned for the leak jacket design concept as applied to CRBRP IHTS piping.

7.2.5.1 Cost Considerations

Prior to termination of the CRBRP Project, ECP A26036 was being prepared to baseline the plant mitigation features required to accommodate postulated liquid metal spills in air-filled cells [5,6]. Since the insulation for the IHTS piping in the SGB was not initially designed to provide sodium fire protection, a complete redesign was required. However, the redesign had to remain within specific requirements, such as weight and size limitations, in order not to invalidate prior analyses and testing.

The key changes to the piping insulation design was to provide a drain connection between each pair of piping support points (clamps), modify the inner jacket/pipe clamp interface by providing a Z-ring seal over the pipe clamp Marinite ring halves, and to fabricate the inner jacket in two halves which was either bolted or riveted together. The location of modified insulation on this piping was to be based upon using modified MEB 3-1 criteria

for defining locations where leaks were postulated (previously discussed in Section 7.2.3.1). However, cost estimates were developed for postulating leaks at any piping location.

In conjunction with the modified insulation concept, drain extensions (downspouts) were to be provided to direct the discharged sodium away from building steel and components to a height of approximately 6 feet above the sodium catch pans and fire suppression decks.

Preliminary cost estimates [19] prepared by the CRBRP Project participants were on the order of \$5-10 million (depending upon the extent of application of the MEB 3-1 criteria). These estimates were only preliminary and based upon many assumptions. Also, certain aspects of the proposed design and resulting effects were not fully developed and understood and additional testing/qualification was to be required. It is believed that the actual cost of the insulation redesign would have exceeded the initial estimated costs after plant construction due to these uncertainties.

7.2.5.2 Small Sodium Leaks

The effects of a small sodium leak on the integrity of the leak jacket was also a part of the MEFS leak mitigation design study [20]. Concern for the integrity of the segmented thin-wall inner jackets arises because of the potential for chemical reaction in the inner jacket/intermediate sodium piping annulus. The hot sodium resulting from a small leak into this annulus would react with the air, which is continuously refreshed at a slow rate by a vacuum pumping system which controls the air flow to the leak detectors. The products from the sodium-air reaction would react with the moisture in the air forming sodium hydroxide (NaOH) which is potentially corrosive to the inner jacket. If holes or weak spots resulted from such corrosion, there would be the potential for a large leak to bypass the drain system and break into the SGB in an uncontrolled manner. There was also a concern that reaction products from a small leak could plug the drain system orifices. Studies were performed to examine the existing sodium corrosion data, especially the effects of NaOH corrosion on candidate materials from which the inner jacket could be constructed, and the leak detection requirements which enable timely shutdown and repair in the event of a small sodium leak.

It was shown that in the event of a small sodium leak in the IHTS, the integrity of the inner insulation jacket would not be compromised if:

- A. The inner liner for all high temperature piping is constructed from Alloy 600.
- B. The leak detection system can detect leaks if 50 grams of sodium are leaked into the annulus in 100 hours.

However, since the NaOH corrosion data have not been measured in the specific IHTS environments, it was recommended that relative corrosion rates of stainless steel and Alloy 600 be measured under these conditions and that additional analyses be performed.

7.2.5.3 Downspouts

Downspouts or drain extensions would be required to direct the discharged sodium to the catch pans and fire suppression decks. Due to the quantity of locations on the IHTS piping requiring leak jacket protection in the SGB, the downspout routing would be extensive to avoid interferences with existing piping and/or structures and also prove to be very costly.

7.2.5.4 In-service Inspection/Degradation

The leak jacket design concept is complex and relatively fragile. In light of this, extreme care must be maintained during plant construction, start-up, and operation to maintain its integrity. Precautions would have to be taken to ensure the "leak tightness" of the design. This may not always be possible since piping insulation is usually stepped on by plant personnel or used as anchor points for lifting components.

The leak jacket design must also accommodate in-service inspection requirements for the IHTS piping imposed by the ASME B&PV Code. Therefore, these concerns should be evaluated further.

7.2.5.5 Licensing Considerations

The leak jacket concept as applied to the IHTS piping may have been considered an engineered safety feature (ESF) by the NRC. The CRBRP leak jacket concept was never formally presented to the NRC due to Project termination.

Therefore, considerable effort would need to be expended and the rationale clearly defined and organized to logically establish the leak jacket concept as an ESF. As a minimum, the leak jacket could be considered a passive safety-related component. Specific impacts and obstacles which must be addressed in the licensing process regarding Quality Assurance requirements, design, testing, and in-service inspection are defined in Reference 21.

7.2.5.6 Additional Consequences/Concerns

The leak jacket design directs approximately 95 percent of the MEFS DBL down to the fire suppression decks. However, the 5 percent that is allowed to spray into the air-filled SGB cells could create problems for components located near the leak area or in the direct path of a leak jet. These include pipe hangers and mechanical snubbers, which are safety-related component supports. Therefore, it must be demonstrated that these components are adequately qualified for the 5 percent MEFS DBL condition.

Tests were performed on prototypic pipe hangers and mechanical snubbers in the large scale sodium fire tests in May 1983 (see Section 2.3.1 of this report). The components remained functional after being in the sodium fire atmosphere. However, they were not directly impinged upon by a sodium jet or spray. Additional tests (or analyses) would be required to demonstrate that they remain functional after being subjected to sodium accident conditions. If their operability cannot be demonstrated, protection by means of shielding would have to be considered to deflect the sodium jet/spray from impinging directly on the hangers and snubbers. Since there are numerous pipe supports on the IHTS piping (greater than 1000), shielding would be very expensive and must also address the accessibility for in-service inspection of the component supports.

Other considerations which must be evaluated are the impact of the leak jacket design on the trace heating and leak detection systems common to sodium piping systems. Also, the impact of the leak jacket design near branch connections must be addressed.

7.2.6 Conclusions

The leak jacket design concept for the CRBRP IHTS piping in air-filled cells has been described. It has been shown that the concept can be made effective in limiting the sodium leakage and resulting fire effects to acceptable levels. However, the proposed design is complex in regards to fabrication, installation, maintenance, and inspection. More importantly, the cost for implementing such a design is extremely high. Also, many uncertainties must be addressed, resolved, or verified to demonstrate technical feasibility and licensability.

The conceptual design for future LMFBR plants should consider mitigation of sodium spray fires in air-filled cells as a design requirement. This will be less complex and costly than as for CRBRP, where the revised insulation design had to be backfitted to satisfy the requirement. However, the insulation package with this capability will be significantly more complex, and issues concerning in-service inspection, installation, design life, and licensing impact will be raised. It is recommended that other options be investigated along with insulation leak jacketing before continuing the development of the CRBRP leak jacket design concept for future LMFBRs. If only short sections of piping need to be leak jacketed, this may well be the best approach.

7.2.7 References

1. Berg, G. E., "Sodium Spray Fire Task Force Report," N099ER26003, September 1982, Rockwell International, Energy Systems Group, Canoga Park, CA.
2. Deleted
3. Deleted

4. Letter CR-747:DF:83-044, C. H. Fox, Jr. to W. J. Purcell and M. C. Ascher, "PO Authorization to Proceed with Plan for Na Fires in Air-Filled Cells," dated March 4, 1983.
5. Letter LR830533, G. H. Clare to P. Brewington, Jr., "CRBRP; Draft ECP A26036, Accommodation of Liquid Metal Spills in Air-Filled Cells," dated September 30, 1983.
6. Letter 83ESG-6196, K. W. Foster to W. W. Dewald, et al., "ECP A26036, Revision 0," dated October 18, 1983.
7. "Fast Flux Test Facility (FFTF) Final Safety Analysis Report," Appendix B, NRC, Public Document Room, 1717 H Street, Washington, DC.
8. "Safety Evaluation Report related to the construction of the Clinch River Breeder Reactor Plant," NUREG-0968, Vol. 1, Main Report, March 1983, U.S. Nuclear Regulatory Commission, Washington, D.C.
9. "Postulated Rupture Locations in Fluid Systems Piping Inside and Outside Containment," NRC Branch Technical Position MEB 3-1, U.S. Nuclear Regulatory Commission, Washington, D.C.
10. Mello, R. M. and Hundal, R., eds., "Integrity of Intermediate Heat Transport System Piping Ex-Containment," ES-LPD-83-011, September 1983.
11. "Quality Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants," NRC Regulatory Guide 1.26, Rev. 3, May 1976, U.S. Nuclear Regulatory Commission, Washington, D.C.
12. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants-LWR Edition," NUREG-0800, December 1977, U.S. Nuclear Regulatory Commission, Washington, D.C.
13. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Subsection NC, New York, NY.
14. Hundal, R., ed., "Design Basis Leaks for Ex-Containment Sodium Piping," ES-LPD-83-002, Rev. 1, June 1983.
15. "Construction of Class 2 or Class 3 Components for Elevated Temperature Service," Code Case N-253-1, American Society of Mechanical Engineers, Section III, Division 1, New York, NY.
16. Mallett, R. H. and Nair, B. R., "Integrity of Primary and Intermediate Heat Transport System Piping In-Containment," CRBRP-ARD-0185, Volumes 1 and 2, October 1977.
17. Smith, R. A., et al., "CRBR Insulation System Design Concepts for MEFS Pipe Leak Mitigation," AESD-TME-3174, Rev. A, February 1983.

18. Nair, B. R. and Smith, R. A., "Clinch River Breeder Reactor Plant IHTS Insulation Fire Mitigation Leak Test Report," AESD-TME-3190, September 1983.
19. Letter LA830354, W. J. Purcell to R. J. Beeley, et al., "CRBRP; ECP A26036 Accommodation of Na/NaK Leaks in Air-Filled Cells," dated October 19, 1983.
20. Svedberg, R. C., "Clinch River Breeder Reactor Plant, Effects of a Small Sodium Leak in IHTS Piping," AESD-TME-3191, October 1983.
21. Letter LSD-83-055, H. C. Huang to R. H. Mallett, "Impacts of Requiring Piping Insulation (Inner Jacket) as an ESF," dated September 12, 1983.

7.3 Sodium Pumps

7.3.1 Summary of Functions and Description

The CRBRP Sodium Pumps were to be used to circulate radioactive liquid sodium in the PHTS and non-radioactive liquid sodium in the IHTS. The CRBRP plant design included three primary loops and three intermediate loops, each of which required a pump. During normal plant operations all six pumps would have been in service, with the three PHTS pumps and the three IHTS pumps sharing the load equally; however, during startup and shutdown, the pumps were designed to support operation of one, two, or three loops.

The equipment specification [1] defined the pumps as free-surface, vertical-standing, centrifugal designs, with top-mounted variable-speed electric motor drives. This specification also required that the PHTS and IHTS designs be as identical as possible, that the pumps be designed, built, and tested in accordance with the ASME Code, as supplemented by RDT E15-2, and that the designs support a thirty-year service life. The principal steady-state functional design requirements are summarized below in Table 7-1.

7.3.2 Evolution of Design

Reference 2 also provides the history of the development of the prototype pump design from its inception in 1975 until delivery of the finished pump in 1981. The original intent of the ordering data [1] was to maintain a common design between the prototype and the plant, based upon a fabrication schedule wherein the plant units would follow directly behind the prototype; however, because of funding constraints, the plant pumps were delayed to the point that prototype assembly, water test, and delivery to ETEC was completed before the planning for the plant units was even started. Thus, when the plant pump fabrication effort was resumed in 1981-82, it was possible to consider the introduction of design improvements based upon the prototype fabrication experience.

TABLE 7-1
PLANT PUMP STEADY STATE FUNCTIONAL REQUIREMENTS (Note 1)

<u>Parameter</u>	<u>PHTS</u>	<u>IHTS</u>
Flow Rate, GPM	33,700	29,500
Head Rise at Design Flow, ft	458	330
Sodium Temperature, °F	400-1015	350-775
Availability	100%	100%
Maintenance Interval, hrs	10,000	10,000

Note 1: For the full set of pump design and operating requirements, see Reference 1.

Moreover, most of the CRBRP system changes which occurred during or after the prototype fabrication cycle in 1979-1980 were not imposed on the prototype because of the perceived need to maintain the ETEC delivery schedule. For this reason, the prototype pump did not meet the plant pump stiffness and mass requirements, or most of the other changes introduced on the plant design since about 1980.

The differences between the prototype and plant configuration are listed in Reference 2. Reference 3 provides a further description of the differences, along with an estimate of the effect that the plant changes could make to the prototype pump test results.

The pump designs were developed by Byron Jackson Pump Division of the Borg-Warner Corporation during the period from 1975 to 1983. The Byron Jackson design may be classified as a double suction, single-stage, free-surface centrifugal pump, with a three volute static hydraulic assembly. The PHTS and IHTS designs were virtually identical, differing only in interface detail and certain internal flow restrictions.

Figures 7-8 and 7-9 illustrate the prototype sodium pump. The prototype, which was fabricated, assembled, and successfully tested at the Energy Technology Engineering Center (ETEC) in 1982-83, was very similar in configuration to the PHTS/IHTS plant pump designs, differing principally in structural detail.

Each pump consisted of three major elements--the pump tank, the pump internal structure/volute, and the pump shaft/impeller. The shaft and the internal structure were designed to be removable vertically from the pump tank, which would have been installed as a permanent part of the heat transport loop. The pump tank featured a side suction nozzle (36-inch) and a side discharge nozzle (24-inch) at the same elevation, 90 degrees apart. The shaft was supported in the internal structure by two hydrostatic bearings located just above and just below the impeller; the shaft was connected to the drive motor by means of a diaphragm coupling which acted as a pin joint while transferring the weight of the pump shaft, through the motor shaft, to the motor thrust bearing. The top of the shaft was sealed by means of an oil-lubricated mechanical seal mounted

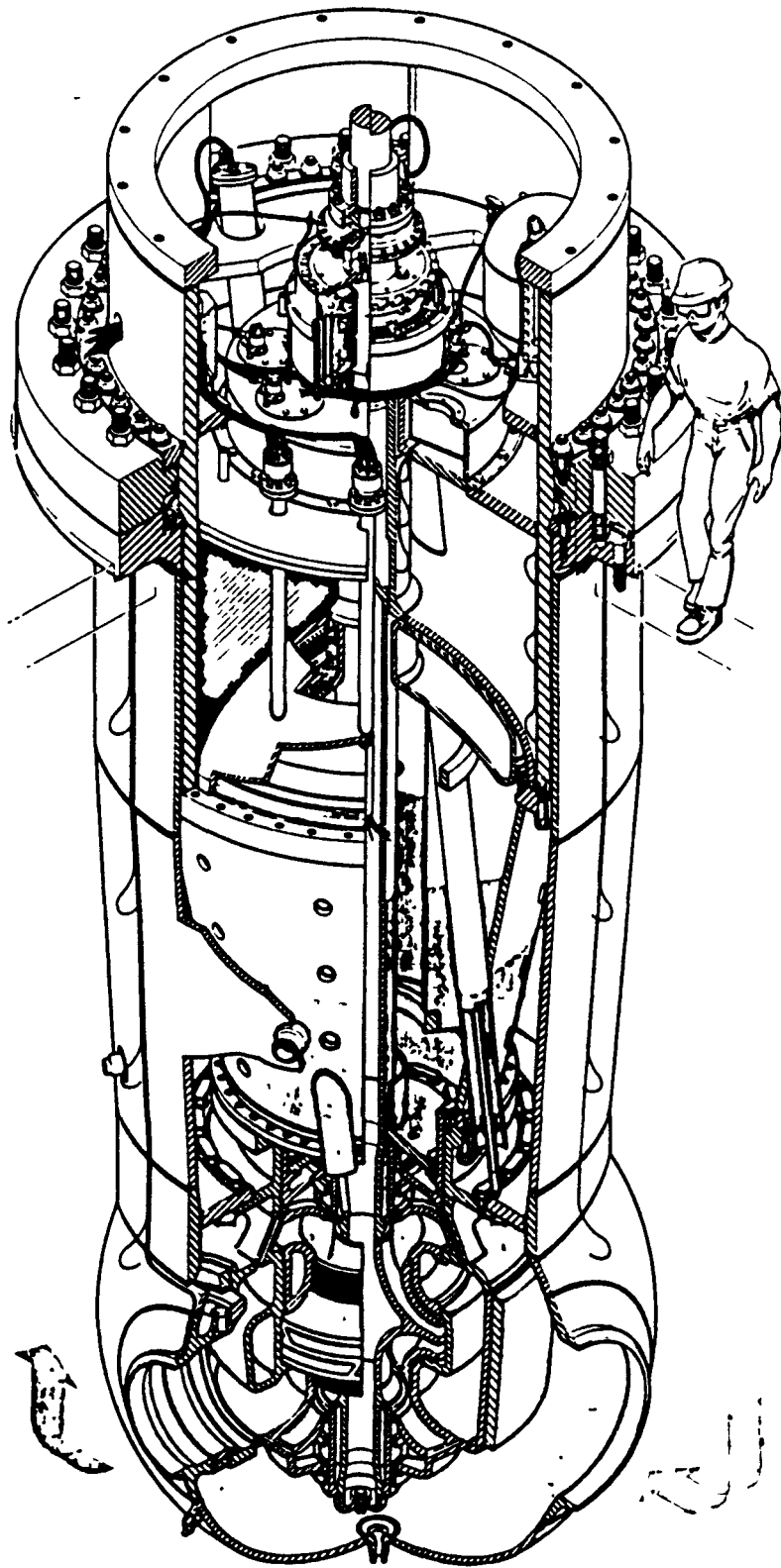


Figure 7-8 CRBRP Prototype Pump - Cutaway View

VERTICAL
DRIVE MOTOR

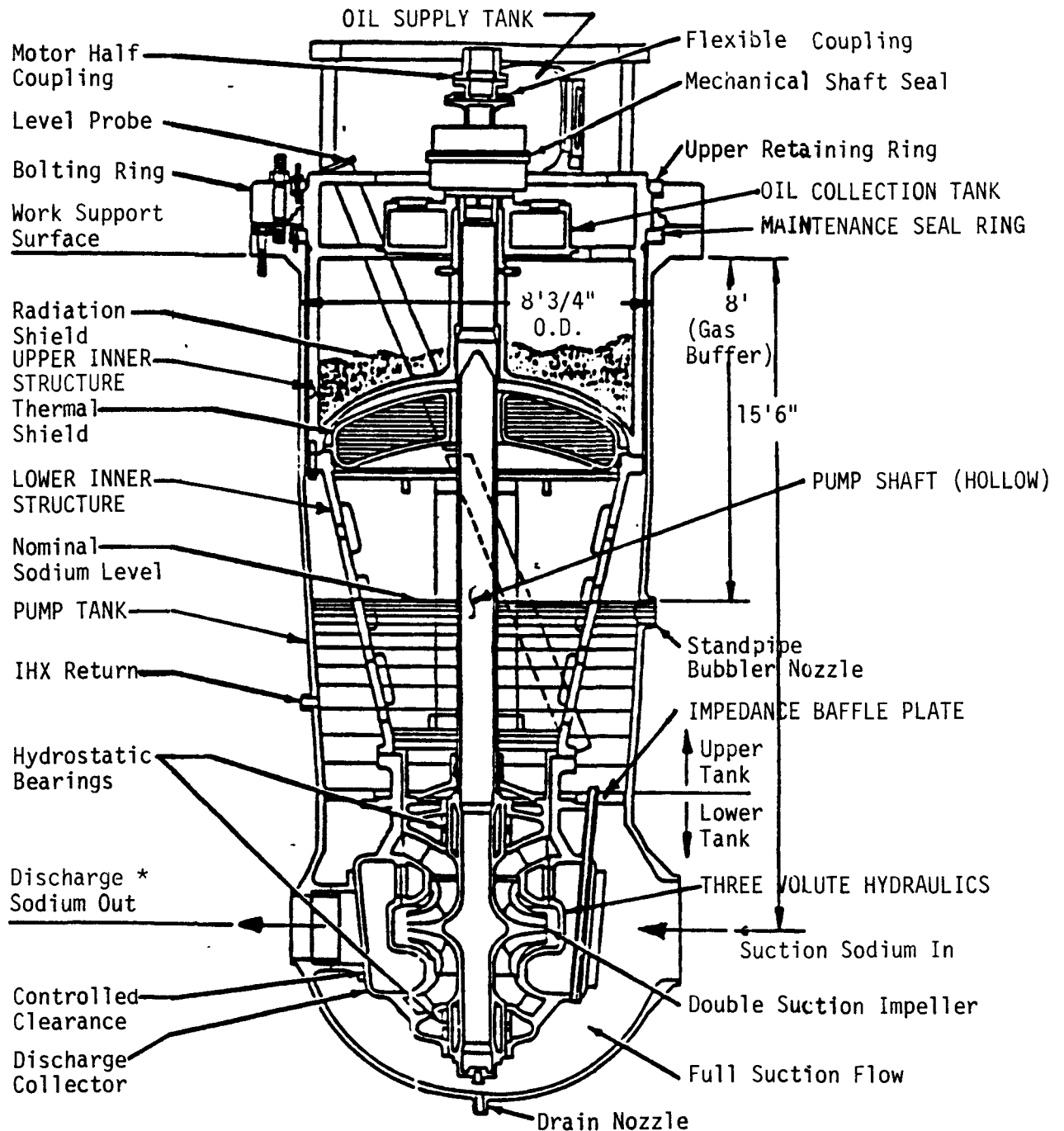


Figure 7-9 CRBRP Prototype Pump

*Note; suction and discharge nozzles were actually 90° apart, as shown in Figure 7-8.

on the internal structure. Thermal and radiation shields were also part of the internals, as was a lubrication system for the shaft seal. An internal baffle was provided to control the rate of sodium interchange between the pump zones during thermal transient conditions. The PHTS version of the pump included a standpipe bubbler to control the level of sodium in the pump tank.

A complete design description of the prototype pump is provided in Reference 2.

7.3.3 Key Problems and Resolutions

7.3.3.1 Cellular Convection through Pump Annulus Cover Gas Space

As shown in Figure 7-9 argon cover gas was introduced into the upper end of the pump as a buffer to control the sodium level. It was anticipated that the combination of hot sodium in the lower pump tank and a colder upper inner structure (UIS) would produce the driving force for convection heat transfer through the gas. Byron Jackson had hoped to break up any tendency for the transfer pattern to develop into cells by reducing the gap between the UIS and the pump tank; however, special tests conducted as part of the prototype pump test series at ETEC showed that the convection currents actually tended to form into an even number of cells, characterized as alternate columns of ascending hot and descending cold gas streams. Variation in the test gas pressure established a single pair case, whereby the hot and cold cells were directly opposed, resulting in a differential heat loading of the structure and consequent bending of the tank; this bending forced the shaft support structure to move relative to the shaft, to the point that rubbing of the upper end of the shaft in the shaft seal bumper bearing was detected.

PROBLEM RESOLUTION

Byron Jackson corrected this problem by reducing the annular gap between the pump tank and the UIS still farther, and by adding four equally spaced axial sheet metal baffles in the gap. Confirmation tests at ETEC demonstrated that the baffles effectively forced the convection streams to ascribe a balanced two-pair opposed configuration, eliminating tank bending. Reference 4

provides the thermal analysis of the pump with the baffles installed; Reference 5, the description of the field modification to add the baffles, and Reference 6, the record of the test results at ETEC.

7.3.3.2 Gas Entrainment at Very Low Liquid Levels in the Pump Tank

As part of water testing of the prototype pump, a series of tests were performed with the liquid at a level which approximated the result of a major pipe break combined with the maximum reactor cover gas pressure. With the water at this so-called minimum safe level (approximately 24 inches above the centerline of the suction/discharge nozzles) and the pump operating at pony motor speed (104 rpm), there was evidence of gas entrainment in the discharge flow from the pump--in apparent violation of the ordering data that required that "no vortexing or gas entrainment shall occur at any flow rate in the operating range."

Viewing ports had been provided in the test loop discharge line; through the use of backlighting, it was possible to make photographic records of gas bubble formation in the flow. However, in order to quantify the problem it was necessary to develop a means to accurately measure both the liquid level in the various chambers of the pump and the exact point that entrainment began.

PROBLEM RESOLUTION

Byron Jackson successfully modified the existing prototype pump tank instrumentation to add a manometer capable of accurately measuring the liquid levels in both the pump tank sump and the plenum on the top side of the upper hydrostatic bearing; with this manometer, it was possible to determine the starting conditions for the various entrainment tests.

During the early trial runs, it was noted that the output signal from an installed Leeds and Northrup #475 Ultrasonic Doppler Flowmeter was responsive to changes in the level of gas bubbles in the stream. By calibrating the flowmeter through the injection of small quantities of gas at a known flow rate, it was possible to validate the measurement of entrainment in the range

from 0.1 to 1000 ppm by volume. With this instrument, backed up with visual observations through the viewing window, it became possible to pin point the start of entrainment.

Running the pump at various liquid levels while at pony motor speed showed that the lowest entrainment-free level was 36 inches above the nozzle centerlines. Tests were performed in two modes: starting up from a dead stop, and with the level constantly falling at a known rate. In both cases the entrainment appeared to start when the level dropped below the bottom surface of the baffle plate which separated the upper and lower tank volumes, apparently as a result of vortexes forming on the free surface.

The problem was resolved by raising the minimum safe level for the plant primary pump to a level corresponding to 37 inches above the nozzle centerlines; at this level the baffle plate lower surface was covered, and the vortexes were not formed. Reference 7 provides a detailed description of the problem and the test results. The associated plant system change was implemented in ECP W1835 [8].

7.3.3.3 Incipient Cavitation Damage in Pump Bearing Following Sodium Test

Post-sodium test inspection of the prototype pump showed indications of incipient cavitation damage in the pockets of the pump shaft support bearings. While it was difficult to measure the extent of the damage, it was estimated that, based upon a constant rate of erosion, the 30-year design life of the pump would not be affected.

PROBLEM RESOLUTION

This problem was not resolved prior to termination; however, it was discussed at the final pump design review, in November, 1983, and resulted in Concern #5 from that meeting. It was agreed that the observed damage was atypical of previous hydrostatic bearings, and may have been as a result of the deep pocket design. At the time of CRBRP termination, General Electric was prepared to recommend a study program to investigate this problem,

particularly since the assumption of constant erosion rates was essentially unsupported and might have been unconservative. Reference 9 describes the observed damage to the prototype pump, and Reference 3 provides the report of the final design review.

7.3.3.4 Correlation of Water and Sodium Performance Measurements for the Pump

The measured head-flow performance of the pump at ETEC did not appear to be correlatable with the water test results for the same pump. This matter was not resolved at the time of CRBRP termination in November 1983.

PROBLEM RESOLUTION

Additional evaluation work was performed by Byron Jackson in 1984 as part of the CRBRP termination effort. As reported in Reference 10, they found that good correlation between sodium and water test performance was obtained using a correction factor for the thermal expansion of the impellers.

7.3.3.5 Qualification of Welds Involving CF8M Casting Material

The ordering data for the pump required that the fabrication welds used in the pressure boundary be qualified to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, as supplemented by RDT-E15-2. The original intent was to extend these requirements to those weld joints which were internal to the boundary and in contact with sodium or sodium vapor; however, application of the ASME qualification rules to those internal joints which involved the welding of CF8M casting material' resulted in the failure of several pre-production weld samples as they were subjected to a standard 180° bend test.

PROBLEM RESOLUTION

Following evaluation by both Byron Jackson and General Electric, it was concluded that the key variable which was causing the failures was the application of the post-weld heat treatments to the samples. Since the

primary object of the bend test was to demonstrate the quality of the deposited weld metal and its adhesion to the parent metal, it was decided that defects which could be related to the effect of the heat treatment on the parent metal could be discounted.

Accordingly, the equipment specification was rewritten to eliminate the requirement of post-weld heat treatment for the weld qualification samples. ECP G1321 [11] is the record of the change to the equipment specification.

7.3.4 Final Status of the Sodium Pump

7.3.4.1 Status of Design/Analysis/Fabrication

DESIGN/ANALYSIS STATUS

The design of the PHTS/IHTS sodium pump had been baselined via the approval of the Byron Jackson-prepared Design Layout Drawings; however, at the time of CRBRP termination, several minor changes were pending covering lube oil supply system interfaces, fastener locking devices and pump tank flange to foundation seal.

General Electric had also approved 26 of 29 designated "Top Level" Drawings (TLDs); these were Byron Jackson working drawings, most at the subassembly level, but others covering critical components. Open issues and pending changes against the TLDs involved fastener locking devices, interface details, jacking devices and removal of Hold on impeller OD.

Byron Jackson had completed almost all of the detail drawings for the pump; however, except for the critical components included under the TLD designation, these drawings were not subject to General Electric approval.

Portions of the Byron Jackson Design Report (e.g., pump assembly stress analyses and thermal analyses of upper/inner structure) and portions of the Dynamics Report were being reviewed by General Electric at time of CRBRP termination; other portions of these reports were in preparation at Byron Jackson.

The primary design requirements documents were the General Electric Equipment Specification, 22A3444, and 3 Interface Control Drawings: CG 52004, CG 52005, and CG 52006. At the time of termination, all of these documents were involved with pending changes as noted in Table 7-2.

General Electric had completed a final design review of the pump in November 1983; Reference 3 provides the minutes of the meeting, including the findings and concerns. In Reference 12, General Electric recommended resolutions for each of the 10 concerns that were identified at the review meeting.

FABRICATION STATUS

General Electric had granted a release-to-manufacture for most of the major pump subassemblies, based upon the approval of Byron Jackson-prepared Design Data Packages.

Fabrication had been started on several elements of the 6 plant pumps, both at the Byron Jackson factory in Vernon, CA, and at their major subcontractor, Foster Wheeler, Mountaintop, PA. All of the critical long-lead material had been ordered and delivered to the fabricators. A significant part of the special jigs, fixtures, and tools had been built or were in construction.

7.3.4.2 Outstanding Activities to Complete Design

The major tasks to complete the design/analysis of the plant sodium pumps were as follows:

- o Revise lube oil system interfaces (ECP G1027).
- o Add cover gas accumulator to IHTS pumps (ECP G1436).
- o Add fastener locking devices.
- o Resolve impact of sodium aerosol environment on shaft seal (ECP L10-200).
- o Add fire suppression provisions.
- o Establish final trim for impeller.
- o Resolve review comments on Dynamics Report.

TABLE 7-2
PENDING CHANGES TO PLANT SODIUM PUMP DOCUMENTATION

<u>ECP No.</u>	<u>Change Description</u>	<u>Impact On Pump</u>
G1027	Redefine Oil Supply Interface	Required new oil lines and supports in upper inner structure.
G1105	Modify Pump Insulation Reqmts	Minimal
G1358*	Add O-Ring Seal between Pump Tank Flange & Foundation	Minimal
G1436	Modify System 82 Supply to IHTS Loops to Provide Isolation and Over-pressure Protection	Required addition of accumulator to provide 30-day argon cover gas supply for IHTS pumps.
G1451	Revise Nozzle Interface Loads	Unknown
G1452	Modify Interface of Argon Gas and Standpipe Bubbler Nozzles	Required remachining of nozzles.
G1469	Add Locking Devices to Critical Internal Fasteners	Required redesign of impedance baffle plate and addition of field-installed welded caps on all critical fasteners.
G1492	General Update of Instrumentation Interfaces with Pump	Minimal
G1494	Remove Spare Accelerometers from Shaft Seal	Minimal, but saves cost of procurement.

* - ECP G1358 had been disapproved, but General Electric planned to reopen.

TABLE 7-2
PENDING CHANGES TO PLANT SODIUM PUMP DOCUMENTATION (Continued)

<u>ECP No.</u>	<u>Change Description</u>	<u>Impact On Pump</u>
L10-200	Revise Sodium Aerosol Environment	Required new thermal analysis or test of Shaft Seal.
W2038	Revise PHTS Instrumentation	Minimal
--**	Modify Attachment of Oil Collection Tank to Upper Inner Structure	Minimal; already incorporated in pump hardware, at vendor's risk.
--**	Increase Length of Pump Bearings	Ditto
--**	Modify Test Requirements for Weld Filler Material in Oil Supply Tank	Minimal

** - ECP had not been submitted prior to termination.

- o Complete Design Report.
- o Complete Installation and Operation Manuals.
- o Complete Acceptance Test Procedures.

7.3.5 References

1. CRBRP Specification 22A3444, "Primary and Intermediate Coolant Pumps," Rev. 143, September 1983.
2. "Final Report, Prototype Pump Development Primary and Intermediate Coolant Pump for Clinch River Breeder Reactor Plant," Byron Jackson Report ER-1223, September 28, 1983.
3. "Design Review Report, System 51 IHTS/PHTS Sodium Pump," GE Memo YL-595-830692, November 16, 1983.
4. "Prediction of Effect of Cellular Convection on CRBRP Prototype Pump and PHTS and IHTS Operating Conditions," Byron Jackson Report ER-1200, June 30, 1982.
5. "Field Modification, Cellular Convection Rework, CRBRP Prototype Pump," Byron Jackson Drawing B400003, June 14, 1982.
6. "Clinch River Breeder Reactor Prototype Sodium Pump Final Report, Prototype Pump Sodium Test Program," GE Report CRBRP-GEFR-688, November 15, 1983.
7. GE Letter XL-593-10228, LN Salerno (GE) to WJ Purcell (W-OR), "CRBRP; System 51A, Primary Sodium Pump Gas Entrainment," March 20, 1981.
8. ECP W1835 Rev. 0, "PHTS Minimum Safe Level Revision," October 8, 1982.
9. Letter, AG Parid, Rockwell International, Energy Systems Group to TJ Boardman, ETEC, "CRBRP Prototype Pump - Inspection of Hydrostatic Bearing Final Report," November 18, 1983.
10. J.Y. Lee, "Correlation of Hydraulic Performance in Water and Sodium for the CRBRP Main Coolant Pump," Borg-Warner Industrial Products Report ER-1248, September 1984.
11. ECP G1321, Rev. 4, "Fabrication of Pump Internals," May 27, 1983.
12. GE Letter XL-593-830139, RJ Michalak (GE) to WJ Purcell (W-OR), "System 51, IHTS/PHTS Sodium Pump Final Design Review November 15, 1983," November 28, 1983.

7.3.6 List of Major ECPs

Table 7-3 lists the major ECPs that were issued during the CRBRP program, against the following principal sodium pump control documents:

22A3444,	Equipment Specification
CG52004,	PHTS Pump Interface Control Drawing
CG52005,	IHTS Pump Interface Control Drawing
CG52006,	Pump Internals Interface Control Drawing
IE3862,	Prototype Pump Design Layout Drawing
IE4061,	Plant Pump Design Layout Drawing (PHTS)
IE4062,	Plant Pump Design Layout Drawing (IHTS)

TABLE 7-3
MAJOR ECP'S FOR CRBRP SODIUM PUMPS

<u>ECP NO.</u>	<u>Title</u>
G0001	Primary & Intermediate Coolant Pump Proposed Change to Rev. 2 of GE E-Spec 22A3444
G0014	Primary & Intermediate Coolant Pump Changes to Appendix B of GE Spec 22A3444
G0051	Primary & Intermediate Pump Instrumentation
G0116	IHTS Pump Support Stiffness & Mass
G0187	PHTS Pump Tank Sphere Increase and Offset; Pump Support Flange Elevation
G0211	Primary Sodium Coolant Pump Support Stiffness
G0258	HTS Pump/System 82 Cover Gas Interface Requirements and Revision to Shaft Seal Lube Oil System
G0267	IHTS Pump Support Stiffness
G0338	Baseline HTS (Plant) Pump Instrumentation
G0370	HTS Pump Inner Structure Support
G0400	HTS Pump - Restructure Design Layout
G0430	HTS Pump Addition of Balancing Port
G0516	Block and Evacuate Hollow Pump Shaft

TABLE 7-3
MAJOR ECP'S FOR CRBRP SODIUM PUMPS (Continued)

<u>ECP NO.</u>	<u>Title</u>
G0679	PHTS.Pump-Support Concept Stiffness & Masses, IHX Vent Nozzle Location & Bubbler Line Slope
G0855	HTS Pumps--Update of Documents with Maintenance Seal Requirements (Inc. a Static Maintenance Seal)
GJ060	PHTS Pump Internals Handling and Shield Plug Requirements
G1134	PHTS Pumps--Requirements to Preclude Loss of Sodium Level
L10-011	CRBRP Load Combination
L10-012	Third Level Structural Margin Requirements

8.0 STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM

The Steam Generator Auxiliary Heat Removal System (SGAHRS) provides an alternate heat removal path for removing the reactor decay heat and the sensible heat in the reactor and primary and intermediate loops following a shutdown when the main Nuclear Steam Supply System (NSSS) heat sink is unavailable. It can also function to remove shutdown heat loads for refueling and other long term outages.

8.1 Summary of Functions and Design Description

8.1.1 Functions

The functions of the SGAHRS are to:

- A. Provide the heat removal path and heat sink for those events which negate the normal heat removal path to the main steam condenser. The SGAHRS heat load includes the decay heat and the sensible heat released by the plant during cooldown to an isothermal plant condition determined by the steam-side saturation pressure.
- B. Provide a heat removal path which can be used to remove up to 1-1/2 percent of rated core power per loop for refueling or other long term outages.
- C. Provide make-up water to the steam generator system when the main feedwater supply is not available.

8.1.2 Design Description

The SGAHRS is a safety system that removes both decay heat and sensible heat from the reactor and the heat transport system if the main NSSS heat sink is unavailable. The system will be used primarily after a shutdown when the normal heat transport path to the condenser is unavailable. A schematic of the system is provided in Figure 8-1.

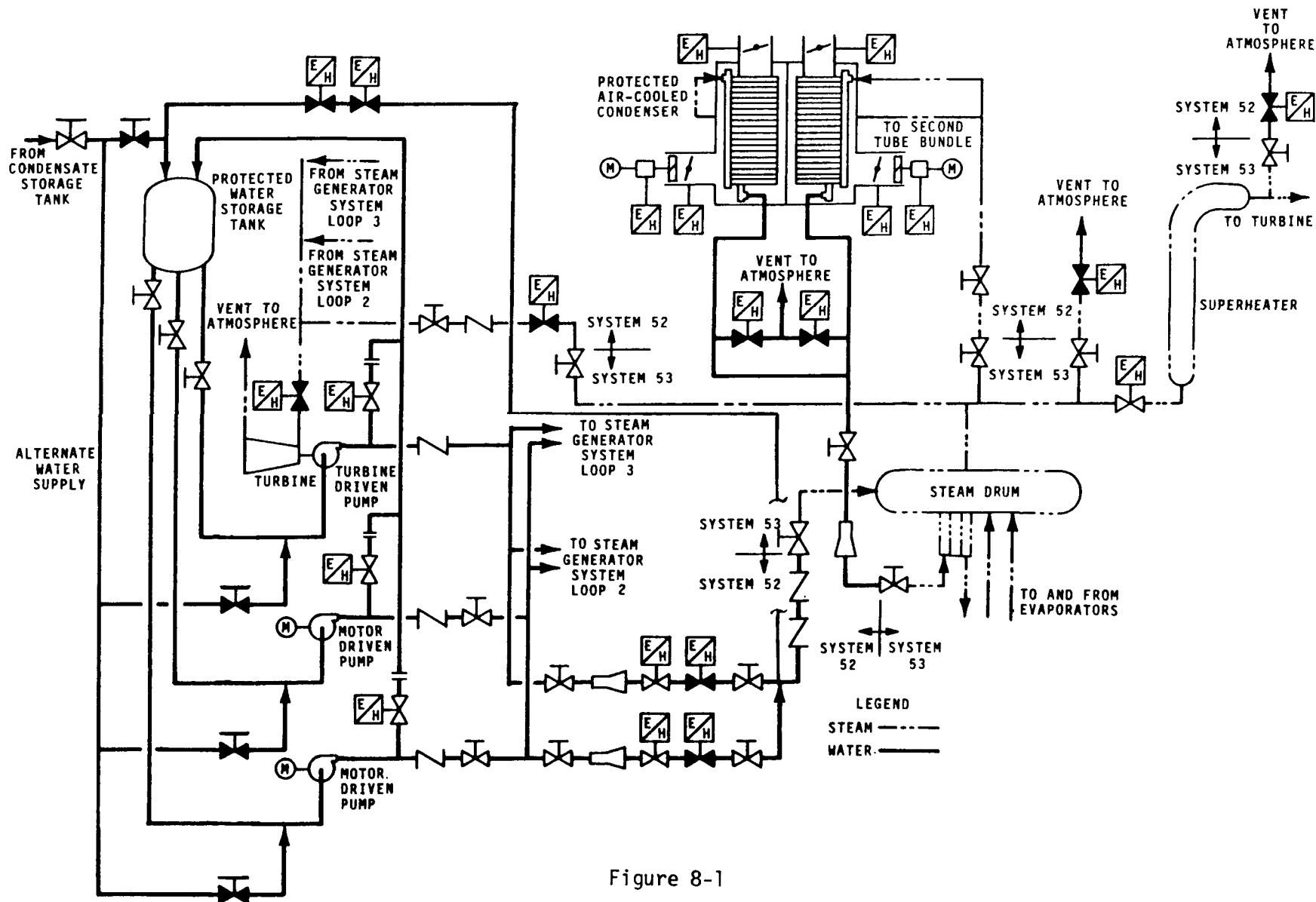


Figure 8-1
STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM SCHEMATIC

The SGAHRS design incorporates two subsystems--a long-term heat removal subsystem and a short-term heat removal subsystem.

The short-term system is called the Auxiliary Feedwater (AFW) Subsystem. It is used during the period (about one hour) immediately following reactor shutdown to remove the reactor decay heat and the sensible heat stored in the sodium in the reactor vessel and to remove the heat stored in the structural elements of the reactor and heat transport systems as the Steam Generator System (SGS) is cooled to a saturation temperature of about 600°F. It is automatically initiated by a high steam-to-main feedwater flow ratio or low steam drum water level on any loop. The subsystem dumps steam directly to the atmosphere through pressure-controlled vent valves located just off the steam drums and superheater exits. The expended water is replaced by water drawn from a protected water storage tank (assuming unavailability of the normal feedwater system) and pumped into the steam drums. When the heat load has decreased to a point where the total heat rejection service can be performed by the long-term subsystem, that subsystem takes over and steam venting ceases.

To assure an adequate supply of auxiliary feedwater for this short-term high-duty system, a protected water storage tank (PWST) with a usable capacity of 68,160 gallons is provided. This volume is adequate to supply the SGS with makeup water for 30 days. In addition, an alternate water supply is provided by a connection to the nonprotected condensate storage tank.

The AFW Subsystem has three feedwater pumps, which take suction from PWST. There are two electric-motor-driven pumps, each rated at 50-percent of the design flow rate, and one steam-turbine-driven pump with 100-percent capacity. The two motor-driven pumps are connected to headers and are capable of supplying water through control valves to each of the three steam drums. The pump drives can be supplied with either off-site electrical power or on-site AC power from the diesel generators (D-G). To assure one of the pumps is available after a single active failure, one drive is connected to one D-G and the other drive to another. The turbine-driven pump supplies AFW flow to each steam drum through an additional set of control valves, providing a parallel, redundant, and diverse water supply. Steam to drive the turbine comes from the SGS steam drums through a headered supply line. The

100-percent-capacity pump is rated at 1432 gpm with a head of 3927 feet. The 50-percent-capacity pumps are rated at half the flow at the same head, ensuring 100-percent redundancy.

The long-term subsystem is called the Protected Air-Cooled Condenser (PACC) Subsystem. It consists of three protected air-cooled condensers located on the roof of the auxiliary bay of the Steam Generator Building. Each PACC is a tube-type steam condenser made of carbon steel and contains two tube bundles, each with its own inlet and outlet louvers and fans for air flow isolation and control. Heat is rejected to the atmosphere as saturated steam rises to the PACCs from the SGS steam drums and is condensed by forced circulation of air over the tube bundles. The condensate is then returned to the SGS recirculation loop. The PACCs are located approximately 40 feet above the steam drums in a protected enclosure; the elevated location allows natural circulation on the steam side. The shell side (air side) of the PACC also can operate by natural circulation, but two fans are required to operate the unit at its design duty (15 Mwt). The PACCs are started automatically with SGAHRS initiation or with any reactor trip.

Under normal plant operation, the PACC louvers are closed to prevent air circulation and minimize heat losses. To increase overall reliability and minimize thermal transient effects, the lines between the PACC and steam drums are kept open.

A pressure actuated recirculation loop, from the AFW pumps discharge to the PWST, is provided to assure minimum flow through the pumps under low AFW flow demand conditions. A minimum of 25-percent of the pump design flowrate is required to prevent damage to the pumps.

To allow testing of the AFW pumps during ASME Boiler and Pressure Vessel Code, Section XI required in-service tests, a test loop line is provided from the loop 1 AFW line to the PWST. This test loop, in combination with the recirculation loop, allows the pumps to be tested without having to pump the PWST water into the steam drums.

8.2 Design Evolution of Steam Generator Auxiliary Heat Removal System

8.2.1 General

The initial reference design for SGAHRS is described in "Clinch River Breeder Reactor Plant Reference Design Report," Volume 1 [1] which was issued in June 1974. A schematic of the initial design is provided in Figure 8-2. The Reference design resulted from a study of several auxiliary heat removal concepts [2]. The system design description, SDD-52, Revision 84 [3], was the system design at the time of CRBRP termination. The following paragraphs in this section describe the major design evolution of the system.

8.2.2 Removal of Recirculating Feedwater Heater (ECP G0032)

In order to limit the thermal shock on the Steam Generator System (SGS) steam drums, recirculation loop and evaporator modules following an event (i.e., loss of feedwater) which requires SGAHRS feedwater, a jet pump powered recirculating AFW heating subsystem was employed. This subsystem mixed saturated water from the steam drum with the colder (~70° to 160°F) auxiliary feedwater from the PWST prior to its injection into the main feedwater inlet line. The AFW would then enter the steam drum at a temperature between 300°F and 400°F.

This subsystem was removed because the steam drum could accommodate the thermal shock of the unheated AFW associated with SGAHRS initiation. A separate nozzle, designed to allow the steam drum to withstand the thermal shock, for injecting the AFW directly into the drum was incorporated into the steam drum design.

8.2.3 System Additions to Meet Branch Technical Positions APCS 3-1 and 10-1 (ECP G0032)

NRC Branch Technical Positions (BTPs) APCS 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," and APCS 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants," had not been released

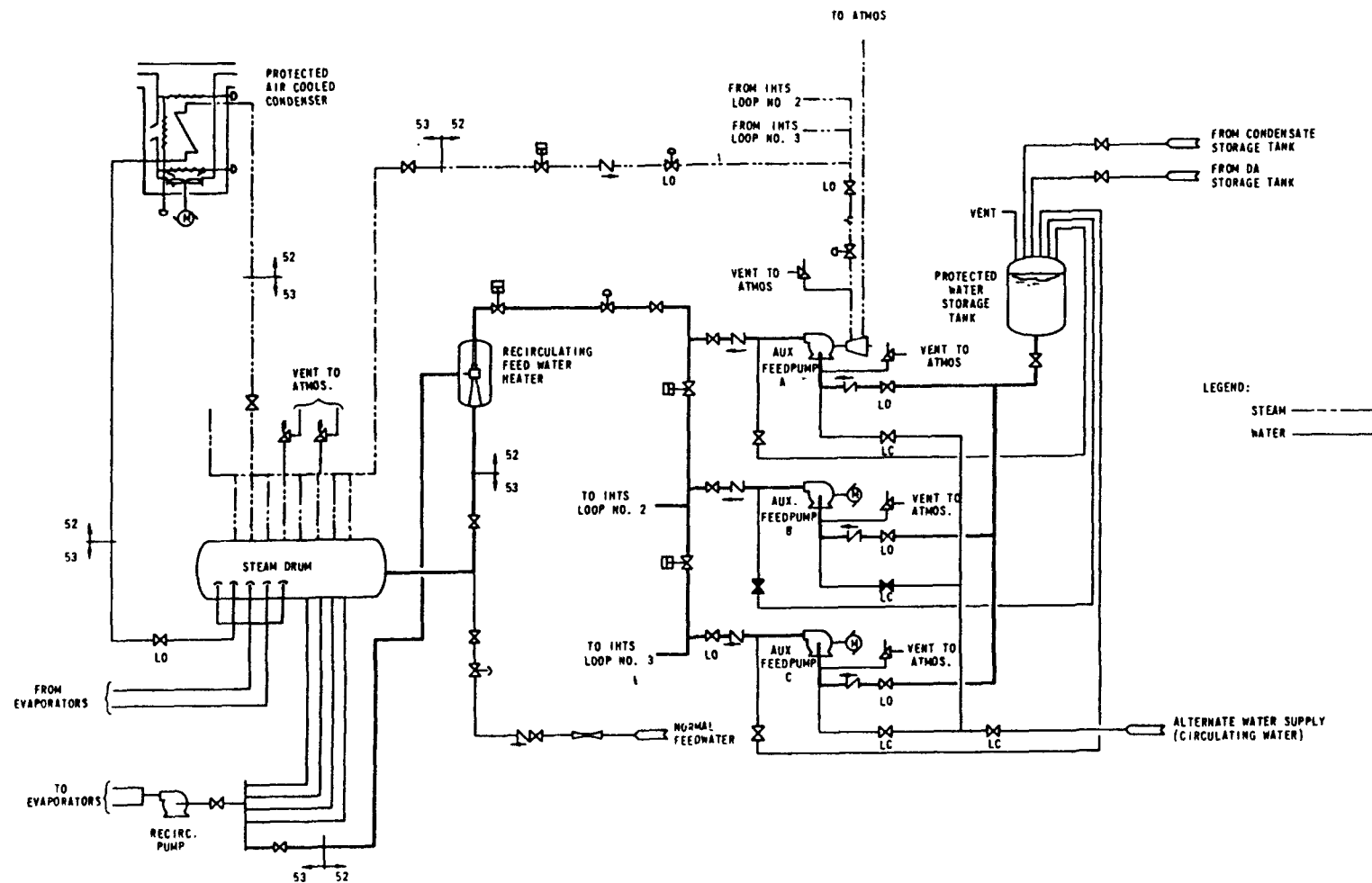


Figure 8-2
Process Streams in Relationship to Major Components Schematic

when the reference design [1] was established. Although they were issued for Light Water Reactors, they were determined to be applicable to the Clinch River SGAHRS [4].

Essentially, the BTPs required SGAHRS to perform its decay heat removal function after an initiating high pressure pipe rupture assuming the main feedwater system is not available (due to loss of off-site power) and assuming a single active failure is also incurred.

To accommodate the above requirement, a redundant AFW supply header, with associated flow control and isolation components, was added to the system. One header was supplied with AFW from the two motor driven pumps and the other header was supplied by the single turbine driven pump. A flow limiting device was added in each of the lines downstream of the headers to prevent excessive PWST water usage through the SGS loop with the pipe rupture. Pressure controlled vent valves, dedicated to SGAHRS, were added at the exits of the steam drum and superheater. The valves would vent steam to remove heat from the SGS during the initial period of SGAHRS operation. In addition, the venting would drop the SGS pressure to allow AFW flow into the steam drum (without an excessively high power pump).

8.2.4 Addition of Automatic AFW Isolation (ECP G0192)

SGAHRS utilizes a flow limiter on each AFW supply leg to maintain the flow below $105 \pm 5\%$ for all system operation. The reason for this is that should a pipe break occur anywhere in a steam/water system causing a depressurization of the steam drum, the flow limiter would prevent the AFW flow from exceeding 110%. This limits PWST volume pumped out the break, assuming a 10-minute operator action time to isolate the appropriate AFW legs. Should a pipe break be accompanied by a flow limiter function failure (single active failure), the volume of water from the PWST pumped out the break would be so great that the remaining AFW in the PWST would not be adequate to cover the 30-day requirement. The failure could be in the AFW control valve or actuator, the controller, or the flow transmitter.

To accommodate the above, capability to automatically close the appropriate AFW isolation valve was added. This change required additional flow transmitter and isolation logic for each AFW supply leg. The specific flow condition selected to provide the automatic isolation was 150% of AFW design flowrate for 5 seconds.

8.2.5 Addition of Non-condensable Venting (ECP G1088)

During normal plant operation, when PACC is on standby, non-condensable gases from the SGS would collect in the PACC. This would occur since the flow path from the steam drum to the PACC is left open and because the PACC is located above the entire SGS. The non-condensibles, if present in sufficient quantity, would prevent the natural circulation of steam from the steam drum when the PACC was operated.

Valves to vent the non-condensibles were added to assure maximum heat transfer in the PACC tube bundles. This venting would be automatic, based upon the difference between the drum steam temperature and the temperature in the non-condensable collection pipe in the PACC. The latter would drop, relative to the saturation temperature, as non-condensibles accumulate in the pipe.

This addition was made before system documents were baselined and subsequently revised by ECP G1088.

8.2.6 System Additions to Accommodate AFW Subsystem Pipe Break (ECP G0198)

If a break should occur in the high energy portion (downstream of the AFW isolation valves) of the AFW piping and be accompanied by a failure of the SGAHRS check valve at the steam drum on the same loop, two adverse circumstances could arise. First, the complete steam/water volume from one loop of the SGS would be deposited in the affected cell. If this were the cell where the SGAHRS turbine driven pump was located, the pump would be submerged and pumping redundancy would be lost. Second, the rise in cell temperature caused by the steam/water release could impact the operation of the AFW control valves. This would result in loss of AFW to one or two SGS loops, depending upon the break location.

To alleviate both of the above problems, a redundant check valve was added on each loop at the steam drum. To ensure operability of the check valve (and other check valves in SGAHRS), it was made to be testable.

8.2.7 Addition of PACC Initiation on all Scrams (ECP L10-054)

With the reference design, when a reactor scram occurred, SGAHRS was not initiated since the normal BOP heat sink and feedwater supply were available. However, this resulted in an overstress condition in the superheater outlet tube sheet, caused by excessive steam cooling.

To reduce this steam cooling, the capability to initiate PACCs, without the rest of SGAHRS, on all scrams was added. During scrams in which the balance of the plant is still available, the main turbine bypass system would continue to accept steam until the steam generator heat load reaches the PACC full operating capacity, at which time all steam generated would be condensed by PACC. For events in which the balance-of-plant has been lost; SGAHRS would be initiated, the superheater outlet isolation valves closed, and steam release through the SGAHRS vent valves would occur until the PACC could accommodate the entire SGS heat load.

8.2.8 Replacement of Pneumatically Operated Valves with Electrically Operated Valves (ECP G0389)

Because the SGAHRS is a safety related system, auxiliary systems supporting SGAHRS must meet the same availability requirements. To assure the operation of SGAHRS pneumatically operated valves, a redundant, seismically hardened, emergency air supply would be required. Since a hardened class-1E electrical supply was available and because the cost of a qualified air supply was high, the actuators were changed from pneumatic to electrically operated.

8.2.9 Mitigation of Sodium Aerosol Effects on PACC (ECP A26009)

In the event of a design basis sodium leak in the Intermediate Heat Transport System (IHTS) loop, sodium combustion product aerosols are released from the

steam generator building. Although design efforts were made to mitigate the aerosol release, up to 630 pounds could be released. After the first hour of the event, a maximum of 100 pounds could be released.

To assure the PACC heat transfer surfaces are not excessively fouled or the PACC air passages are not clogged, a means of delaying the startup (opening of louvers and activation of fans) of the PACCs for 5000 seconds was added. Monitors were added near the inlet of the PACCs to sense for sodium aerosol. The PACCs are not allowed to start or are stopped automatically for the above period of time. If the PACCs are still required after 5000 seconds, they would be restarted automatically. Ingestion of 100 pounds of sodium aerosol into a single PACC unit would not significantly reduce its heat removal capability.

8.2.10 Addition of AFW Pumps Inservice Testing Capability (ECP G0627)

To provide conformance with the ASME Code, Section XI, Article IWP-3000, "Inservice Test Procedures", AFW pump performance must be verified during quarterly inservice test. It was assumed that when Section XI was extended to sodium cooled reactors, the article would be applicable.

A test loop, with actuated isolation valves, was added between the SGS loop 1 valve station and the PWST to allow testing of the AFW pumps. Using the test loops with the recirculation loops, the motor driven pumps could be tested at 100% flow and the turbine driven pump at 64%. The water was not pumped into the steam drums because the drum pressure during plant operation is too high to achieve full flow, it is not desirable to thermally shock the AFW sparger with cold feedwater on a monthly basis, and it is inadvisable to pump the more impure PWST water into the SGS on regular basis. Redundant actuated valves were added to automatically isolate the AFW supply from the PWST should SGAHRS be initiated during testing.

8.2.11 Addition of Steam Vent Noise Reduction Devices (ECP L10-158)

Upon SGAHRS initiation, steam is vented to the atmosphere through the superheater and/or steam drum vent valves. This venting, which may last

intermittently for approximately one hour, is required to release the initially high heat load from the heat transport system, and to reduce the steam drum pressure for proper SGAHRS operation. Line velocities would reach sonic at the exit causing excessive noise levels at the discharge points on the building roof. Additionally, the shock wave generated at sonic velocities would cause severe damage to buildings and structures. Noise level limits at the site boundary were specified in the Oak Ridge zoning ordinance and HUD "24CFR51 Environmental Criteria and Standards."

To meet the noise level limits, exhaust restrictors were added at the vent discharges. The restrictor was designed to provide a large discharge area allowing low velocity release and correspondingly reduced sound levels, which met site requirements. In addition, because the exhaust restrictor can accommodate a relatively high pressure drop, the vent valve pressure drop can be reduced, resulting in lower noise level contribution in the building.

8.2.12 System Revisions to Accommodate Replacement of Diverse Power Supply with Diesel Generator (ECPs B10-159 and G1135)

This topic was one of the key problems encountered in the SGAHRS design. Section 8.3.1 describes the system design changes resulting from the replacement of the diverse power supply (250 VDC battery) with a diesel generator.

8.2.13 Reduction in Requirements for Actuated Isolation Valves (ECP G1135)

So that the operator could conveniently fill the PWST or change the AFW water source from the PWST to the condensate storage tank, valves used for these operations were remotely operable. Since all of the valves upstream of the AFW pump inlets (1) did not require automatic operation, (2) did not serve an active safety function and (3) were to be used infrequently, they were changed to be manually operable.

8.3 Key Problems and Resolutions

8.3.1 Shutdown Heat Removal with Loss of AC Power and with Limited DC Power

In the reference design, standby on-site power was to be provided by two (Divisions 1 and 2) Class 1E diesel generators and one (Division 3) Class 1E diverse power supply (250 VDC battery), capable of providing a relatively large amount of power. However, the Division 3 diverse power supply inverters received power from either the Division 1 or Division 2 power distribution system. During interactions with the NRC, a strong preference toward complete independence between the Class 1E divisions was indicated. Based on this, the diverse power supply was replaced by a third diesel generator, with limited 120 VDC non-interruptible battery supply (ECP B10-159). For the reference plant design, SGAHRS was designed to accommodate a two hour loss of both on-site and off-site AC power supplies. SGAHRS components which were required to perform the shutdown heat removal function were either powered by the diverse power supply or were designed to revert to the "safe" position. As shown in the SGAHRS power division assignment diagram [5], Division 3 power was distributed throughout SGAHRS to allow controlled heat removal from two steam generator system loops, using the turbine driven pump to supply the auxiliary feedwater (AFW). With the replacement of diverse power supply with the diesel generator, Division 3 power would not be available to SGAHRS components during the loss of bulk (on-site and off-site) AC power event.

PROBLEM RESOLUTION

When all of the AC power is lost, the AFW has to be supplied to the steam drum by the turbine driven pump and energy released from the SGS through the PACC and the superheater and/or steam drum vent control valves. To assure that SGAHRS would adequately remove reactor decay heat and heat transport system sensible heat during the two hour loss of bulk AC event, the following changes were made (ECPs B10-159 and G1135).

- A. For the valves delineated below, the actuators were changed to operate using a 105-135 VDC UPS supply, instead of 120 VAC or 480 VAC.

- (1) AFW control valves on turbine driven pump AFW supply header
 - (2) All SGAHRS superheater and steam drum vent valves
 - (3) All PACC non-condensable valves
 - (4) Turbine pressure control valve
- B. Turbine governor control changed to operate using 105-135 VDC UPS supply, instead of 250 VDC.
- C. PACC inlet louvers changed to fail open, instead of fail closed.

Making the changes assured the following functions were maintained: (a) control of the turbine drive speed; (b) control of the auxiliary feedwater flow; (c) control of the steam venting to assure the steam drum pressure drops sufficiently to allow the required feedwater flow; (d) venting of the non-condensable gases in PACC to assure that the PACC heat rejection capability is not reduced and (e) the natural circulation of air through the PACC.

A change to DC supplied louver actuator was not made since PACC heat rejection rate control would not be required during the two hour period.

8.3.2 Loads on Steam Supply Line to Auxiliary Feedwater (AFW) Pump Turbine Drive

In SGAHRS, three pumps are used to supply AFW from the PWST to the SGS steam drums. Two of the pumps are electric motor driven and the third is driven by a turbine, which in turn is driven with steam from the drums. Figure 8-3 provides a schematic of the steam supply lines to the turbine. During normal plant operation, electro-hydraulically operated isolation valves V1, V2 and V3 are closed and the majority of the lines down steam of the valves are at ambient conditions. Upon SGAHRS initiation, these isolation valves are automatically opened within three seconds and saturated steam from the drums enter the lines. Because the lines are relatively cold, some of the steam will condense in the lines and a high velocity steam/water mixture will

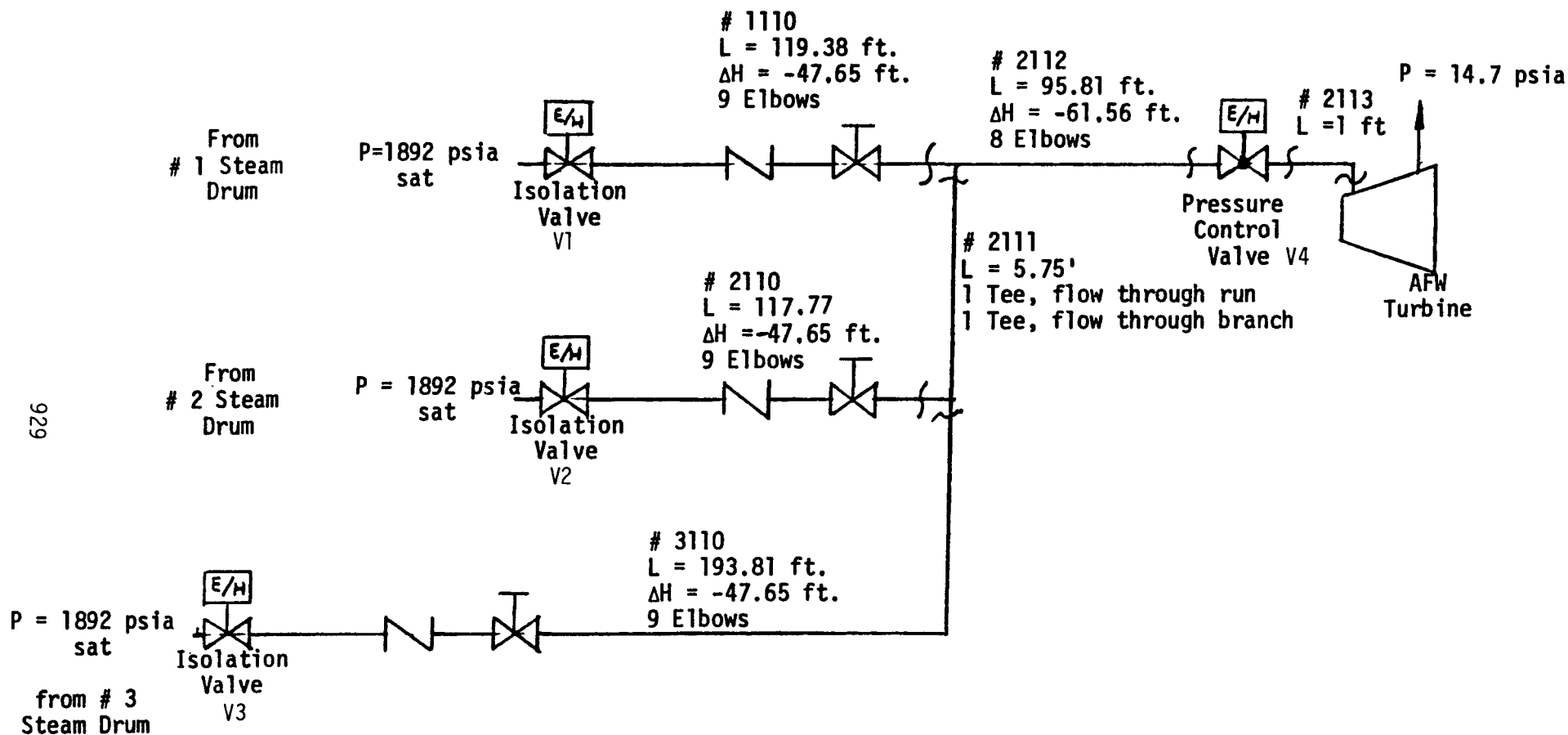


Figure 8-3
STEAM SUPPLY LINE SCHEMATIC

result. Due to previous experience of piping support failures under similar conditions (steam released into cold piping), a concern was raised on the adequacy of the supply piping and supports.

PROBLEM RESOLUTION

To assure the adequacy of the piping system, either of the following paths could have been taken.

- A. Preheat the steam supply lines to the pressure control valve, V4.
Design the piping system to accommodate the steam dynamic loads.
- B. Design the piping system to accommodate the steam/water mixture dynamic loads.

In evaluating (A) above, three methods of heating the steam supply line were considered, namely: steam heating, electrical trace heating and trace heating using auxiliary steam [6]. Table 8-1 provides a comparison of the advantages and disadvantages of the three methods. The key requirement in selecting the heating method is that the heating method must be safety related, since the operation of the turbine driven pump serves the safety function of shutdown heat removal. This requirement would rule out both steam and electrical trace heating for the following reasons. Steam trace heating would require steam from the auxiliary steam supply portion of System 72, a non-safety related system which could not be upgraded to be safety related. All electrical trace heaters (System 68) used in CRBRP were non-Class 1E. The use of similar trace heaters for 1E application would create problems of 1E qualification, of physical separation of electrical power divisions (heaters supplied by Division 3, attached to piping located in loops supplied by Divisions 1 and 2), and of additional 1E-UPS electrical loads. The approach of electrical trace heating is thus undesirable even though it may have been possible to make it safety related.

This left steam heating as the most plausible heating method. This method would bleed steam from the steam drums to fill the steam supply line and keep the line heated. The steam would be bled through an additional valve,

TABLE 8-1
AFW TURBINE DRIVE STEAM SUPPLY LINE HEATING
AN EVALUATION OF ALTERNATIVE HEATING METHODS

<u>Heating Method</u>	<u>Advantages</u>	<u>Disadvantages</u>
1. Steam Heating with a bypass valve to bypass steam supply isolation valve	- o Steam from SGS steam drums is used for heating. No additional heat source is required. - o Can be made safety related. - o Minimal hardware changes to existing piping design. Need to add a bypass valve and associated piping for each loop. Another actuated isolation valve may also be required for each line from the steam drums to isolate the break.	- o Piping must be reclassified as high energy line. Need to postulate a line break and to evaluate cell temperature and pressure responses. - o Installation of humidity monitors and temperature sensors may be required in the SGAHRS cells to automatically isolate the break. - o May have impact on PWST inventory during SGAHRS operation.
2. Trace Heating	- o No change in piping system design is required. - o No need to postulate a line break.	- o Non-safety related. - o Need additional power supply to provide the heat.
3. Steam Trace	- o Auxiliary steam (System 72) is used. No impact on PWST inventory. - o No need to postulate a break.	- o Non-safety related. - o Pipe can only be heated to the auxiliary steam temperature of approximately 365°F.

approximately 1/4-inch size, around each of the steam supply isolation valves V2 and V3. The condensed steam would be drained through a steam trap near the pressure control valve, V4. Two undesirable consequences of steam heating are the following.

1. The steam supply lines from isolation valves V1, V2 and V3 to the pressure control valve become classified as high energy lines. As such, a double-ended break must be postulated. To mitigate the consequences of the break, automatic action to isolate the break would be required. This would involve the installation of humidity monitors and temperature sensors in the cell housing the drive turbine and in the Loops 1 and 2 pipe chases. To assure the isolation of the break in case of a single failure of an active component, an additional actuated valve would be required in each line from the steam drum. Guiding modification to accommodate the resulting pressures and temperatures would be required.
2. Another complication of steam heating arises when the turbine-driven AFW pumps is used in the long-term cooldown mode during the SGAHRS operation. The steam required for heating (and drained through the steam traps) has to be made up from the PWST inventory. The PWST design could meet (by utilizing the margin) the additional inventory requirement by limiting the heat loss through the insulated steam supply line to 90 Btu/hr/ft.

In evaluating (B) above, Burns and Roe performed an analysis [7] of the effects of steam/water hammer on the subject line. A fluid force time history analysis using the RELAP 4 computer program was performed. The RELAP 4 analysis included the following effects.

- o Friction losses in the pipes
- o Steam condensation
- o Expansion and reflective waves
- o Two-phase flow
- o Temperature dependent fluid density

The resultant forces were found to be small and did not add significantly to the stress levels in the piping.

Since the analytical results did not seem to agree with experience; an evaluation of the past failures and successes was made. For those cases with failures, design modifications to assure heating were made; however, there was no clear evidence that the failed piping system had originally been analyzed for dynamic fluid loads and that an adequate supporting system was installed. For those cases without reported failures, such as on the discharge piping for relief valves, analysis which account for dynamic loads had been properly considered.

Therefore, it was decided that because of the adverse modifications required to accommodate heating of the steam supply line and because it could be shown that properly analyzed piping can withstand the loads due to steam/water mixture hammer, path (B) above was chosen to resolve the concern.

8.3.3 PACC Cell Environment

During normal plant operation, the PACCs are maintained in a hot standby condition by natural circulation of steam from the SGS steam drums. Although the PACC plenum was insulated, the PACC cell would be subjected to a heat load of approximately 100,000 Btu/hr [8]. Since the PACC cell had openings to the atmosphere which could not be closed (this feature was desirable since controlled openings would have to be Class 1E to assure operation), no air conditioning or ventilating systems were provided. These conditions resulted in cell and local temperatures exceeding the limits of safety related components within the cell [9] [10]. During PACC operation, this was not a problem since the cell was cooled by the outside air drawn in by the PACC fans or drawn in by the natural circulation of air through the PACC.

PROBLEM RESOLUTION

The following options were considered as possible single solutions to the PACC cell temperature problem.

- A. Increase the thickness of the PACC plenum insulation.

Because of space limitations, especially at the top of the PACC, and because of the structural changes required to account for the increase in PACC weight (insulation and plenum to support insulation), this option was found to be undesirable.

- B. Provide Class-IE active components which could withstand long-term high cell temperatures.

Although they could withstand short-term high temperatures (180°F), the actuators for the PACC louvers and fan blade pitch control had a temperature limit of 104°F. Since the PACC cell would exceed this limit without cooling, this option was discarded.

- C. Provide air conditioning in the cell.

To increase the efficiency of the air conditioning system, closure devices would have to be provided on the PACC cell air inlet openings. To assure the safety related operation of the PACC, these devices would have to be class-IE active components. Because this option would cost more than the forced ventilation system, it was discarded.

- D. Provide forced ventilation through the cell.

- E. Provide natural draft capability in the cell.

At the time of project termination, the adequacy of natural air draft to sufficiently cool the entire PACC cell had not been demonstrated. However, the following design actions were in process to resolve the problem [11].

1. The PACC air supply inlets, at the building (not the PACC inlet) were moved from the south wall to the roof of the steam generator building (SGB). By locating the openings at the cell high point, natural draft of air, from a lower elevation cell, thru the PACC cell was enhanced.
2. The PACC cell was divided by a full height, full width wall at the PACC fan inlets. This would ensure the PACC safety related equipment located outside the plenum would not be subjected to more than a non-severe aerosol environment.
3. Cooling (by forced or natural air movement) of local hot spots, which were to be determined.
4. Possible forced ventilation in the PACC cell east of the wall described in (2) above. If forced ventilation were required, temperature sensor/alarms would have been added to the PACC cell to alert the operator of cooling malfunction. The operator could open the PACC louvers to promote cell air circulation.

8.3.4 Steam Drum Depressurization During SGAHRS Operation

Upon SGAHRS initiation, the following automatic actions occurred:

- o Steam is released from the SGS through vent valves located at the superheater outlet and/or steam drum. The venting is required to lower (and maintain) the steam pressure to a level compatible with the AFW pump head.
- o Cell temperature AFW is supplied from the PWST to the steam drums, through a sparger elevated above the drum water level. The AFW was added at the maximum possible rate to recover the steam drum water level (to make up the steam vented).

- o The PACC is initiated for maximum air flow and, correspondingly, maximum heat rejection. The limit on the PACC heat rejection rate change was 1.5 Mwt (10% of rated) every 200 seconds.
- o Steam (80,000 lb/hr) is supplied to the AFW pump turbine drive from the steam drums. The pump is shut down by the operator after he has confirmed the proper operation of the system and, specifically, the proper operation of the motor driven pumps. This may be 20-30 minutes after SGAHRS initiation.

When the plant response was analyzed, using DEMO-4 and CANDY computer codes, for the loss of off-site power (U-18) and three-loop natural circulation (beyond-design-basis loss of bulk AC power event) events, severe steam depressurization and system oscillations were noted.

PROBLEM RESOLUTION

In the evaluation of the stated problem, it was found that all of above automatic actions contributed to the problem. To assure that the steam drum pressure was maintained within acceptable limits (>1400 psia) and to minimize system oscillations (steam venting, AFW), the following changes to the control system were made.

- A. The PACC air flow was revised to be controlled continuously using the steam drum pressure set point. The control system design for the PACC fan blade pitch, for forced air circulation, had been completed [12]. Design of the inlet louver control system, for air flow control under natural air circulation conditions, had not been completed at the time of project termination.
- B. Instead of always injecting AFW at the maximum rate, a flow limit based upon steam drum pressure was added [12] [13]. This minimized the amount of steam quenching caused by the injection of cold AFW into the drum. This also minimized AFW oscillations early in the SGAHRS initiation events.

- C. Control logic to automatically shutdown the turbine-driven pump was added, as shown on Figures 8-4 and -5 [12]. Under normal SGAHRS operation, steam to the turbine drive would be stopped 60 seconds after SGAHRS initiation if, (1) the motor-driven pumps were maintaining the water level sufficiently, or (2) venting had stopped and the steam drum water level had recovered. This feature would prevent plant cooldown by SGAHRS below the hot standby conditions (600°F) after the PACC had been shutdown. To accommodate the loss of bulk AC power event, control logic for isolating/opening the steam supply using the DC-power supplied turbine drive inlet pressure control valve was added.
- D. To minimize the cycling of the vent valves and correspondingly, the AFW valves, the superheater vent valve controller was made slower than the AFW valve [13]. This was accomplished by increasing the vent valve gain and reducing its time constant [12].

8.4 Final System Status

8.4.1 Status of Design/Analysis

The system design at the time of CRBRP termination is reflected in SDD-52, Revision 84 [3]. All six sections, including appendices, of the SDD had been baselined. Except for vendor documents, all required system design documentation (i.e., P&ID, hydraulic profile, interface documents, equipment specifications) had been baselined. Preoperational test specifications [14-17] and construction test requirements document [18] were completed. A reliability evaluation of SGAHRS was performed and the results were reflected in the SGAHRS reliability design support document [19]. Reliability assessments [20,21] were also made on the CRBRP shutdown heat removal system, of which SGAHRS is a major part. System maintenance platform requirements were defined [22] and baselined in Westinghouse document CL54068.

The final working system design review (FWSDR) for SGAHRS was conducted in May 1982 [23]. All FWSDR comments were evaluated and dispositioned for appropriate action [24-27]. Design verification through Revision 72 of SDD-52

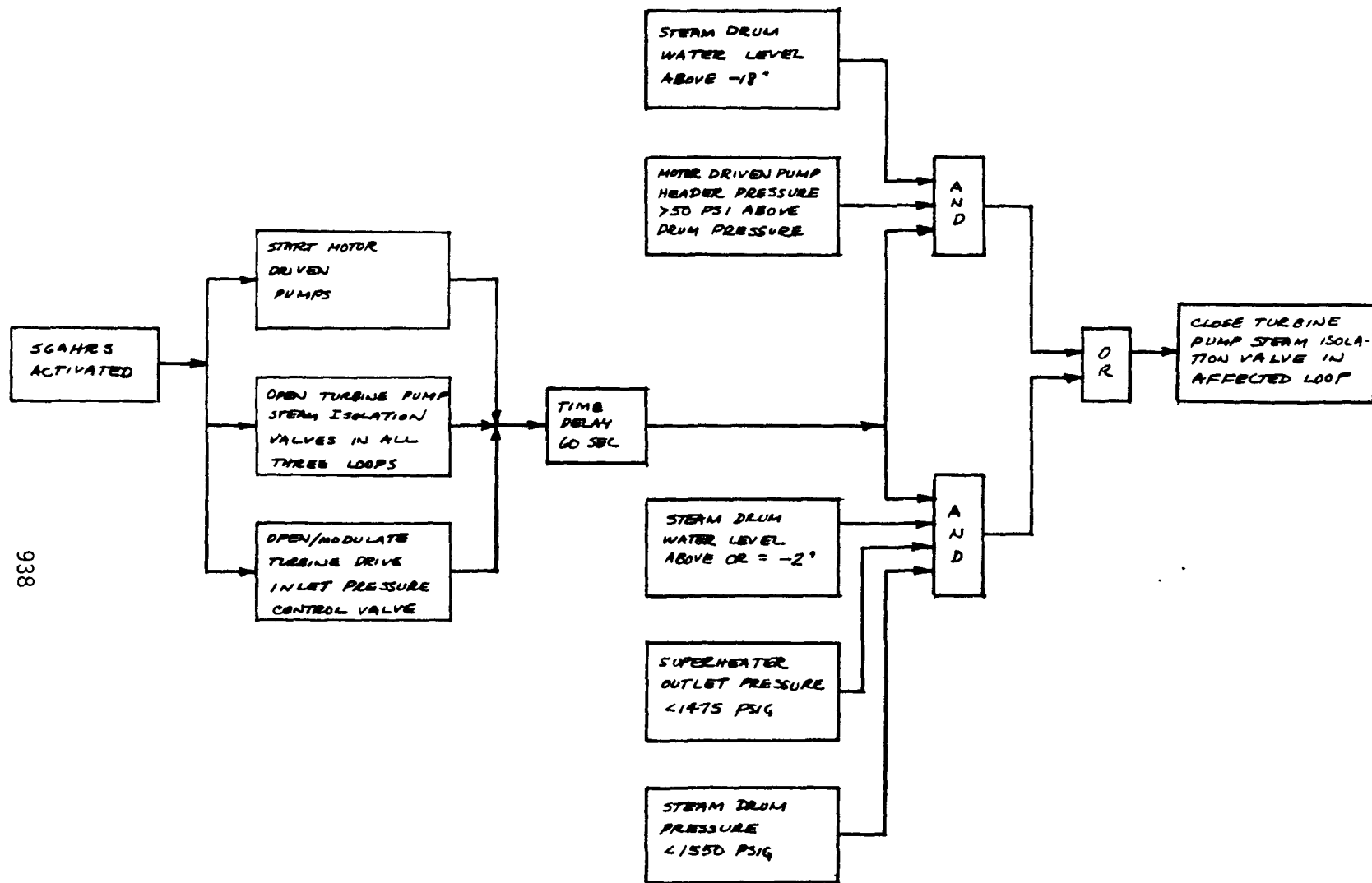


Figure 8-4
CONTROL LOGIC FOR TURBINE DRIVEN AFWP

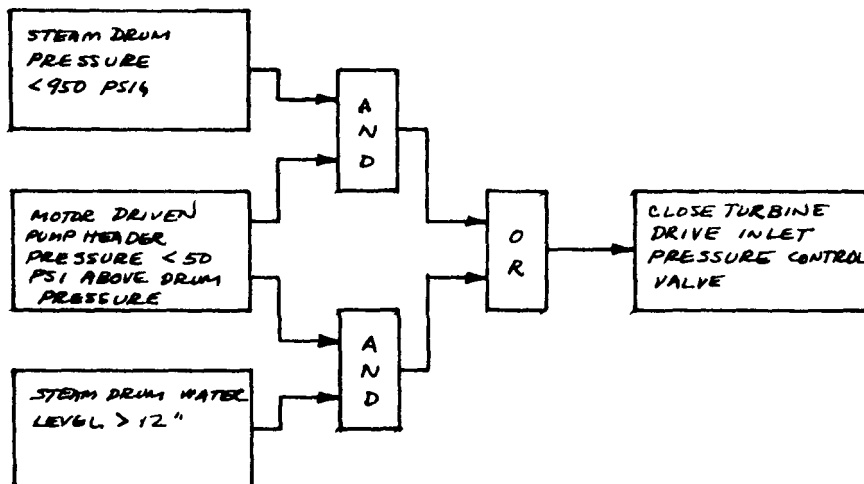
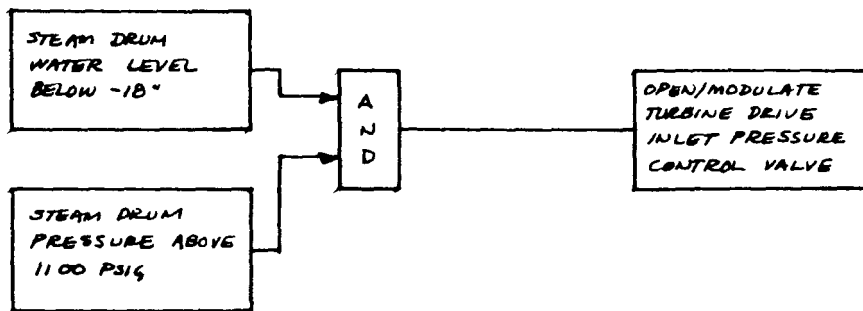
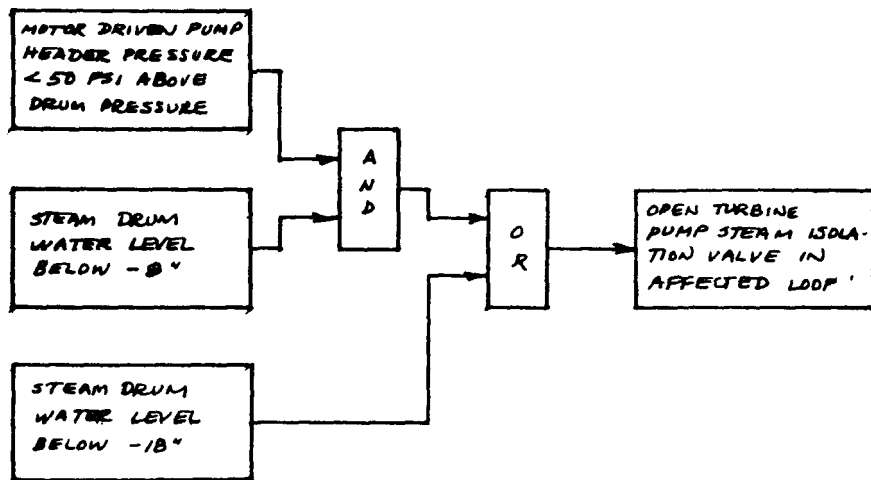


Figure 8-5
CONTROL LOGIC FOR TURBINE DRIVEN AFWP

was completed [28]. An independent review of the SGAHRS design verification was completed and documented in a report [29]. Design reviews were conducted and all resulting concerns were resolved for the following components. AFW pumps and motor drives, AFW pump turbine drive, control valves, actuated isolation valves, check valves, manual valves and protected air cooled condenser [30-36]. The PWST was reviewed using the design verification process.

The DAHRS computer code, which was used to evaluate the performance of SGAHRS under long-term operational modes, was documented and verified [37]. The effects of various events on PWST water usage was analyzed [38]. The CANDY computer code, which performed short-term dynamic analysis of the IHTS and steam/water systems of Clinch River, was being documented and verified [39] when the project was terminated. The CANDY code was used primarily for determining the control requirements on SGAHRS pumps, valves and the PACCS.

The analyses of the SGAHRS superheater and steam drum vent valves piping (Drawing 852E520) and PACC piping (Drawing 852E530) were performed as part of the SGS piping analysis since all the subject piping was closely coupled. Expansion joints had been added on the vent valve lines to assure stress limits were met. This was necessary since the exhaust restrictors located at the line discharges had to be secured to the building roof. The analyses of the remaining SGAHRS piping (Drawings NN1520 through NN1525) were performed by Burns & Roe. Preliminary analyses of all the piping had been completed.

SGAHRS components that had been completed and delivered were: protected water storage tank; AFW pumps and electric drive motors; check valves; and instrument, vent and drain valves. At the time of Project termination,

- o The control valves were in the final stages of assembly, with one earlier assembled valve in qualification testing.
- o The actuated isolation valve designs were nearing completion, with most drawings completed.

- o The design phase of the manual valves was completed, and material procurement and valve fabrication were initiated.
- o Design layouts for the PACC were completed.
- o Design documents were completed to support the procurement of the PACCs, flowmeters and exhaust restrictors.

Following Project termination the AFW pump turbine drive performance tests were completed and the drive was shipped to a storage site in Oregon.

8.4.2 Outstanding Activities to Complete Design

The principal remaining work on System 52 design was in the following areas.

- A. Completing the PACC inlet louver control design to allow air flow control under natural circulation operation conditions. Modeling the design into CANDY to verify its proper operation.
- B. Completing verification of the CANDY computer Code.
- C. Reanalyzing the AFW piping to reflect the reduction/revision to the AFW supply valves (Reference ECP G1135).
- D. Determining the setpoint of the temperature differential, between the steam drum steam and PACC noncondensable collection pipe fluid, used to activate PACC venting.
- E. Determining the discharge vortex breaker design on the PWST to mitigate air entrainment in the AFW supply.

The following engineering change proposals (ECPs) and other defined changes, which affected System 52 system or component design, were in process at the time of project termination.

- o G1449 Revised the PACC thermocouple locations to reduce the number required, and deleted the thermocouple safety function.
- o G1393 Provided exhaust restrictor mass and stiffness requirements for use in seismic analysis.
- o L10-162 Specified applicable USNRC Regulatory Guides and Branch Technical Positions.
- o L10-214 Revised the environmental conditions for SGAHRS components.
- o L10-213 Added specific surveillance requirements for the plant.
- o B45-017 Established storage space interface requirements.
- o B69-059 Established the Plant Service Building Combined Laboratory as facility to analyze PWST water quality.
- o B76-051 Specified SGAHRS waste oil disposal requirements.
- o B10-231 Established electrical, instrumentation and control criteria for plant, which affected equipment supplied by vendors.
- o Update the System 52 requirements on the Plant Data Handling and Display System to show the necessary data [40].
- o Update the control requirements for the PACC, pump turbine drive, vent valves and AFW pumps to resolve the problem of steam drum depressurization [12].
- o G1420 Specified larger valves and valve snubber interface requirements.
- o G1496 Specified preferred handwheel orientation for System 52 manual valves.

The following approved ECPs had not been incorporated or were only partially incorporated at the time of project termination. G1135, G1472, G1489, G1495, G1505, L10-125 and L10-134.

8.5 References

1. Clinch River Breeder Reactor Plant Reference Design Report, Volume 1, June 1974.
2. NEDC-14013, Steam Generator Auxiliary Heat Removal System Concept Selection Study, April 1974.
3. System Design Description SDD-52, Steam Generator Auxiliary Heat Removal System, Revision 84, October 19, 1983.
4. Westinghouse Letter WG50358, "CRBRP; Steam Generator Auxiliary Heat Removal System," August 25, 1975.
5. GE Drawing 13401525, SGAHRS Power Division Assignment, Revision 0.
6. GE Letter XL-590-20487, "CRBRP; Heating of System 52 Steam Supply Line to Auxiliary Feedwater Pump Turbine Drive," December 1, 1982.
7. Burns & Roe Letter BD30169, "System 52; Steam, Water Hammer Analysis for the Auxiliary Feedwater (AFW) Pump Turbine," February 18, 1983.
8. GE Letter YL-600-82099, "PACC Plenum Heat Loss," August 8, 1982.
9. GE Letter XL-590-20262, "CRBRP; Transmittal of PACC Cell Heat Load/Temperature Limitations," June 11, 1982.
10. GE Letter XL-497-20621, "CRBRP; ECP G1023, Rev. 3, PACC Cell Cooling and System 25A Interface," October 15, 1982.
11. Westinghouse Letter LR830555, "CRBRP; Resolution of Technical Issues for the Protected Air Cooled Condenser," October 10, 1983.
12. GE Letter XL-590-830574, "CRBRP; Control Requirements for SGAHRS, System 52," November 11, 1983.
13. GE Letter YL-612-830186, "Cycling of Superheater Vent Valve Flow During a U-18," October 19, 1983.
14. 23A3016, CRBRP Preoperational Test Specification - SGAHRS Functional Test at Ambient Conditions (T-52-1B202), Revision D1, March 25, 1983.
15. 23A3017, CRBRP Preoperational Test Specification - SGAHRS Functional Test at Low Power (T-52-3B014), March 25, 1983.
16. 23A3018, CRBRP Preoperational Test Specification - SGAHRS Readiness Checks (T-52-1B211), March 25, 1983.
17. 23A3019, CRBRP Preoperational Test Specification SGAHRS Piping and Component Dimensional Survey at Ambient Conditions (T-52-1B052), March 25, 1983.

18. GE Letter XL-590-830062, "CRBRP: System 52, SGAHRS Construction Test Requirements," February 22, 1983.
19. GE Letter XL-599-830101, "CRBRP; SGAHRS Reliability Design Support Document (RDSD), Revision 3, September 16, 1983.
20. CRBRP-GEFR-00552, A Reliability Assessment of the CRBRP Shutdown Heat Removal System, June 1983.
21. CRBRP-GEFR-00554, Reliability Data for the CRBRP Shutdown Heat Removal System, June 1983.
22. GE Letter XL-590-830127, "CRBRP; System 52 Maintenance Platform Requirements," March 24, 1983.
23. DOE Letter CR:742:RE:82-243, "FWSDR on the SGAHRS," August 1982.
24. Westinghouse Letter LR820572, "SGAHRS (System 52) FWSDR," September 3, 1982.
25. Westinghouse Letter LR820702, "SGAHRS, Sys. 52 FWSDR Comments, Resolution of," November 3, 1983.
26. Westinghouse Letter LR830226, "SGAHRS, Sys. 52 FWSDR Comments, Resolution of," April 18, 1983.
27. Westinghouse Letter LR830390, "SGAHRS, Sys. 52 FWSDR Comments, Resolution of," July 18, 1983.
28. GE Letter XL-590-830143, "CRBRP; SGAHRS 52 Design Verification Report," March 30, 1983.
29. GE Letter XL-590-830174, "CRBRP; SGAHRS 52 Design Verification Review Report," April 22, 1983.
30. GE Letter XL-590-10341, "CRBRP; System 52 Auxiliary Feedwater Pump and Motor Drive Final Design Review," August 28, 1981.
31. GE Letter YL-792-10333, "Systems 52 and 53 Control Valves Preliminary Design Review - Resolution of Review Concerns," June 26, 1981.
32. GE Letter XL-590-20177, "Resolution of Design Team Concerns SGAHRS Auxiliary Feedwater Turbine," April 9, 1982.
33. GE Letter XL-590-830576, "CRBRP System 52/53; Responses to Concerns Resulting from July 13/14, 1983 Internal Preliminary Design Review for the Actuated Isolation Valves," November 1, 1983.
34. GE Letter XL-590-20166, "CRBRP; Closeout of System 52/53 Check Valve Design Review Comments," April 5, 1982.
35. GE Letter XL-590-830058, "System 52 PACC Resolution of Design Review Concerns," February 14, 1983.

36. GE Letter XL-590-20313, "CRBRP; Closeout of System 52/53 Manual Valve Design Review Comments," July 21, 1982.
37. 23A2451, CRBRP Specification: DAHRS Computer Code, June 3, 1983.
38. GE Letter YL-612-830101, "Record DAHRS Runs for PWST Water Storage and Usage," June 27, 1983.
39. GE Letter YL-601-830160, "CANDY Code Verification Summary," November 16, 1983.
40. GE Letter XL-590-830515, "CRBRP; Update of System 52 Interface Requirements on PDH&DS (ICD C652637)," September 22, 1983.

8.6 Major Engineering Change Proposals (ECPs) Affecting Design

The following provides a list of ECPs which had a major affect on the SGAHRS design:

- G0032 Modification of System 52 to Achieve Compliance with NRC BTPs
 APCSB 3-1 and 10-1, October 15, 1975.
- G0192 Accommodation of AFW Flow Limiter Failure, System 52,
 April 20, 1977.
- G0198 Accommodation of Pipe Break in System 52 AFW Subsystem,
 May 9, 1977.
- G0313 System 52 Instrumentation and Control Requirements, June 2, 1978.
- G0318 System 52 SGAHRS Initiation Logic, May 30, 1978.
- L10-054 PACCs Initiation on all Scrams, July 25, 1977.
- G0389 System 52--Eliminate Pneumatic Valve Actuators, May 10, 1978.
- A26009 Aerosol Release Mitigation (SGB).
- G0627 SGAHRS Periodic Testing, August 3, 1981.
- G0418 Revision of System 53 Requirements for Operation at Hot Standby,
 Refueling, and Wet Layup (Water chemistry requirements), April
 3, 1980.
- L10-158 Applicable Noise Level Requirements for the Area Surrounding the
 Plant, OPDD-10, June 1981.
- G1088 System 52 Modification to PACC Piping, July 20, 1982.

B10-159 Addition of Third Class 1E Diesel Generator, May 12, 1982.

G1135 SGAHRS 52: Update Electrical Requirements, January 8, 1983.

G1253 PACC Embedment Definition, July 7, 1982.

9.0 STEAM GENERATOR SYSTEM

The Steam Generator System (SGS) receives reactor generated heat from the Intermediate Heat Transport System (IHTS) and produces superheated steam for electrical power production by the plant turbine-generator system. The SGS is comprised of three independent and identical steam generation loops, each capable of utilizing 325 MWt at full power to convert subcooled water to superheated steam.

9.1 Summary of Functions and Design Description

9.1.1 Functions

The functions of the SGS are to.

- A. Convert subcooled water into superheated steam by transferring heat to the water and steam from the IHTS sodium.
- B. Provide superheated steam at the temperature, pressure, and flow rate required by the turbine over the 40 to 100 percent load-following range.* In addition, provide appropriate steam flows for turbine warm-up and initial turbine loading to 40%.
- C. Cool the IHTS sodium to the temperature levels required for safe reactor cooling during transient and steady state conditions, and be capable of cooling the IHTS down to the refueling temperature of 400°F.
- D. Provide a high integrity pressure boundary to assure separation between the sodium and water/steam in the steam generators.

*Turbine inlet requirements for 100% load are 3.3 million pounds per hour superheated steam at 900°F and 1450 psig for the total SGS. Steam pressure and temperatures from the superheaters are slightly higher to account for losses between superheaters and turbine.

- E. Detect water-to-sodium and steam-to-sodium leaks in the Steam Generator modules and, to the maximum extent possible, identify the module in which the leak has occurred and the approximate size of the leak.
- F. Prevent the pressure generated by a sodium-water reaction (SWR) from reaching a value which would cause damage to the Intermediate Heat Exchanger (IHX).
- G. Provide rapid blowdown of the steam/water side of the steam generators to mitigate the consequences of a large or intermediate SWR and provide the capability of gas backfilling to prevent backflow of sodium into the water/steam side.
- H. Provide storage for the IHTS sodium which may be contaminated with SWR products following an SWR in a steam generator module.
- I. Control the solid, liquid, and gaseous products of a large SWR so that the solids and liquids are contained in appropriate vessels and the gas (hydrogen) is released to the atmosphere in a safe manner.
- J. Prevent the steam/water side pressure from exceeding a safe value.
- K. Maintain separation of heat transfer loop functions so that a failure in or shutdown of one loop does not prevent operation of the remaining loops.
- L. Provide saturated steam to the Steam Generator Auxiliary Heat Removal System (SGAHRs) Protected Air-Cooled Condenser (PACC) loop when the SGAHRs is in the long-term operating mode.
- M. Provide saturated steam to the SGAHRs turbine driven auxiliary feed pump.
- N. Provide storage for the sodium inventory of each loop to accommodate loop filling and draining.

- O. Provide a sufficient water inventory to allow continued cooling of intermediate sodium for 30 seconds until SGAHRS is started after a loss of feedwater flow.
- P. Provide the capability to isolate the three SGS loops from the Balance of Plant (BOP) in case of a major steam or feedwater leak. Isolation must be accomplished with sufficient speed to insure continued water inventory in the steam generator system for removal of decay heat.

9.1.2 Design Description

The SGS provides independent steam generation capability for each of the three Reactor Heat Transport Systems. As shown on Figure 9-1 each independent SGS is comprised of the following subsystems:

- o Water/Steam Subsystem
- o Sodium/Water Reaction Pressure Relief Subsystem
- o Leak Detection Subsystem
- o Sodium Dump Subsystem
- o Water Dump Subsystem

The Water/Steam Subsystem (WSS) obtains feedwater from the feedwater system. The feedwater enters the steam drum where it is internally mixed to subcool the saturated water from two parallel evaporators. The subcooled water is then circulated by the recirculation pump from the drum to the evaporators. In the evaporator tubes the subcooled water is preheated and partially vaporized by sodium flowing counter-current on the shell side. At 100% reactor power the fluid leaving the evaporator is 50% by weight steam. The two phase fluid flows from the evaporator to the steam drum where separators, internal to the drum, separate the water and steam. A fraction of the saturated water (before the saturated water is mixed with the incoming feedwater) is then drained from the drum and returned to the feedwater and condensate system. The saturated steam then flows through dryers, internal to the drum, to the superheater. In the superheater tubes the saturated steam is

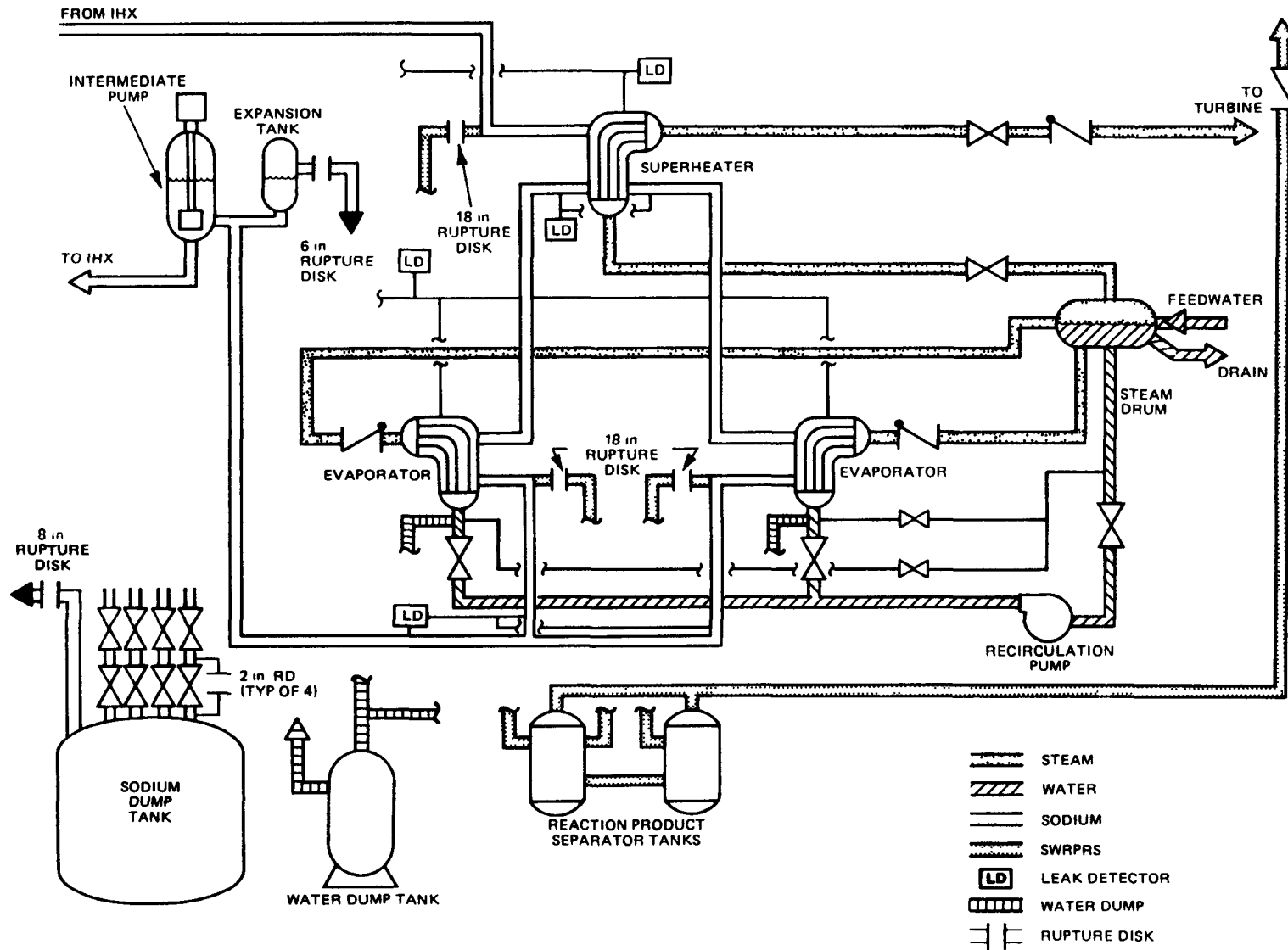


Figure 9-1 CRBRP STEAM GENERATOR SYSTEM SCHEMATIC

superheated to the desired temperature by sodium flowing counter-current on the shell side. The superheated steam exiting from the individual superheaters flows to main steam headers and then to the turbine.

Purity of the recirculation water is maintained by a continuous drain flow from the steam drum through isolation valves to the flash tank in the feedwater/condensate system. Water from the feedwater line and from the drum drain line is transported through sample lines to a monitoring station for measurement of specific feedwater and recirculation water impurities.

Combination safety and power relief valves are provided in the evaporator outlet lines and superheater outlet lines for overpressure protection. Also safety relief valves are provided on the steam drum and on the evaporator outlet lines for overpressure protection. Isolation valves and water dump valves are provided for the evaporators for isolation and removal of the water to mitigate the effects of an SWR from a tube leak in these units. In the event of an SWR in a superheater the steam would be discharged through the power relief/safety valves to the atmosphere.

"Hockey Stick" type tube-in-shell units are employed for the CRBRP superheaters and evaporators. The units contain single wall, 2-1/4 Cr-1 Mo tubes welded to tubesheets by internal bore butt welds.

The Sodium/Water Reaction Pressure Relief Subsystem (SWRPRS) is a passive system which only becomes operational in the event of a large SWR within an evaporator or superheater module. For this event the SWRPRS protects the sodium side of the evaporator and superheater modules, the IHTS and the IHX from overpressure by the use of duplex, reverse buckling rupture disks on the superheater inlet line and evaporator outlet lines. In the event of a large SWR, sodium and SWR products expelled through the rupture disks are directed by SWRPRS piping to reaction products separator tanks (RPSTs) where separation of liquid, solid and gaseous products takes place. The gaseous reaction products are then vented via a flare stack to the atmosphere. Liquid and solid SWR products would be stored in the RPSTs.

A check valve and an atmospheric seal are located upstream of the stack outlet. The check valve prevents backflow of air into the SWRPRS after activation of the SWRPRS and the atmospheric seal (essentially a low pressure rupture disk) normally seals the inert gas in the SWRPRS from the atmosphere. In addition, nitrogen gas is admitted to the SWRPRS for purging purposes. An igniter is located downstream of the atmospheric seal and the check valve to ignite any hydrogen gas exiting from the SWRPRS piping.

Intermediate size steam or water leaks, up to approximately two pounds per second can be accommodated without failure of the SWRPRS main rupture disks by relieving IHTS system pressure at the expansion tank through relief of the expansion tank/dump tank vent line rupture disk. These rupture disks also initiate water side isolation and blowdown of the steam generator modules in the affected loop.

The Steam Generator Leak Detection Subsystem monitors the sodium leaving each evaporator and superheater, and the sodium leaving the vent line from each evaporator and superheater. The leak detection subsystem would detect small steam generator tube leaks by measurement of hydrogen and oxygen concentration in the liquid sodium stream. Hydrogen measurement is accomplished by allowing hydrogen to diffuse through a thin nickel membrane to high vacuum held by an ion pump. An electrochemical cell is used to monitor in-sodium oxygen concentration.

The Sodium Dump Subsystem provides storage of IHTS sodium for initial fill & storage of IHTS sodium after a rapid dump (e.g., following an SWR in the steam generator modules) or after normal drain of the IHTS loop.

The subsystem consists of a sodium dump tank, located on the lowest building level beneath the evaporator and superheater modules, and a rupture disk which provides overpressure protection. The discharge from the rupture disk assembly is piped into the SWRPRS main vent line.

The Water Dump Subsystem is provided for each SGS to accept and store the water from the evaporator modules when rapid depressurization of the evaporator modules is required. The water dump valves discharge to the water

dump piping which directs the water/steam mixture to a water dump tank where the water is temporarily stored, and the flashed steam is vented to the atmosphere.

Carbon steel is employed for the steam drum, recirculation pump, recirculation water system piping and SWRPRS. Low alloy steel (1-1/4 Cr-1/2 Mo) is used for superheater steam piping. 2-1/4 Cr-1 Mo steel is used for superheaters and evaporators and 304H stainless steel is used for leak detection system piping.

All critical SGS components are located within the Steam Generator Building (SGB) which is of reinforced concrete and designed to withstand the extremes of environmental conditions, including tornadoes. Each of the three SGS loops is essentially independent of the others and critical components in each loop are contained in separate cells which provide missile shielding. A further separation is provided by locating the steam drum, recirculation pump, majority of recirculation water piping, safety and power relief valves and the majority of the water side control and isolation valves in a cell which is separate from the cell containing the evaporator and superheater modules and sodium piping. Equipment is arranged to facilitate natural circulation.

The sodium dump tank is located in a separate cell on the lowest building level, beneath the IHTS sodium pump. The water dump tank is located in a separate cell, on the second building level, below the steam drum cell.

The SWRPRS RPSTs are located adjacent to the sodium dump tank on the lowest building level. The vent lines to the atmosphere for all three loops are located external to the SGB at a location adjacent to the maintenance shop and warehouse building and opposite to the SGB maintenance bay.

9.2 Steam Generator System Design Evolution

9.2.1 General

A recirculation type steam generator system was specified at the outset of the Clinch River Breeder Reactor Plant (CRBRP) project. Thereafter, the system design evolved as the result of extensive design studies, analyses and

supporting development. The system design and operating/maintenance procedures were modified in response to various SGS design reviews and key system reviews. The SGS system was initially baselined January 1975 [1]; the latest baselined system design is described in Reference 2. The major SGS design evolution is described by subsystem in the following subsections.

9.2.2 Water/Steam Subsystem Design Evolutions

9.2.2.1 Steam Generator Modules

Initially in 1973 the SGS concept contained three evaporators and three superheaters in each of the SGS loops; the water recirculation ratio (i.e., water flow/steam flow) was 6:1 to avoid Departure from Nucleate Boiling (DNB) in the evaporators. As the result of cost reduction studies carried out in early 1974, the number of steam generator modules (SGMs) were reduced to two evaporators and one superheater and the water recirculation ratio was reduced to 2:1 (this recirculation ratio allowed DNB in the evaporators.) [3]. This SGM configuration and water recirculation ratio became part of the early reference SGS design [4]. Subsequent analyses confirmed acceptable thermal strain cycling in the evaporators under DNB conditions [5]. A test program, carried out at General Electric with a single tube steam generator operating at CRBRP conditions showed. (1) no localized damage or accelerated corrosion in the DNB zone, (2) uniform corrosion of the entire tube (extrapolations of the test data would indicate long term evaporator tube life in the order of 30 years) and (3) minimal deposition/fouling [6].

During the initial period of 1973-1974, plant arrangement studies were carried out for three candidate steam generator concepts; single tube hockey stick concept, protected bayonet tube concept and double wall tube--with third fluid leak detection concept. In April 1974 the CRBRP project selected the single wall hockey stick concept as the reference evaporator/superheater for CRBRP [7] [8]. The selection of the hockey stick concept was based on this unit's advanced state of development, recognizing the successful 10,000 hour thermal/hydraulic test of the 35 MW model at the Liquid Metal Engineering Center (now known as the Energy Technology Engineering Center).

Later during the detail design phase design changes were made to the CRBRP steam generators as the result of operating experience with 7-tube test models of the CRBRP evaporators and superheater [9]. These design changes included changes to the tube-to-tube support spacing, changes in tube support geometry and change of tube support material from 2-1/4 Cr-1 Mo to Inconel 718.

9.2.2.2 Steam Drum

A. Reduction in the Number of Risers/Downcomers and Steam Outlets

The initial steam drum proposals from the steam drum manufacturers offered drum concepts similar to those used on conventional fossil fired power plants. These drums had numerous small diameter risers entering the bottom of the drum, numerous small diameter downcomers leaving the bottom of the drum and numerous steam outlets leaving the top of the drum.

After the reduction of SGMs to two evaporators and one superheater per SGS loop had been made, the selected steam drum vendor was requested to provide a design which would minimize the number of risers, downcomers and steam outlets and thus minimize the amount of required external headering. The final result was a steam drum design that would provide adequate internal flow distribution with two risers entering the bottom of the drum, four downcomers leaving the bottom of the drum and one steam outlet leaving the top of the drum.

B. Addition of Special Drum Drain Header

In order to meet the recirculation water chemistry requirements, (discussed in Section 9.2.2.5) the drum drain flow (maximum 10 percent of recirculation water flow) had to be composed of the water from the evaporators prior to mixing with the entering feedwater. This required special attention to the location of the drum drain header.

A design was developed where the drum drain header was located directly below the cyclone steam separators. The feedwater was admitted to the drum in a direction that was away from the drain header. Additional separation of the evaporator water and the feedwater was obtained by vertical baffles extending upwards from the main feedwater header. This arrangement is shown in Figure 9-2. This design ensured that the drum drain water would be the same as the water exiting the evaporator modules.

C. Addition of Elevated Auxiliary Feedwater Sparger

In order to mitigate severe downside temperature transients resulting from injection of cold PWST water into the SGS steam drums during the early phase of heat removal, the auxiliary feedwater sparger was relocated from below the steam drum water level to the steam space near the top of the drum just below the cyclone separators. In addition, spray nozzles were added to the elevated sparger to ensure adequate preheating of the injected PWST water, and a flow limiter was added to the PWST water injection flow control valve to prevent steam drum depressurization. These design problems and their solutions are described in Section 9.3.4 of this report.

9.2.2.3 Recirculation Pumps

A. Reduction in Number of Recirculation Pumps

The original conceptual design of the SGS had three 50 percent capacity recirculation pumps per SGS loop so as to provide the maximum pump availability for each loop. However, during the evolution of the CRBRP design, it was determined that the cost of providing three 50 percent capacity pumps could not be justified considering thermal transients involved in bringing a standby pump up to power, the maintenance of seal cooling for the pump in the standby mode, the reliability of the seal cooling system and the cost of the extra valves. The decision was then made to use one 100 percent capacity recirculation pump per SGS loop.

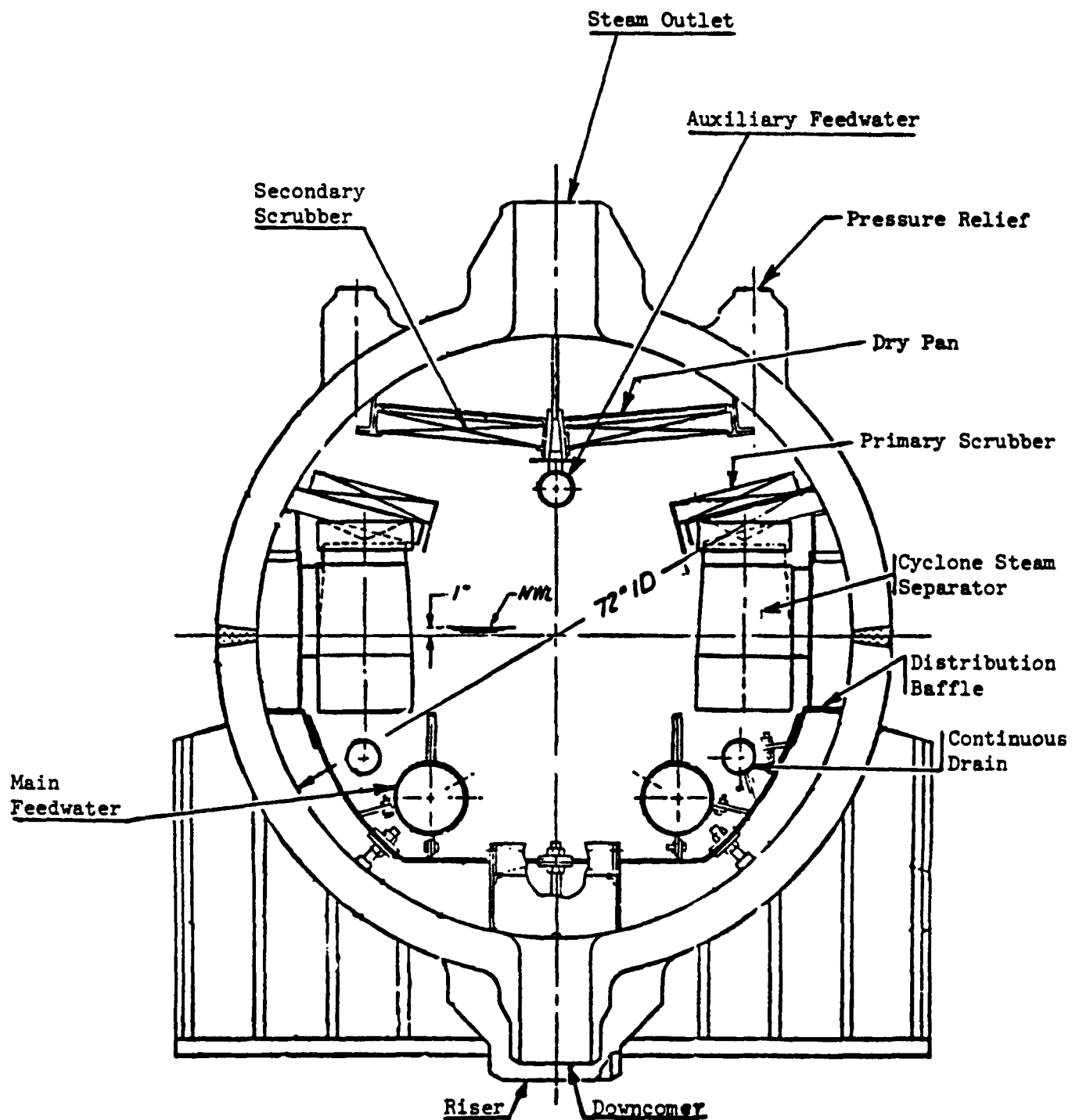


Figure 9-2 CRBRP STEAM DRUM

B. Provision for Cooling of Recirculation Pump Seal

Cooling of the recirculation pump seals was one of the key SGS design problems. The problems encountered and the design solutions are described in Section 9.3.5.

9.2.2.4 Water/Steam Piping and Valves

A. Revised Motor/Pneumatic Operated Isolation Valves to E/H Operated

Sixty five actuated isolation valves were originally required for the SGS and SGAHRS (39 for the SGS and 26 for the SGAHRS). The valves ranged in size from 3/4-inch to 18-inch. Most of the valves were required to be remotely operated during abnormal and shutdown conditions and therefore were required to meet IEEE Class 1E requirements. At the outset of the valve procurement phase, it was anticipated that either motor actuators or pneumatic actuators would be proposed by the valve manufacturers.

However, no supplier could be found to provide any motor actuator in any size or large pneumatic (greater than 3-inch) operators to meet the IEEE Class 1E requirements or CRBRP qualification requirements. Further, of the eleven valve manufacturers requested to bid, only two valve manufacturers were finally responsive. Valves with electro-hydraulic operators were selected for the remotely operated actuated valve application based on their proven capability to meet the IEEE Class 1E requirements.

At the time of Project termination the actuated valve design with electro-hydraulic actuator was nearing completion. Engineering change proposals G1135 and G1186 had been processed to reflect the electro-hydraulic actuators and their associated electrical control and instrumentation requirements [10] [11].

B. Addition of Recirculation Pump Bypass Piping and Valves

Whenever maintenance was required for recirculation pump seals, the water/steam subsystem (WSS) would have to be cooled to below 212°F, preferably below 120°F. This cooling and subsequent reheating operations would extend the plant downtime.

In order to alleviate the above conditions and allow maintenance on the recirculation pump while maintaining the WSS hot by natural circulation of water, a bypass loop around each recirculation pump was added in each WSS recirculation loop.

C. Addition of Preheating Piping

Preheating the WSS piping was another key design problem encountered. This problem and the design solution of additional piping is discussed in Section 9.3.6.

D. Added Provision for BOP Steam Heatup

During the evolution of the CRBRP plant design it was determined that some means had to be provided to heat up the BOP steam lines and the turbine. The Auxiliary Steam Boiler (ASB) located in the BOP could only provide steam at 400°F and its steam generation capability was limited. The requirement of the plant was to achieve the condition known as "hot standby" where the Primary Heat Transport System (PHTS), IHTS, SGS and the BOP steam lines were all at 594°F (saturation temperature at 1450 psig) and ready to rise to power. This condition had to be achieved from a condition where the sodium systems and the steam generator systems were at 594°F (1450 psig) and the BOP and turbine were at room temperature.

This capability was provided by the addition of a steam supply line (with flow control valve) from the superheater inlet to downstream of the superheater outlet check valve.

The method of operation was to bring the reactor critical and then generate sufficient heat to heat the SGS from 400°F to 594°F. When the SGS was at 594°F, reactor power was increased up to where 5 percent power was being generated and the steam generated was being dissipated in the PACC units in each loop. At this point the reactor power was held constant and steam was slowly released through the bypass line to the BOP steam lines. As the SGS pressure was reduced, due to the steam release to the BOP, the PACC units would close down to maintain 1450 psig SGS pressure. In this manner the BOP steam lines would be heated to 594°F. When the BOP steam lines were pressurized to within 25 psi of the SGS pressure, the superheater outlet isolation valves would open and the bypass steam line closed down. The final end point would be when the PACC units were closed down, the steam generated by the SGS was flowing to the steam dump located in the BOP and the BOP steam lines were at 594°F.

9.2.2.5 Water Chemistry

A. Update of Water Chemistry Requirements

The chemistry requirements for the feedwater and the recirculation water were periodically reviewed and updated as additional experience was gained from conventional fossil fuel and Light Water Reactor (LWR) power plants. These revisions were in the form of small changes in the allowable impurity levels and the pH. Essentially the feedwater requirements were comparable to those existing for the modern, fossil fired, once through steam power plants, with one exception. This exception was that the sodium ion concentration in the CRBRP feedwater was limited to a maximum concentration of one part per billion (ppb).

Low sodium concentrations were required to preclude stress corrosion cracking of the evaporator tubes under the DNB conditions that existed in the evaporators with a recirculation ratio of 2:1. Review of the conventional fossil fired, once through steam generators water quality practices showed that the sodium concentration in the

recirculation water should be held to 6 ppb or less. For the 2.1 recirculation ratio, it was determined that the concentrating factor of recirculation water compared to feedwater could be held to 5.5 ratio with a 10 percent drum drain rate. This was considered acceptable since it was believed that a sodium concentration of 1 ppb could be measured and the capability existed to control the sodium level in the feedwater to one ppb or less. Experiments were then conducted at the Nueces Bay fossil fired power plant which demonstrated the capability of maintaining a maximum of 1 ppb of sodium in the feedwater.

B. Specification of 10 percent Steam Drum Drain Rate

As discussed in Section 9.2.2.2B, the requirement of a 10 percent steam drum drain rate resulted from the requirement that the recirculation water concentration factor be held to less than 6.0 for a recirculation ration of 2:1. The 10 percent value was derived from a simple analyses which specified that the drum drain flow had to be the same composition as the water exiting the evaporators as discussed in Section 9.2.2.2B. No mixing of the water exiting the evaporators with the entering feedwater would be allowed until after the drain water had been removed from the drum.

C. Addition of Feedwater Topping Heater

As discussed previously, a 10 percent steam drum drain rate was specified to maintain the proper water chemistry. However, this provision raised the problem of disposition of the drum drain flow after it left the steam drum. The thermal efficiency of the plant would have been seriously affected if the drum drain flow for three loops (330,000 pounds per hour of saturated water at 625°F) was just discarded.

An economic study was performed to determine the most cost effective way of disposing of the drum drain flow. The results indicated a topping heater should be added to the feedwater stream to allow

heating of entering feedwater while cooling off the drum drain flow. The drum drain flow leaving the topping heater was then piped to a flash tank where depressurization occurred. Steam from the flash tank was directed to the deaerator and the water was cascaded back through the feedwater heaters to the condenser.

9.2.3 Design Evolutions of the Sodium Water Reaction Pressure Relief Subsystem

9.2.3.1 Deletion of Centrifugal Separators

In the early referenced design [4], centrifugal separators were provided downstream of the SWRPRS RPSTs to remove any particulate reaction products that had not been removed in the RPSTs before venting the gaseous reaction products to the atmosphere via a flare stack. Primary separation of liquid, solid and gaseous products would take place in the RPSTs in the event of a large SWR. As the SWRPRS design evolved, many large SWR tests were being conducted in the Large Leak Testing Rig (LLTR) facilities at the Energy Technology Engineering Center (ETEC). This testing experience showed that major separation of sodium, SWR products and hydrogen was taking place in the RPS tank. In addition, conservative analyses showed that the sodium oxide concentrations at the CRBRP site boundary would be within acceptable limits without the cyclone separators [12]. On these bases, the centrifugal separators were deleted from the SWRPRS design by ECP G0405 [13].

9.2.3.2 Relocation/Increase in the Number of Reaction Products Separator Tanks

In the early reference design of the SGS, the RPSTs were located on the roof of the SGB; one RPST was provided per heat transport loop. This initial location of RPSTs was selected to minimize the amount of sodium lost from the IHTS in the event the rupture disks failed during an SSE. An additional consideration was to have the tanks easily accessible for maintenance and/or replacement.

As the IHTS and SGS designs progressed, analyses were conducted which indicated that, when consideration was given to the IHTS expansion tank pressure and the coastdown time of the IHTS pumps, sufficient sodium would be

lost from the IHTS so that sodium circulation in the IHTS could not be maintained if the main rupture disk failed. In addition, the SWR reaction loads on the SWRPRS piping were high due to the long pipe runs to the top of the SGB. On the basis of these considerations, the RPSTs were relocated to the bottom of the SGB. Moreover, with the RPSTs requiring a vertical configuration, with the SWRPRS piping inlet at the top and with the desire to minimize the lengths of the SWRPRS piping runs (to minimize the piping loads), the decision was made to utilize two smaller RPSTs per loop rather than one large single RPST per loop.

The relocation of the RPSTs to the bottom of the SGB was done with the knowledge that should a tank or tanks ever have to be replaced, the dismantling of the old tank(s) and the fabrication of a new tank(s) would have to be done in place within the SGB. However, as discussed in Section 9.3.3, the latest information from the LLTR tests at ETEC indicated that the SWR reaction products could be left in the RPSTs. The RPST would be re-used following confirmatory inspection of the affected RPSTs as part of SWRPRS requalification.

9.2.3.3 Revision of Water/Steam-To-Sodium Design Basis Leak

The initial SWR Design Basis Leak (DBL) was specified to be a very conservative seven tube Double Ended Guillotine (DEG) SG tube failure event. Initial analyses disclosed very high reaction loads in the IHTS piping due to acoustic pressure pulses and very high thrust loads in the SWRPRS due to high volume throughput of sodium, liquid/solid reaction products and gaseous hydrogen.

Review of on-going SWR test data from the LLTR tests at ETEC as well as foreign test data disclosed that this SWR DBL was unduly conservative. Accordingly, the SWR DBL was reduced from a seven tube DEG tube failure event to a three DEG tube failure event, with each tube failure occurring at one second intervals by ECPs L10-137 and G1072 [14][15].

Selection of the SWR DBL and accommodation of SWR loads in the steam generator modules and SWRPRS were key SGS design problems. These topics are also respectively discussed in Sections 9.3.1 and 9.3.2 of this report.

9.2.4 Leak Detection Subsystem

9.2.4.1 Reduction In Number of Water/Steam-To-Sodium Leak Detection Modules

In the initial reference SGS design, a water/steam-to-sodium leak detection module (LDM) was located at each of the following locations: superheater sodium vent line, two evaporator sodium vent lines, two superheater sodium exit lines, two evaporator sodium exit lines, and main sodium line to the IHTS pump. In addition, a gas chromatograph was located in the IHTS sodium expansion tank. Several cost reduction studies were performed which resulted in reducing the number of LDMs.

The first cost reduction study resulted in deleting the gas chromatograph located in the IHTS sodium expansion tank since the time for a leak signal to be received was excessive. Also, the LDM located on the main sodium line to the IHTS pump was deleted since it was only a backup to the other leak detection modules.

With the LDMs having isolation valves between the modules and the IHTS piping, the second cost reduction study was directed at combining LDMs. By proper operation of the LDM isolation valves, one could still identify the particular SGM which had the water/steam-to-sodium leak. A study was made relative to the transport times and the various time intervals between a leak developing in a superheater or an evaporator module and the LDMs providing a signal. The results of the study indicated various leak detection modules could be combined. The final locations for the LDMs were established by ECP G0210 [16] as follows:

- o Superheater sodium vent line
- o Evaporator sodium vent lines
- o Superheater sodium exit lines
- o Evaporator sodium exit lines

9.2.4.2 Addition of Remotely Operated Sodium Isolation Valves

In the initial reference SGS design, manually operated sodium isolation valves were located in the inlet and outlet lines for each LDM. These valves were provided to allow the LDMs to be maintained or replaced without having to drain the IHTS. In order to provide double isolation capability, it was planned to establish a frozen sodium plug adjacent to the valve on the IHTS side for LDM maintenance or replacement.

There was opposition within the Project to installing the isolation valves since similar leak detectors developed by Argonne National Laboratory (ANL) and used at the Experimental Breeder Reactor (EBR-II) had no valves. However, the operation of the leak detectors at EBR-II was not required for plant operation since they were development items. At EBR-II, if a leak detector malfunctioned or became inoperative, the leak detector was left alone until some convenient future time when the sodium loop was drained and the module could be repaired or replaced. In the case of CRBRP, a minimum number of LDMs was required for plant operation.

As the CRBRP plant design progressed, it became apparent that in the event of a LDM sodium leak, the capability to remotely isolate the faulted module should be provided. Also, with the reduction in the number of LDMs, the capability of remote valve operation would allow one LDM to monitor two evaporator modules and by valve operation determine which evaporator module was leaking if a water/steam leak was detected. Accordingly, the remotely operated valve feature was added by ECP G0068 to the LDM subsystem design [17].

9.2.5 Sodium Dump Subsystem

9.2.5.1 Sodium Dump Tank Material Change

The preliminary design layout for the sodium dump tank utilized 2-1/4 Cr-1 Mo material based on the requirement that the tank would have to receive and store IHTS sodium after a "hot" dump, with the sodium at about 900°F. As the

design progressed, it was realized that a large cost savings could be achieved if the material could be changed to low carbon steel. With this incentive, a review of the "hot" dump requirement was made.

The system design requirements were reviewed and the following IHTS operational changes were made. In the event the average sodium temperature in the IHTS loop was above 700°F, no dumping of the sodium to the dump tank would be permitted. Sodium circulation in the IHTS would be continued until such time that the average sodium temperature in the IHTS was 700°F or less; then the sodium would be dumped to the sodium dump tank. This operating philosophy was accepted by the Project and the sodium dump tank material was changed to low carbon steel by ECP G0134 [18].

In the unlikely event a sodium dump did occur when the sodium temperature was above 700°F, the event would be in the emergency classification and the tank would have to be requalified. With the sodium dump tank being designed for 700°F and 50 psig pressure, and with the actual pressure conditions of a "hot" dump being much less than the design pressure, the tank would be expected to experience no damage from this transient.

9.3 Key Problems and Resolutions

9.3.1 Selection of Steam Generator Module Design Basis Leak

The CRBRP steam generator is a single wall tube design with water/steam on the tube side and sodium on the shell side. Two evaporators and one superheater module are incorporated in each of three independent heat transfer loops. A SWR relief system is provided to limit the pressure increase and damage to the intermediate heat exchanger during a large leak event.

Recognizing the possibility of a steam generator tube leak occurring during the plant lifetime, it was determined in the early design stage that a conservative upper bound DBL would need to be defined that would provide a design basis for the steam generator and piping systems.

PROBLEM RESOLUTION

In order to establish a DBL, an intensive evaluation was made of the available information on large leak tests and tube wastage. The information available at the time (1974) was a small amount of foreign data on large leaks and a limited amount of U.S. data on tube wastage. With the information that was collected, a conservative position for the DBL was selected as an initial small leak of one tube causing wastage on the surrounding six tubes.

On the basis of the above, the initial DBL was defined as the instantaneous guillotine failure of a single tube followed by the failure of the six surrounding tubes. The initial central tube failure was assumed to result in instantaneous steam/water flow into the sodium at the maximum rate. The steam/water flow rate from the failure of the six remaining tubes was assumed to build up at a rate governed by friction and inertial forces.

As the CRBRP design progressed, analyses were performed using the seven tube DBL. The results were very large forces being applied to the IHTS and SWRPRS piping, resulting in piping and support changes to accommodate the large forces. A study was then undertaken to review the basis for the DBL with the objective of removing some of the conservatism, and thereby reducing impact on the design.

In order to support the design basis leak and develop comprehensive data on the effects of sodium/water reactions, a test program was underway using the Large Leak Test Rig (LLTR) at the Energy Technology Engineering Center (ETEC). The program consisted of two series of tests. The principal objective of Series I was to provide sodium/water reaction (SWR) data for validation of the TRANSWRAP code. Primary objectives of Series II were to define the potential for secondary tube failures and to provide data for verification and calibration of the design methodology. Series II included Double-Ended Guillotine (DEG) tests and leak progression tests in a full-diameter, partial-length, prototypic segment of a CRBRP steam generator. No secondary tube failures occurred in the DEG tests. It was concluded from these tests and from foreign data that secondary failures following DEG event

are highly unlikely. Rapid expulsion of sodium from the leak site region along with prompt action to isolate and blowdown of the water side mitigates the effects of the event.

LLTR Series II tests included intermediate size leaks created by exposing a pre-drilled hole used to simulate the self-wastage leak. These tests resulted in overheating/wastage failures of adjacent tubes. Based on these tests, a plausible leak progression event was developed for the CRBRP steam generator. The plausible leak progression event results in wastage of adjacent tube(s) followed by failure of the intermediate expansion tank rupture disk, and is terminated by blowdown of water from the steam generators in the affected loop. The plausible leak scenario is a conservative progression of events which does not result in failure of the main rupture disks and drainage of the IHTS sodium into the SWRPRS.

In order to redefine a clearly conservative design basis leak (which does not represent a plausible or mechanistic sequence), it was necessary to assume a burst of the SWRPRS main rupture disks. A rapid Equivalent Double-Ended Guillotine (EDEG) failure serves analytically to burst the SWRPRS disk and also to conservatively bound the failure magnitude. The DBL was redefined as follows:

An EDEG failure of a steam generator tube followed by two additional EDEG tube failures. The two additional EDEG failures occur as follows.

One additional EDEG failure occurs at one second after the initial EDEG failure.

A second initial EDEG failure occurs at two seconds after the initial EDEG failure.

This sequence of three EDEG failures occurs after an intermediate-size leak (less than a DEG) from a steam generator tube has increased local pressures in the IHTS to the threshold of SWRPRS main rupture disk burst. The CRBRP DBL is conservative in both the magnitude and the timing of secondary failures.

The revised 3-tube DBL was applied to the steam generator system and IHTS system design. The resulting analyses indicated that loadings were reduced below the original seven-tube design basis and that pipe restraint problems were reduced significantly.

9.3.2 Accommodation of Sodium Water Reaction Loads in the SGS

A DBL event located within the evaporator or superheater component produces substantial structural loadings on components throughout the IHTS and SGS.

The initial system loading is an acoustic pressure pulse resulting from the tube failures and bursting of the reverse buckling main IHTS rupture disks. The acoustic pulse(s) pass through the IHTS piping, producing unbalanced loads on piping supports and component nozzles. A secondary loading results from the quasi-static pressure which rises throughout the IHTS until the SWRPRS lines are cleared. The SWRPRS piping must be designed to withstand the dynamic loadings associated with the sodium slug which is accelerated through the SWRPRS piping into the RPSTs.

PROBLEM RESOLUTION

The TRANSWRAP computer program is used to determine the acoustic pressure loadings in the IHTS and SWRPRS piping system. IHTS piping loads are discussed in Section 7.1.3.1. SWRPRS piping loads and steam generator piping and nozzle loads are discussed below.

The SWRPRS system receives the sodium and sodium/water reaction products and transports the products through piping into the reaction products tank. The main IHTS disk(s) rupture as the pressure rises rapidly from a large SWR and sodium and reaction products are ejected into SWRPRS piping. The fluid entering the piping is accelerated along the SWRPRS piping as the formation of a gas bubble in the affected SGM forces the liquid sodium slug into the SWRPRS RPSTs. Once the liquid clears the SWRPRS line, the pressure in the IHTS is reduced and gas is vented through the SWRPRS lines to the reaction products tank and then to atmosphere through the SWRPRS flare stack.

A significant structural dynamic loading is created by the acceleration of the sodium/reaction products slug passing through the SWRPRS piping. These dynamic loads are resisted by piping structural supports and snubbers. The structural supports carry the dynamic loads into the SGB.

In order to evaluate the structural loadings on SWRPRS piping a number of TRANSWRAP runs were made to establish peak loadings. SWRPRS piping support arrangements were also evaluated by various computer codes to optimize the structural support configurations. Special forgings were required in certain locations to carry SWRPRS loads into snubbers and to the building.

ECP G1072 [15] describes the latest SWRPRS piping arrangement, designed for the three tube DBL. This arrangement provides a structural design that meets all the ASME Code allowable stress criteria for ASME Section III, Class 3 components.

During a large sodium water reaction, acoustic pressure waves develop which travel throughout the SGS and IHTS. These pressure waves result in unbalanced forces on steam generator piping nozzles. Although the steam generator in which the large leak occurs is considered to be faulted, the loadings are also transmitted to adjacent units in the same loop for which the loading is taken as an emergency event. Sodium water reaction loads are but one loading condition for the steam generator. Seismic loads, pipe break loads, dead weight, thermal expansion must all be taken into consideration. Also, it was found during the analysis that the system piping, steam generator component and building all interact in such a way that their individual stiffnesses influence the final loading solution. In order to properly account for the influences of these systems, a total integrated model was developed, which included the key components and piping systems.

A steam generator/piping task force was convened which took into account all of the system loadings applied to the steam generator. The result of this effort was Engineering Change Proposals G0598, G0613 and G0680 covering piping and steam generator interface loads [19-22]. These proposals were accepted and baselined into the steam generator system design.

9.3.3 Requalification of the Sodium Water Reaction Pressure Relief Subsystem

In a large steam generator tube leak event such as the SWR DBL, the pressure rise in the IHTS will cause the main IHTS rupture disk(s) to burst, relieving system pressure and ejecting sodium and reaction products into the SWRPRS piping and RPSTs. The sodium and reaction products would remain in these tanks.

The SWR products will include sodium hydroxide. If a sufficiently large quantity of moisture is present, a caustic solution may form under certain conditions of temperature, stress, electrochemical potential, and concentration. Under these conditions it was postulated that Stress Corrosion Cracking (SCC) of SWRPRS components might occur.

It was highly desirable to demonstrate the integrity of the system following a large SG tube leak event, and thereby avoid the costly alternative of component replacement. To achieve the latter goal it was necessary to establish a method for requalification of the SWRPRS following a SWR event.

PROBLEM RESOLUTION

The requalification program was based upon the following approach:

- A. Selection of SWRPRS materials resistant to SCC.
- B. Provision for direct measurement of the potential for SCC under realistic prototypic conditions, through a test program conducted at ETEC.
- C. Development of an inspection plan for direct visual and dye penetrant examination of selected areas of CRBRP SWRPRS piping to confirm the integrity of the system following a SWRPRS event.

The material selected for the SWRPRS system was carbon steel, SA-155, SA-420 and SA-541. Stress corrosion tests at GE had shown that carbon steel is more resistant to SCC than either 2-1/4 Cr-1 Mo steel or austenitic stainless steel.

A materials test program was conducted at ETEC in the LLTR in conjunction with the LLTR Series II large leak program [23][24]. This test rig was utilized to study the effects of large SWRs. The rig contained a relief system similar to the CRBRP relief system design.

The materials test program contained the following elements.

- A. Installation of CRBRP material U-bend specimens in the LLTR relief piping.
- B. Inspection of U-bend specimens and LLTR relief piping following each SWR test.
- C. Examination of internal surfaces of piping by dye penetrant.
- D. Examination of piping external surfaces by ultrasonic methods.
- E. Destructive examination of selected portions of piping upon completion of LLTR testing.

A series of six SWR tests were conducted in the LLTR. This exceeds by a significant amount the number of SWRs expected in CRBRP. The LLTR piping was exposed to SWR products for a total duration of approximately 60 weeks. Results of the materials test program [25] were.

- A. No SCC was observed in any of the approximately 128 bend samples located in the reaction products tank, or 86 bend samples located in the relief line.
- B. Examination of LLTR piping by ultrasonic testing (UT) was not reliable in detecting SCC, although techniques used were better than required by ASME Code.
- C. Liquid penetrant examination of LLTR piping was an effective examination method and showed no evidence of SCC.

- D. Destructive examination of sections of LLTR piping following the test program showed no evidence of SCC.

A literature survey was also conducted to determine if SCC had been noted as a result of foreign SWRPRS testing. A review of Japanese, French, and U.K. tests indicated that no SCC was reported as a result of testing conducted during SWR test programs.

Based upon the above results, it was concluded that there is very minimal risk of SCC in the SWRPRS piping following an SWR. Requalification would be based upon successful liquid penetrant examination of selected accessible high stress regions of SWRPRS piping.

Based on the above test results, an inspection plan [26] was prepared for the SWRPRS piping and RPST that would provide a strong basis for requalification following a large leak event.

The following requalification approach was developed:

- A. Remove SWRPRS expansion joints and rupture disk assemblies. Clean and inspect. Replace as required.
- B. Examine inside surface of selected accessible field welds by liquid penetrant.
- C. Examine RPST wall thickness for thinning using UT.
- D. Minimize air ingress into system during examinations through use of inert atmosphere.
- E. Retain solid reaction products in RPST. Decant liquid sodium from tank.
- F. Perform system leak test as required by ASME Code.

Based upon the test program and results discussed above, it was concluded that a valid requalification program for the SWRPRS had been developed. The requalification procedures were being incorporated in Section 6 of the SDD at time of Project termination.

9.3.4 Use of SGS and SGAHRS for Decay Heat Removal

CRBRP utilizes the main heat transport system and the SGS in the reactor decay heat removal train. A dedicated heat removal system was provided to supply water to the SGS and condense the steam produced during the decay heat removal phase. This dedicated system is the SGAHRS described in Section 8.0 of this report.

As the result of safe shutdown requirements and incorporation of NRC Regulatory Guides the following system requirements were developed.

- A. Systems are enclosed in structures designed to withstand a design basis tornado, a design basis flood and a safe shutdown earthquake (SSE).
- B. Critical systems are provided with uninterruptible source of power.
- C. Systems are designed to perform their decay heat removal function during and after an SSE.
- D. Systems are designed to perform their decay heat removal function after an initiating failure event followed by a single active failure.
- E. Systems are designed to prevent common mode failures.
- F. Steam/water systems are designed to withstand a guillotine pipe break and still perform their decay heat removal function.

As noted in Section 8.0 the design of the SGAHRS utilized the following major components:

- A. One PWST which had sufficient capacity to supply the SGS with water for decay heat removal and system leakage over a 30 day period.
- B. Three PACC units (one per heat transport loop) located near the top of the SGB in enclosed vaults, over the SGS steam drums.
- C. Two electric motor driven pumps to pump water from the PWST to the steam drum.
- D. One steam turbine driven pump to pump water from the PWST to the steam drum.

The SGS and the SGAHRS operate in the following manner to remove reactor decay power. Upon a turbine trip and loss of feedwater, the SGS is isolated from the BOP and the reactor is tripped. During a short time period, about 20 minutes, the heat load imposed on the SGS by the IHTS sodium flow is composed of the reactor decay heat and the sensible heat of the plant as the plant cools off. This initial heat load exceeds that which can be handled by the PACCs condensing steam from the drum, with the result that steam is vented to the atmosphere through the SGAHRS power vent valves at the steam drum exit and at the superheater exit. With steam being lost from the SGS via the power vent valves, the SGAHRS pumps are then activated to supply water from the PWST to the steam drum to maintain drum water level. After about 20 minutes, the IHTS heat load imposed on the SGS is reduced to the point where the PACCs, operating at full load, can condense sufficient steam from the steam drum, to handle the heat load. The SGAHRS operation to supply water to the steam drum is then only required to make up for system leakages.

Under long term decay heat removal conditions, the PACCs accept steam from the steam drums, condense the steam and return the condensate to the steam drums. No serious temperature transients or control problems were evident during this mode of operation. However, in the initial steam venting phase, temperature transients and control problems were identified.

In the initial phase, immediately following isolation of the SGS from the BOP and initiation of the SGAHRS, water is lost from the SGS through steam release through the SGAHRS vent valves. This loss of water is then made up by pumping water from the PWST into the steam drum. PWST water at 70°F is injected into the steam drum at 625°F causing a severe downside temperature transient of the water leaving the steam drum. This severe downside transient was due to the PWST water being injected into the drum below the surface of the drum water and near the steam drum outlets to the downcomers and would result in a severe downside temperature transient imposed on the evaporator at the inlet nozzles. This transient was considered unacceptable. It should be noted here, that with this method of PWST water injection into the drum, no depressurization of the steam drum occurred and no control problems were noted.

PROBLEM RESOLUTION

In order to alleviate the severe downside temperature transient of the water leaving the steam drum, a design change was initiated. This change moved the PWST water injection sparger above the drum water level, near the top of the drum, just below the scrubbers. The PWST water would then be injected through orifices into the steam volume within the drum. The purpose of the change was to heat up the PWST water by condensation of steam, so that when the PWST water was mixed with the steam drum water, the downside temperature transient was reduced to an acceptable level.

However, there were indications that the PWST water droplet size would be excessive with the existing sparger orifices so that adequate preheating of the PWST water would not be obtained. A study was then initiated into the installation of spray nozzles to provide a spectrum of droplet sizes with small enough mean droplet size so that adequate preheating of the PWST water was achieved. The study confirmed that spray nozzles would provide acceptable droplet sizes and that the drop time (time to drop to the surface of the water) would be acceptable considering the spray was into a steam atmosphere at 1500 to 1800 psig. Therefore, spray nozzles were installed on the elevated sparger.

The installation of spray nozzles on the elevated PWST water inlet sparger in the steam drum reduced the severe downside temperature transient of the steam drum downcomer water to an acceptable value. However, with injection of the 70°F PWST water droplets into the steam drum steam volume, steam condensation resulted and, under this condition severe depressurization of the steam drum pressure would occur.

In the initial phase, with steam being lost through the SGAHRS vent valves, a rapid drop in the drum water level occurs. Since there is a twenty second delay in water injection from the PWST when the water injection is initiated, the control system, because of the low drum water level, would call for maximum PWST water injection into the steam drum. With the maximum PWST water injection into the steam drum steam volume, depressurization of the steam drum pressure to an unacceptable low value would occur. In order to alleviate this characteristic, a flow limiter was added to the PWST water injection flow control valve. This flow limit would be imposed whenever the steam drum pressure decreased to 1400 psig. Analyses conducted, using the flow limiter, indicated the drum water level was maintained within acceptable limits and the steam drum did not undergo any severe depressurization.

In the design of the elevated sparger inside the steam drum, there was concern as to water hammer on the sparger when the PWST water injection was initiated. This potential condition was investigated and alleviated by the addition of a small hole in the sparger below the drum water line, so that the sparger was always filled with drum water up to the drum water level.

9.3.5 Recirculation Pump Seal Cooling

Based upon their LWR experience and test experience, the supplier of the SGS recirculation pumps indicated a requirement for clean high pressure seal cooling water so that reasonable seal life requirements could be met. In the LWRs, this cooling water was supplied by the control rod drive system, a highly reliable water source. However, in CRBRP, a similarly reliable system was not available.

PROBLEM RESOLUTION

The seals along the recirculation pump shaft are required to minimize the loss of water from the SGS. To limit the pressure drop required across each seal, the design included three mechanical seals in series [27] [28]. Due to material limitations, the seals could not withstand temperatures greater than 200°F. To prevent the seal area from reaching the SGS temperature (up to 600°F) and to remove the frictional heat from the sealing faces, cooling was required.

Many alternatives were considered for the seal water injection system [29]. Among those considered were the following:

- A. A system which took 5 gpm from the SGS, at the recirculation pump suction, and passed it through an injection pump, a water-water or water-air heat exchanger, and filters before injecting the cooled water into the pump. One system would be provided for each SGS loop. The active components were to be supplied with Class 1E power to assure seal cooling with the loss of offsite power. The vented seal water was to be directed to the SGAHRS PWST to conserve the emergency water inventory. If a water-water heat exchanger was used, cooling would have been provided by the emergency chilled water system.
- B. A system which extracted 5 gpm from the SGS, at the recirculation pump discharge, and passed it through a water-water heat exchanger and filters before injecting the cooled water into the pump. This system relied upon the pump head to drive the cooling water. To ensure injection water when the pump was not operating, connections were to be provided to the balance of plant (BOP) and SGAHRS feedwater pumps, for normal and emergency operating conditions, respectively.
- C. A system which took 5 gpm from a BOP system, which was capable of providing injection water under normal and emergency operating conditions.

After evaluating the alternatives for cost and adequacy, the system with the following features was chosen by ECP G0163 [30] [31].

- A. Seal injection system with water from the main feedwater (MFW) system. Water would be extracted from downstream of the MFW pumps and cooled, through a heat exchanger, with condensate water. During plant startup, the MFW makeup pump would be used. If water was not available from the condenser, water would be taken directly from the condensate storage tank (CST). Since it was not desirable to subject the recirculation pump seals to extremely cold water, the water from the CST would be heated prior to injection.

The selection of an injection water system which was not qualified to be a safety system was based upon economic considerations. If the recirculation pump seal cooling water system was subjected to 16 loss-of-offsite power events, as specified in the CRBRP overall plant design description, and the system was not supplied with Class 1E power, a cost penalty of approximately \$6M would be imposed. This cost would include replacement parts for all three recirculation pumps seal assemblies and "loss-of-availability" cost. The cost penalty would be greater than the cost (~\$2M) of the safety grade injection system. However, a Project decision was made that the seal injection water system design should be based upon a realistic assessment of the number of loss-of-offsite power events.

Reliability analyses indicated the probability of the subject event to be 4×10^{-6} /hour, which converts to one event in the life of the plant. Based upon this judgment, the cost penalty (approximately \$400K) did not justify a safety grade injection system.

- B. Recirculation pump bypass and isolation capability. The bypass is required to maintain the steam generator system heat removal capability, in conjunction with SGAHRS, in the event of a loss-of-offsite power event. Under this condition, cooling would be lost to the recirculation pump seals and would result in seal failure. Since the main feedwater would be lost, makeup water during plant cooldown would have to be provided from the SGAHRS PWST.

Because this is a limited supply and because there would be an unacceptable leak rate (11 gpm/pump) from the SGS through the failed pump seals, capability to isolate the pump was provided. So that natural circulation could be maintained in the recirculation loop (steam drum to evaporators and back to drum), the pump bypass piping and associated valves were provided.

9.3.6 Design Problems Associated with Preheating of the SGS

The SGS is required to be preheated from room temperature to 400°F at 10°F per hour [2]. Preheating is accomplished, using only the BOP feedwater and steam from the Auxiliary Steam Boiler (ASB), under the following two conditions:

- A. The steam generator modules (SGMs) initially at room temperature (RT); the IHTS and SGMs empty of sodium; with the SGS under wet layup conditions;
- B. The SGMs at 400°F with sodium at low flow circulating through the modules and the balance of the SGS empty of water and at room temperature.

The first requirement results from the normal preheating requirement with the SGS including the evaporators and superheater at RT, the IHTS empty of sodium, and the IHTS either at 400°F or at RT.

The second requirement results from the desire to avoid draining sodium from the IHTS every time the WSS of the SGS was required to be shut down and drained. It was anticipated that maintenance on the numerous safety relief and isolation valves would require the WSS to be cooled down to room temperature many more times than the IHTS would be required to be drained of sodium. The ability to preheat the balance of the WSS with the evaporators and superheater at 400°F also reduces the plant downtime and improves the plant availability.

PROBLEM RESOLUTION

In order to meet the first of the above requirements, a startup recirculation loop was added to the WSS by ECP G0418 [32] [33] to allow the circulation of water from the recirculation pump discharge to the superheater inlet line and then from the superheater outlet to one of the evaporator riser lines to the steam drum. In this manner, water could be circulated through the two evaporators and one superheater and as the water was gradually heated, the evaporators, superheater, and the balance of the WSS (piping, pump, steam drum) were also gradually heated.

A second method of preheating the superheater had been evaluated, which involved the introduction of auxiliary steam into the unit. This alternative was not pursued because of concerns over non-uniform heating of the large steam generator structure. Matching of the preheat rate with the balance of the steam generator system would also have presented control and monitoring problems.

A second application of the superheater recirculation loop is to provide the capability for initial water flushing of the superheater. A third application of the recirculation loop is circulation of the wet layup water in the superheater, with the WSS completely filled with water (evaporator, superheater, steam drum, recirculation pump, and piping).

In order to meet the second of the requirements, a bypass loop was provided around the evaporator modules by ECP G0418 [32] [33]. With this arrangement, the evaporators and superheater could be isolated from the balance of the WSS to allow the balance of the WSS to be heated from room temperature to 400°F at 10°F per hour by circulation of heated water. The interconnecting piping, from the isolation valves to and from the evaporators and superheater are heated at the same rate as the balance of the WSS by admission of ASB steam. The condensed steam in the interconnecting piping was drained off through steam traps.

Use of a temporary piping system for the recirculation loop and the bypass piping was evaluated as a cost savings alternative. An economic evaluation was performed comparing temporary and permanent piping. This evaluation showed that permanent recirculation loop and bypass was the most economic solution.

With the addition of the superheater recirculation loop and the evaporator bypass, the WSS has the capability of being preheated from room temperature to 400°F at 10°F per hour by circulation of water which is gradually being heated from the BOP. Feedwater from the BOP at a temperature of 300°F is admitted to the steam drum at a controlled rate to heat up the WSS recirculation water at 10°F per hour. As the feedwater was being admitted to the steam drum, an equal amount of water is taken from the recirculation water entering the steam drum. This water bypasses the topping heater and then returns to the BOP.

When the WSS reaches the same temperature on the BOP feedwater, namely 300°F, any additional heating of the recirculation water is accomplished by admitting ASB steam directly to the steam drum. In this manner, the condensing steam in the steam drum provides the heat input to the recirculation water. The water drained from the steam drum is then directed through the topping heater before being returned to the BOP.

The method of preheating the SGMs by circulation of heated water through the tubes results in a substantial heatup time. The long heatup time is due to the thermal resistance to heat transfer from the outer rows of tubes to the shroud and then to the outer shell of the steam generator modules. In order to reduce this heatup time, a Project decision [34] was made to install electrical heaters on the outside of the steam generator module shell and thereby heat up the shell at the same rate as the tubes were being heated.

9.3.7 Evaporator Chemical Cleaning

One concern in the water chemistry area was related to determining when the evaporators should be removed from service and chemically cleaned on the water side. One solution considered was to monitor the performance of the

evaporators and whenever the sodium temperatures or flows reached some prescribed limits (indicating that fouling was significantly affecting performance) the plant would be shut down for evaporator cleaning. This approach was questioned in view of the concern that excessive corrosion of the evaporator tubes might occur if the thickness of the water-side fouling layer exceeded 70 microns.

Another solution considered in the problem of scheduling evaporator cleaning was to clean the evaporators whenever the thickness of fouling layer reached 70 microns. The problem with this approach was the ability to know when a 70 micron fouling layer thickness had been reached since the evaporator performance would probably not have noticeably degraded with this relative thin fouling layer.

This concern had not been resolved at time of Project termination. However, additional insight into evaporator fouling behavior was expected from the CRBRP prototype steam generator test at ETEC.

Experimentation had been carried out at General Electric to determine the appropriate evaporator chemical cleaning solution. The reference cleaning solution derived was a deoxygenated water solution containing 10% EDTA, 2% citric acid and 1% hydrazine adjusted to a pH of 7 with ammonium hydroxide. This solution was found to effectively remove fouling scale with acceptably low base metal corrosion of about 0.1 mil [35].

9.4 Final System Status

9.4.1 Status of Design/Analyses

SDD-53, Sections 1 and 2 were baselined and updated as the system design evolved [1] [2]. The final RDSD for the SGS [36] had been provided in December 1982. The remaining SDD-53 sections [37] [38] [39] [40]- covering limitations, set points and precautions (Section 3); operations (Section 4); casualty events and recovery procedures (Section 5) and maintenance and in-service inspection (Section 6) were scheduled to be updated and submitted

to Westinghouse/CRBRP Project Office for approval by April 30, 1984. Pre-Operation Test Specifications and Construction Test Requirements were scheduled for updating and reissue to Project participants in the same time frame.

The System 53 FWSDR had been held in May 1983 [35] [41] [42] [43]. Resolution of about half of the FWSDR team comments had been identified [44]; the remaining comments were to be resolved through the aforementioned updating of the SDD draft Sections 3-6. In addition, a detailed SGS Operation, Maintenance and Test (OM&T) procedures review was held on August 30-September 1, 1983 at Burns and Roe. OM&T comments from this review [45] were to be dispositioned by March 1, 1984 to support the submittal of updated draft SDD-53 Sections 3-6.

Open technical issues for the SGS were addressed in a meeting among CRBRP Project Office, Westinghouse and General Electric on September 22, 1983 [46]. Those issues, relating to main IHTS rupture disk margin during an SSE and the potential "hot tee" condition during a SWR DBL are discussed in Section 7.1.4 of this report. Other issues related to revisions of water/steam-to-sodium LDM interface requirements given in ECP G1445 [47] and procedures; SWRPRS relief line revisions in ECP G1072 [15]; revision of control and instrumentation design for SGS valve operators in ECP G1186 [11]; additional seismic analysis required because of the water dump tank/recirculation water pump coupling; operational requirements associated with recirculation water pump heatup after maintenance, startup strainer and SGS heatup for system hydrotest; provision of water dump valve interlocks; water chemistry specification revision; and provision of SGS maintenance platform requirements to Burns and Roe. Disposition and/or action items to resolve these issues are given in Reference 46. In October 1983, GE provided a recommendation on interlocks for the water dump subsystem [48]. In October 1983, maintenance platform requirements were provided for the SGs for various elevations up to and including 823 feet and for the balance of SGS components up to elevation 765 feet [49].

The LDM design layout drawing was baselined and detail design was completed. A design review was planned for January 1984 prior to releasing the design package for procurement. Final stress analyses for the LDMs and for the LDM-to-IHTS interconnecting piping was scheduled to be completed in FY84 and early FY85 respectively. ECP G1445 [47] updating the LDM interface requirements was in the Project review cycle at time of Project termination.

In October 1983, a final design review was held on the 6-inch IHTS expansion tank rupture disks, 8-inch sodium dump tank rupture disks and 18-inch main IHTS duplex rupture disk assemblies. The design review team recommended full fabrication release for the smaller disks and partial fabrication release for the 18-inch rupture disk assemblies. Holds were recommended for certain details on the 18-inch disk design drawings pending resolution of design team concerns [50]. ECP G1149 updating ICDs and other documentation for conformance with the final rupture disk design was receiving Project review at time of Project termination [51]. Final stress analyses for the rupture disks were underway at General Electric with completion of stress reports scheduled for FY84.

The SWRPRS design was in the final design phase. ECP G1072 [15] describing the revised SWRPRS design in response to the approved SG 3-tube DBL (ECP L10-137 [14]) was undergoing CRBRP Project Office/Westinghouse review at time of Project termination. The final SWRPRS piping stress report was scheduled for issue in September 1984.

The steam/water subsystem piping had been designed. Stress analyses had been initiated. The final stress report for this piping was scheduled to be completed in FY85.

A number of SGS components had been designed, manufactured and delivered to the CRBRP storage site. Included were sodium dump tanks, reaction product separator tanks, steam drums, water recirculation pumps, steam/water check valves and instrument, vent and drain valves. Designs were complete for the manual valves, safety relief valves and water dump tanks; manual valves and relief valves were partially fabricated. Design work was nearing completion

for the actuated isolation valves and SWRPRS expansion joints. SGS control valves were in the final stages of assembly at time of Project termination. Final design and stress analyses of the SGMs were underway; fabrication had been initiated.

9.4.2 Outstanding Activities to Complete Design

The following ECPs covering SGS design were undergoing Project review at time of Project termination.

- A. G1072, (Rev. 3) - ECP updated the SWRPRS documentation and modified SWRPRS piping in accordance with the revised SWR DBL approved in ECP L10-137.
- B. G1149, (Rev.2&3) - ECP revised main IHTS rupture disk documentation to be consistent with the rupture disk detail design.
- C. G1445, (Rev.D-1) - ECP revised LDM ICD to be consistent with LDM detail design.
- D. G1186, (Rev. 2) - ECP revised control and instrumentation design for SGS electro-hydraulic valve operators used on actuated isolation valves.

The principal remaining engineering work for the SGS was as follows:

- A. Update draft Sections 3,4,5 and 6 of SDD-53 and submit for approval to baseline.
- B. Issue SGS overpressure protection report.
- C. Update WSS and SWRPRS design reports.
- D. Prepare LDM design report and O&M Manual.

- E. Conduct LDM design review, resolve any comments and release design drawing package for procurement.
- F. Resolve outstanding technical issues [46] and process ECPs as required.
- G. Complete stress analyses and issue final stress reports for LDMs, LDMs-to-interconnecting piping, SWRPRS and steam/water piping.
- H. Complete detail design of actuated isolation valves and SWRPRS expansion joints.
- I. Update and reissue Pre-Operation Test Specifications and Construction Test Requirements.
- J. Complete preparation of SGS Maintenance Platform Requirements (765 feet and above) and issue ICD.

9.5 References

- 1. "System Design Description, Steam Generator System, SDD-53," General Electric Company Specification 22A3525, Revision 0, January 27, 1975.
- 2. "System Design Description, Steam Generator System, SDD-53," General Electric Company Specification 22A3525, Revision 101, August 2, 1983.
- 3. Letter D890-74-64, R.E. Skavdahl (GE) to C. V. Backlund (SAN) and W. M. Jacobi (W), "Number of Modules, Water Recirculation Ratio and Action Plan for Evaporator/Superheater Selection for CRBRP Steam Generator System," March 18, 1974.
- 4. "Clinch River Breeder Reactor Plant Reference Design Report," Westinghouse Electric Corporation, June 1974.
- 5. Letter XL-791-00965, F.E. Tippets (GE) to W. M. Jacobi, "Clinch River Breeder Reactor Plant-Steam Generator System 53-Evaluation of Strain Cycling Effects in the DNB Zone of the CRBR Evaporators," December 29, 1975.
- 6. Letter XL-498-80004, L. M. Downes (GE) to J. E. Nolan (W), "Transmittal of DNB Effects Test Program Final Report," CRBRP-GEFR-00100, June 1977.

7. Letter WR4024, J. J. Buggy (W) to A. E. Mravca (DOE-CH), "Clinch River Breeder Reactor Plant; Steam Generator Program Management Review Meeting Minutes," April 17, 1974.
8. Letter RRD:PG:102, D. R. Riley (DOE-RRD) to R. D. Thorne (DOE-SAN) and R.H. Bauer (DOE-CH) "Steam Generator Development Program," April 18, 1974.
9. J. R. Impellezzeri, et al., "Few Tube Model Post-Test Examination," CRBRP-GEFR-00505, July 1980.
10. ECP G1135 "SGAHRS 52: Update Electrical Requirement" approved August 24, 1983.
11. ECP G1186, Revision 4, "System 53 Valve Electro-hydraulic Actuators; System 53/56 C&I Update," September 19, 1983.
12. Letter LR81105, D. K. Goesser (W-OR) to L.W. Caffee (CRBRP-PO), "CRBRP; System 53 Engineering Change Proposal G0405," October 12, 1978.
13. ECP G0405 "System 53: Delection of Requirements for a Centrifugal Separator," approved November 7, 1978.
14. ECP L10-137 "Redefinition of the Design Basis Sodium/Water Reaction Event," approved July 14, 1980.
15. ECP G1072, Revision 3, "System 53-SWRPRS Incorporation of 3-tube DBL and General Update," August 1, 1983.
16. ECP G0210 "System 53: Leak Detection Reduction," approved December 6, 1977
17. ECP G0068 "System 53: Leak Detection Subsystem," (added remote valve operators) November 19, 1976.
18. ECP G0134 "System 53: Change Sodium Dump Tank Thermal Transients and Code Classification," approved February 22, 1977.
19. ECP G0598 "System 53: Water/Steam Pipe Break Loads," approved August 21, 1980.
20. ECP G0613 "System 51B Main Loop Piping and Support Arrangement," approved December 12, 1980.
21. ECP G0680 "Resolve Local Stress Problem-Steam/Water Piping," approved September 26, 1980.
22. ECP G0547 "Superheater and Evaporator Vent Line Piping Routing," July 26, 1979.
23. D. E. Knittle and J. C. Amos, "LLTR Series II Test Program Large Leak Tests Final Report," GEFR-00609, October 1982.

24. J. C. Amos and P. M. Magee, "LLTR Series II Test Program Intermediate Leak Tests Final Report," GEFR-00663, June 1983.
25. Letter XL-590-20333, F. P. Crimi (GE) to W. J. Purcell (W), "CRBRP; Sodium Water Reaction Product Relief Subsystem (SWRPRS) Requalification Program Information," August 5, 1982.
26. Letter XL-590-20331, F. P. Crimi (GE) to W. J. Purcell (W), "CRBRP; Sodium Water Reaction Product Relief Subsystem (SWRPRS) Requalification Program Information," July 30, 1982.
27. Bingham Willamette Company Drawing E17564, Revision K, Seal Assembly/Layout.
28. Bingham Willamette Company Drawing CSML 00N21, Revision 42, Contract Specific Material List.
29. Letter LG90517, N. A. Pollack (W) to G. G. Glenn (GE), "Clinch River Breeder Reactor Plant; System 53 Cleanup ECP G0163, Recirculation Pump Seal Water Injection Subsystem and Cooling," August 28, 1979.
30. Letter XL-497-10412, P. Bohaboy (GE) to W. J. Purcell (W), "CRBRP; ECP G0163, Revision 11, Recirculation Pump Cooling Requirement," July 8, 1981.
31. Letter EN.CM.81-221, D. R. Riley (CRBRP-PO) to W.J. Purcell (W) and M. Ascher (B&R), "Approval of ECP G0163, Revision 11, CRBRP; SGS 53: Recirculation Pump Seal Injection and Cooling Water for Pump and Drive Motor," September 25, 1981.
32. Letter EN.CM:81-173, D. R. Riley (CRBRP-PO) to W. J. Purcell (W) and M. C. Ascher (B&R), "Approval of ECP G0418, Revision 1, Revision of System 53 Requirements for Operation at Hot Standby, Refueling and Wet Layup," November 5, 1981.
33. Letter LG810627, W. J. Purcell (W) to G. G. Glenn (GE) and R. J. Seeley (AI), "CRBRP; Approval of ECP G0418, Revision of System 53 Requirements for Operation at Hot Standby, Refueling and Wet Layup," November 18, 1981.
34. Letter WR820052, B. Levy (W) to P. R. Brewington (CRBRP-PO), "CRBRP; Recommended Methods for CRBRP Steam Generator Dry Heat-Up and Dry Cool-Down," April 27, 1982.
35. Letter XL-590-830116, F. P. Crimi (GE) to W. J. Purcell (W), "CRBRP; System 53, Steam Generator Final Working System Design Review (FWSOR) Package Transmittal," March 18, 1983.
36. Letter XL-599-20258, W. W. Phelan (GE) to W. J. Purcell (W) and W. W. Dewald (W), "CRBRP; SHRS Reliability (WAG612); RDSO of the Steam Generator System," December 28, 1982.
37. "System Design Description, Steam Generator System, SDD-53, Section 3 (Draft)," General Electric Company Specification 22A3525AA, Revision D-6, April 15, 1983.

38. "System Design Description, Steam Generator System, SDD-53, Section 4 (Draft)," General Electric Company Specification 22A3525AB, Revision D-6, December 28, 1982.
39. "System Design Description, Steam Generator System, SDD-53, Section 5 (Draft)," General Electric Company Specification 22A3525AC, Revision D-2, February 15, 1983.
40. "System Design Description, Steam Generator System, SDD-53, Section 6 (Draft)," General Electric Company Specification 22A3525AD, Revision D-6, February 25, 1983.
41. Letter XL-590-830148, F. P. Crimi (GE) to W. J. Purcell (W), "CRBRP; Addendum System 56 (RHTIS) Supplement to the System 53 (SGS) FWSOR Data Package Transmittal," April 6, 1983.
42. Letter XL-590-830149, F. P. Crimi (GE) to W. J. Purcell (W), "CRBRP; System 53, Steam Generator System Final Working System Design Review (FWSOR) Agenda Transmittal," April 11, 1983.
43. Letter CR-747:MD:83-142, C. H. Fox (CRBRP-PO) to W. J. Purcell (W), "Transmittal of Steam Generator Final Working System Design Review Findings," June 29, 1983.
44. Letter XL-590-830567, F. P. Crimi (GE) to W. J. Purcell (W), "CRBRP: Responses for Steam Generator System, System 53 FWSOR Comments, Due October 31, 1983," October 28, 1983.
45. Letter LG830426, G. G. Ritter (W) to G. G. Glenn (GE), "CRBRP; System 53 Operation, Maintenance and Test Procedures' Review," September 22, 1983.
46. Letter LG830452, W. J. Purcell (W) to G. G. Glenn (GE), "CRBRP; System 53, Steam Generator System Open Technical Issues," October 14, 1983.
47. ECP G1445, Revision D-1, "Small Leak Detector Module ICD," August 11, 1983.
48. Letter XL-590-830572, F. P. Crimi (GE) to W. J. Purcell (W), "CRBRP; System 53 Open Technical Issue No. 14, Interlocks for the Water Dump Subsystem," October 28, 1983.
49. Letter XL-590-830549, F. P. Crimi (GE) to W. J. Purcell (W), "CRBRP; System 53 Maintenance Platform Requirements up to the Elevation 765 Feet in the Steam Generator Building," October 14, 1983.
50. D. F. Casey and J. C. Whipple, "Design Evaluation of CRBRP Rupture Disks," General Electric Company Report CRBRP-GEFR-00714, October 1984.
51. ECP G1149, Revisions 1 and 2, "Rupture Disk Documentation Update," August 3, 1983 and October 13, 1983.

9.6 Major ECPs Affecting SGS Design

The following ECPs were the major ECPs which affected the SGS system and piping baselined designs:

G0043	Superheater Outlet Piping & Valve Material Changes.
G0059	System 53, Steam Drum Feedwater Sparger.
G0068	System 53, Leak Detection Subsystem (Added remote Operators).
G0077	System 53, Water/Steam Line Rupture Flooding Protection for SGB.
G0134	System 53, Change Sodium Dump Tank Thermal Transients and Code Classification.
G0142	System 53, Steam Water Piping Material and Vent Line Size Change.
G0163	System 53, Recirculation Pump Seal Injection and Cooling Water for Pump and Drive Motor.
G0210	System 53, Leak Detection Reduction.
G0224	System 53, System Modification to Provide for Startup/shutdown Operation.
G0248	SWRPRS, Removal & Replacement of RPS Tanks.
G0317	System 53, RPS Tank Removal & Replacement in Segments.
G0405	System 53, Deletion of Requirements for a Centrifugal Separator.
G0418	Revision of System 53 Requirements for Operation at Hot Standby, Refueling and Wet Layup.
G0476	Steam Generator, Water/Steam Side Code Reclassification.
G0478	Leak Detection Module Code Classification Change
G0547	Superheater and Evaporator Vent Line Piping Routing.
G0598	System 53, Water/Steam Pipe Break Loads & Restraints.
G0613	System 51B, Main Loop Piping and Supports Arrangement.
G0680	Resolve Local Stress Problems--Steam/Water Piping.
G0691	Superheater Outlet Isolation Valve Bypass Valve Revision.
G0974	Baselining of Water/Steam-to-Sodium Leak Detection Module Locations.

G1072* System 53, SWRPRS Incorporation of 3-Tube DBL and General Update.

G1097 Modification of Water/Steam-to-Sodium Leak Detection Alarms & Monitoring Instrumentation.

G1149* Rupture Disk ICD Update.

G1186 System 53, Valve Electro-hydraulic Actuators; System 53/56 C&I Update.

G1245 Superheater Outlet Valve Logic Revision.

G1339** Revision of SWRPRS Activation Signals to Class 1E.

G1445 Small Leak Detector Module ICD.

W1906 Replacement of Bolted Steamheads with Integrally Welded Steamheads

L10-137 Redefinition of the Design Basis Sodium/Water Reaction Event.

*Under Project Review at time of CRBRP termination.

**In Preparation at time of CRBRP termination.

10.0 REACTOR HEAT TRANSPORT INSTRUMENTATION AND ELECTRICAL HEATING & CONTROL SYSTEMS

10.1 Reactor Heat Transport Instrumentation System

10.1.1 Summary of Functions and Description

The Reactor Heat Transport Instrumentation System (RHTIS) measures parameters in the Primary Heat Transport System (PHTS), the Intermediate Heat Transport System (IHTS), the Steam Generator System (SGS), and the Steam Generator Auxiliary Heat Removal System (SGAHRs). It provides signal conditioning equipment to prepare signals obtained from sensors for use in the Plant Control System (PCS), Plant Annunciator System (PAS), Plant Data Handling and Display System (PDH&DS) and Plant Protection System (PPS). It provides the Primary and Intermediate Sodium Pump Drive Systems and the control functions for the PHTS, IHTS, SGS and SGAHRs that are not part of the PCS.

In general, the sensors provided by the RHTIS for monitoring the system process are located on or in close proximity to the pipes, tanks, and related equipment containing the process. The process variable is converted to an electrical signal at this location and transmitted to local instrument and control panels or termination cabinets for subsequent rerouting to the Main Control Room. These panels contain the signal conditioning equipment, instrument power supplies and equipment required for local alarm, indication, and control functions. Local Instrumentation and Control (I&C) panels are provided on the mezzanine in the Reactor Containment Building, on the mezzanine in the Intermediate Bay, and in the Steam Generator Building. Back panels are also provided in the Main Control Room for control logic equipment and Post Accident Monitoring indicating recorders. Measurement channels which are part of the PPS requirements, are separated, isolated, and identified according to appropriate safety criteria. When process measurements are required by both the PPS and other instrument systems, they are generally supplied first to the PPS which then provides a buffered output for use in the other systems. If the signal is required locally, buffers are provided in the local signal conditioning equipment. This sharing of measurements between PPS and other instrument systems eliminates the need for an additional measurement

channel and minimized the duplication of signals transmitted to and from the Main Control Room. Where a 1E buffered signal is used for the remote shutdown panel indication, a transfer device is provided in the local PPS panel to isolate the PPS signal to the PPS in the event of Main Control Room inaccessibility.

SGAHRS controllers are in the local SGAHRS panels with local and Main Control Room control stations. Local transfer switches allow transfer of control from the Main Control Room to the local SGAHRS panel in the event of Main Control Room inaccessibility. Each control station has an analog indication of the controlled parameter, setpoint adjust, manual loading control and an auto-manual selector switch.

Controllers for the PHTS, IHTS and SGS are in the Main Control Room back panel with control stations on the main control panel.

Manual control devices on the main control panel for electrically controlled safety related equipment are generally duplicated in local 1E panels. This provides operator control when the Main Control Room is not accessible. Remotely operated valve position indication is provided at local and main control room panels.

Redundant PPS channels are physically separated and electrically isolated according to the Class 1E requirements. Each channel of a measured parameter is powered by a separate power division, i.e., Division I for Channel 'A', Division II for Channel 'B', and Division III for Channel 'C'. Where one or two 1E channels are provided for a measured parameter, power division assignment is by loop, i.e., Division I for Loop 1, Division II for Loop 2, and Division III for Loop 3.

Power division for SGAHRS is assigned to maintain maximum control of the system in the event of a power division failure.

Nuclear safety related equipment is Class 1E qualified and it consists of the PPS instrumentation, the accident monitoring instruments, instrument channels

for remote shutdown panel indication and the Decay Heat Removal equipment. Non-nuclear safety related equipment that is associated with Class 1E equipment are Class 1E qualified.

10.1.2 Evolution of Design

The scope of the RHTIS was as follows.

- A. Provide engineering design and associated instrumentation support for the PHTS, IHTS, SGS and SGAHRS.
- B. Prepare and maintain documentation to include: System Design Description, Logic Diagrams, Interface Control Documents (ICDs), Elementary Diagrams, Panel Arrangements and Equipment and Instrumentation Tests.
- C. Provide interface requirements with all systems interfacing with the RHTIS.
- D. Provide control system analysis and updates as required.
- E. Develop design and requirements, and provide support engineering to procure instrumentation equipment for the heat transport system. The effort includes: development of E-specifications, temperature sensor requirements, and general instrumentation and control requirements.
- F. Provide procurement support for RHTIS equipment.

The first issue of the RHTIS (System 56) System Design Description [1] was approved in April 1975. The current revision is number 105 which incorporated in excess of 180 ECPs submitted by General Electric and other project participants with interfacing systems. There were further changes planned at the time of termination of the CRBRP project.

The first RHTIS preliminary design review was held in July 1976 and a design report [2] issued in September 1976. A second preliminary design review was held in January 1977 with a design report [3] issued in February 1977. This report contained 19 specific recommendations and 5 generic recommendations. A response to the design review findings [4] with all comments addressed and resolved was made in July 1977. Logic Diagrams, Elementary Diagrams, and Panel Arrangement Drawings were completed in 1978, 1980 and 1981 respectively. Updating of the drawings for design changes and interface requirements were made by ECPs after being issued. The RHTIS provided task force support for Decay Heat Removal and responded to Key System review recommendations [5]. A control room task force was supported over several years with General Electric resolving 19 recommendations [6]. Concurrently the primary and intermediate sodium pump drive systems were specified and procured. Five drive systems were delivered to storage at Oak Ridge, Tn. and one drive system was used for the prototype sodium pump testing with water and sodium. Closeout of items from the design reviews were completed in July 1983 [7][8]. The RHTIS responded to Key System Reviews of Class 1E and non-1E electrical power distribution [9] and Na/NaK Leaks [10]. The Final Working System Design Review (FWSDR) was held in February 1983.

10.1.3 Key Problems and Resolutions

10.1.3.1 General

A major portion of the RHTIS work was that of a secondary system. A secondary system is the system which provides the functions and services or support required by others at the specific interface so that the primary system can satisfy the required criteria. The RHTIS's major effort was to provide the instrumentation for the plant systems delineated in Section 10.1.2. Key System Reviews that impacted the RHTIS design efforts were Decay Heat Removal [5], Control Room Task Force [6], Power Distribution (1E and non-1E) [9], and Pump Level Control/Leak Detection [10]. The RHTIS responded to and implemented ICDs, SDD requirements of other systems, Key System findings, Task Force recommendations, ECPs of other systems and recommendations from the Project Office as a result of reviews or submittals.

10.1.3.2 Drive Motor Failure During Design Basis Event Qualification Test

The Vertical Drive Motor (VDM) Assembly failed to survive the Design Basis Event (DBE) qualification test; this test was intended to validate the operation of the VDM, at pony motor speed, in a chamber filled with high levels of sodium aerosol, simulating the effect of a spilled sodium storage tank in the Reactor Containment Building and a subsequent fire.

Both the 75 HP pony motor and a smaller auxiliary blower motor became clogged with sodium oxide deposits and failed after their bearings overheated.

PROBLEM RESOLUTION

After an extensive analysis of both the specification requirements and the possibilities for redesign of the failed motors, it was concluded that the only feasible resolution was to screen the motors from the potential high sodium aerosol environment. General Electric and Westinghouse tentatively agreed on a design solution, involving both enclosures over the motors and certain improvements to the motors to enhance their operation in the higher temperature environment inside the enclosures.

Although the enclosure design was not baselined at the time of termination, the high temperature changes to the motors had either been incorporated (in the case of the pony motors) or organized as mod kits (in the case of the blower motors).

Reference 11 provides descriptions and analyses of the DBE test. The proposed enclosure and the resulting changes to the VDM environmental requirements are described in Westinghouse ECP L10-200.

10.1.4 Final System Status

10.1.4.1 Status of Design/Analysis

The instrumentation diagrams for the PHTS, IHTS, SGS, and SGAHRS were completed and baselined. The control logic diagrams for the SGAHRS, SGS, IHTS (sodium dump and sodium flow control), and SGAHRS initiation and for SGAHRS Class 1E power division assignment were baselined.

Cable reduction programs had been conducted, which included minimizing nonessential redundant instrumentation.

The baselined control logic for the startup and main feedwater control valves was modified to implement the recommended control scheme from the main feedwater control system. Similarly, the Sodium/Water Reaction Pressure Relief Subsystem (SWRPRS) trip control logic required a modification to preclude a SWRPRS trip due to a single component failure. (The SWRPRS trips the intermediate sodium pump pony motor and isolates the main and auxiliary feedwater supply to the steam drum of the faulted loop.)

The primary and intermediate sodium pump drive cable ICD and control logic diagrams were baselined. Also baselined were the test and calibration ICD, covering PPS signals that the RHTIS is providing to the PPS, and the cable termination ICD between the PPS and RHTIS.

The Main Feedwater Control System analysis covering the startup/low-power control valves and the bumpless transfer feature was completed.

The elementary diagrams for control of the SGAHRS, SGS, and IHTS were completed and baselined. Also baselined were the SGS/SGAHRS logic panel arrangement drawings. The portion-of the RHTIS instrumentation and controls required in support of SGAHRS was reviewed as part of the SGAHRS FWSDR. The review included resolution of all Key Systems Review and Main Control Room Task Force comments. All RHTIS and SGAHRS interface documents were evaluated and made to be consistent.

The equipment specification for the SGS and SGAHRS controls, and the general instrumentation ordering data, were released for Project use.

A draft update of the RHTIS (System 56) SDD Sections 3, 4, 5, and 6 was completed and issued for the FWSDR review. Sections 1 and 2 were revised, and all comments were resolved.

The ALARA data were completed, and all comments were resolved. The instrument diagrams for the primary and intermediate sodium pumps were issued for Project use.

The RHTIS Final Working System Design Review (FWSDR) was conducted in February 1983. All FWSDR comments were evaluated and dispositioned for appropriate action.

The RHTIS Electrical Instrumentation and Control (EI&C) Instrument Index was updated and computer-generated instrument data sheets were prepared from this index to be made a part of the instrument procurement materials. All RHTIS instrument drawings (i.e., Panel Arrangements, Logic Diagrams, ICDs and Elementary Diagrams) were updated to include all approved changes subsequent to baselining of the drawings.

The General Electric Large Motor and Generator Department was awarded the contract for the drive systems. All the drive systems were built and delivered. The first delivered drive system successfully supported water testing of the prototype sodium pump at the pump vendor test facility and sodium testing of that pump at the ETEC. The drive system performance was evaluated at ETEC during the pump sodium test. This program demonstrated that the drive had the capability to satisfy all operating requirements. Test included closed loop control test with a CRBRP electromagnetic flow meter and with the plant flow control system providing a signal to the drive system speed controller. The other five drive systems were delivered to storage near the plant site at Oak Ridge, Tennessee.

10.1.4.2 Outstanding Activities to Complete Design

Work-in-progress included incorporation of design changes including Inoperability Status Monitoring System [12], Valve changes [13] [14], Small Leak Detection Design (ICD CG52030), Accident Monitoring [15], Identification of Class 1E Equipment [16], Redrawing of documents to meet CRBRP-PO guidance [17] [18], RDT Standard F2-2 Design Verification, Pre-Operational and Start-up Test Specifications [19].

Completion of the design, to include all ECPs and obtaining approval to baseline, was in progress with 26 out of 43 drawings and parts lists baselined, 5 out of 9 specifications baselined, and 35 out of 44 ICDs baselined. There were 71 active ECPs at the time of termination with 21 approved but not incorporated. The new Leak Detection Panel design was started. The FWSDR Package D was in the process of getting final GE approvals before being sent to Westinghouse/CRBRP Project Office.

10.1.5 References

1. System Design Description SDD-56, "Reactor Heat Transport Instrumentation System," GE Specification 22A3555, Rev. 105, September 23, 1983.
2. Letter XL-590-10621, G. Billuris (GE) to W. M. Jacobi (W-OR), "CRBRP; System 56 Preliminary Design Review," September 28, 1976.
3. Letter XL-590-77097, G. G. Glenn (GE) to W. M. Jacobi (W-OR), "CRBRP; System 56 Preliminary Design Review," February 1, 1977.
4. Letter XL-590-70490, D. H. Krueger (GE) to W. M. Jacobi (W-OR), "CRBRP; Response to Design Review Team Findings of System 56 Preliminary Design Review," July 25, 1977.
5. Letter XL-590-00379, G. B. Kruger (GE) to W. J. Purcell (W-OR), "CRBRP; Key Systems Review Recommendations Response --Decay Heat Removal," December 19, 1980.
6. Letter XL-590-10141, G. B. Kruger (GE) to W. J. Purcell (W-OR), "CRBRP; Resolution of CRTF and Key System Review Recommendations," March 16, 1981.
7. Letter XL-593-830089, R. J. Michalak (GE) to W. J. Purcell (W-OR), "CRBRP; Closeout of Sodium Pump Drive System Preliminary Design Review," July 14, 1983.

8. Letter XL-593-830098, R. J. Michalak (GE) to W. J. Purcell (W-OR), "CRBRP; Closeout of Sodium Pump Drive Related Open Items from System 56 Design Review," July 29, 1983.
9. Letter XL-590-830131, F. P. Crimi (GE) to W. J. Purcell (W-OR), "CRBRP; Response to Key Systems Design Review of Class 1E and non-1E Electrical Power Distribution," March 25, 1983.
10. Letter XL-590-00301, D. H. Krueger (GE) to W. J. Purcell (W-OR), "CRBRP; Resolution of Commitments for Na/NaK Leaks, Key System Review Task Group," October 20, 1980.
11. Letter XL-593-10833, L. N. Salerno (GE) to W. J. Purcell (W-OR), "Design Basis Event Test of the Sodium Pump Drive System," October 1, 1981.
12. ECP L10-125, "Revision of OPDD-10 Regulatory Guide 1.47 Implementation," December 9, 1981.
13. ECP G1135, Rev. 5, "SGAHRs Update Electrical Requirements," August 24, 1983.
14. ECP G1186, Rev. 4, "Revision of Systems 52 & 53 Actuated Valves to Electro-Hydraulic Actuators," September 21, 1983.
15. ECP L10-156, Rev. 3, "Accident Monitoring," January 29, 1982.
16. ECP L10-134, "Identification of Documents Related to Class 1E Equipment," December 15, 1981.
17. Letter LG830201, G. G. Ritter (W-OR) to G. G. Glenn (GE), "CRBRP; Criteria for Preparation of Electrical Instrumentation and Control (EI&C) Drawings," May 9, 1983.
18. Letter LG830178, G. G. Ritter (W-OR) to G. G. Glenn (GE), "CRBRP; Drawings Needed to Install and Maintain Hardware for Systems 56 and 68," April 20, 1983.
19. Letter XL-590-830543, F. P. Crimi (GE) to W. J. Purcell (W-OR), "Resolution of GE Comments on Pre-operational and Startup Test Sequence Document," Rev. 8, September 30, 1983.

10.1.6 Major ECPs Affecting System 56 Design

Listed below are the major ECPs which affected System 56 design:

G0338	Pump Instrumentation
G0418	SGAHRS (PWST) Water Impurity Requirements; Transfer of Requirements to SDD 53
G0438	IHTS Pump Drive System Instrument Channel Requirements
G0574	Modification of Selected MCP Instrument Ranges and Scales to Provide a Better Man-Machine Interface
G0627	Addition of Flowmeter to AFW Pump Recirculation Lines
G0691	Superheater Outlet Isolation Bypass Valve Revision
G1015	MCP Instrument Ranges & Scales
G1044	Modify Turbine Driven AFW Pump Control Logic
G1056	Misc. Changes to Steam Drum Equip./ISIS & BOP Feedwater Temperature
G1062	System 51B's Incorporation of the Definitions of Hot Standby & Shutdown
G1094	Modification of SGS Logic Diagram and Main Control Panel ICD
G1126	System 56: Movement of Pump Drive Controls
G1133	System 56: Changes to the Main Control Panel
G1134	PHTS Pumps - Requirements to Preclude Loss of Sodium Level
G1135	SGAHRS (System 52) Update Electrical Requirements
G1139	Update Control Requirements in SDD 52 (C&I)
G1174	General Updating of SGAHRS I&C
G1186	Revision of Systems 52 & 53 Actuated Valves to Electro-Hydraulic Actuators
G1242	System 56 Update of Affected Documents to Implement TIRT Recommendations
G1243	General Update of SDD 53

G1245	Superheater (Outlet) Bypass Valve Logic Revision
G1311	AFW System Bypass During Startup & Maintenance
G1315	Steam Generator Sodium Instrumentation Update
G1351	SDD 56 Environmental Requirements - Sodium Fire Conditions
G1362	General Updating of SDD 56
G1379	SDD 52 Update
G1407	RHTIS 56 Update SGAHRS Light Indication
G1450	System 53/56 I&C Update

10.2 Piping & Equipment Electrical Heating & Control System

10.2.1 Summary of Functions and Description

The Piping and Equipment Electrical Heating and Control System (PEEHCS) provides electric heating of the piping and equipment of the interfacing systems. The system is used to preheat the piping and equipment preparatory to filling with sodium, to hold a given temperature and to progressively melt sodium frozen in a pipe starting at the end connected to a molten (free) mass.

The power circuits are located in twenty one Local Control Centers (LCC) located throughout the plant so each circuit can be as near as possible to the heated pipe or equipment. Each LCC receives electrical power (480 Volts, 60 Hertz, 3-phase, 4-wire) via one circuit from one of the two non-1E electrical power distribution systems (System 12 interface). Circuit breaker panel boards provide branch circuit protection for each heater circuit.

The Operator's control devices are in one Master Control Center (MCC) and three Operator Control Centers (OCC). The MCC is located in the Main Control Room and one OCC is located in each building of the plant and is contiguous with one of the local Control Centers.

The system is usually operated from the MCC but a portable control can be carried to one of any of the LCC's for system operation from that location. In addition, the OCC can be used as an operating position for those LCC's served by the OCC.

The temperature is regulated by a microcomputer system which controls the on-time of the heater through a solid state switching device. A feedback signal is obtained from a thermocouple sensing the temperature of the heated zone. The setpoint temperature is established by a prearranged program but the setpoint can be altered by operator action. The difference between the setpoint and feedback temperatures is used to determine the proportion of a 17-second period during which the heater will be energized. The range of on-time is from 1/120-second (one-half cycle) to 17-seconds (full on). For example, an 8.5-second on-time, 8.5-second off-time, would produce 50 percent of the rated power of the heater. The amount of on-time is established by a

computer calculation using the thermal heating process characteristics. Proportional (Gain), Integral (Reset) and Differential (Rate) parameters are entered for each heating circuit. These parameters are known as the PID parameters and the computer calculation is made by the PID Algorithm.

For main piping in the heat transport systems and on all equipment, the heaters are spaced away from the apparatus to be heated but are within the inner jacket of the insulation. The spacing varies from 0.25-inch to 1.25-inch. The heaters are tubular, mineral insulated, 0.5-inch in diameter and formed into a "hairpin" configuration in order to place the terminals near one another. Small piping (less than 8-inch diameter) is generally heated by mineral insulated (MI) heating cable strapped directly to the pipe. Since the heating cable is applicable only to pipe lengths of twenty to thirty feet (depending on the heating requirements), short lengths of small piping are heated by tubular heaters. One-inch freeze-vent piping is heated by straight tubular heaters 3/8-inch in diameter.

Thermocouples provide the signal for temperature regulation and for general temperature monitoring. All control circuits have one thermocouple which serves both the control and the monitor function but large zones or critical areas are fitted with two or more thermocouples. The output of all thermocouples, both control and monitor, are displayed as digital numbers in units of degrees Fahrenheit.

All information is displayed on a Cathode Ray Tube (CRT) and/or a high speed printer. The operator communicates with the system by means of a keyboard. The printer provides a written (hard copy) record of operator and system actions. The three devices (CRT, keyboard and printer) are available at the MCC and each OCC. The portable control equipment does not include a printer.

When abnormal conditions are detected by the microcomputer, a circuit from the OCC to the Plant Annunciator System indicates "ELEC HTR SYS TROUBLE" and identifies the building (RCB, SGB, RSB) in which the abnormal condition exists. The specific trouble is determined by reference to the printer output or by interrogation of the system through the keyboard and CRT display. An annunciator signal is generated by the following abnormal conditions:

- o Setpoint deviation limit
- o High temperature limit
- o Low temperature limit
- o Open thermocouple
- o Open circuit breaker
- o Loss of load (more than 10%)
- o Communication failure (OCC to LCC)

The design of the PEEHCS is described in detail in Reference 1. This report also summarizes the testing program carried out in support of the MI cable design.

10.2.2 Evolution of Design

The trace heating design was based on that for the FFTF with effort applied to reducing costs and improving the control system. The PEEHCS (System 68) SDD [2] was issued in April 1975 and is now to Revision 38 with other ECPs pending. While not all the revisions were major, the number indicates the evolution and iterations that had taken place. System conceptual arrangements consisting of 1/2-inch diameter tubular heater application for vessels, components, and piping were completed.

There have been changes as the PEEHCS design has evolved and as a result of Interface Control Document (ICD) implementation. When ICDs and construction drawings were received from other system designers, the layouts of heaters on vessels, components and large pipes were started. The basic layouts have remained the same unless a change was requested by the other system engineers.

A testing program evaluated the possibility of utilizing mineral insulated cable for heating small piping (8 inches in diameter or less). A Development Requirements Specification [3] was issued in August 1978. The program was performed at ETEC. As a result of the testing, a final design was made and a MI Cable Equipment Specification [4] was issued in August 1983. Test results are reported in Reference 1.

The heater control scheme design was finalized, consisting of three "stand alone" systems, one for each building (SGB, RCB, and RSB), linked to and supervised by the MCC in the Main Control Room. A Heater Power and Control Panels Specification [5] was issued in October 1978 and Ordering Data [6] issued in September 1979. A contract was issued to Research Incorporated of Minneapolis, Minnesota [7] in February 1980 for design and fabrication of the control panels. At the time of Project termination panel fabrication/acceptance testing was about 70% complete.

The instrument tag identification numbering system [8] changed for all drawings. The connection diagrams were revised to accommodate addition of junction boxes, terminals, cable numbers and circuit identification. The original trace heater specification [9] was rewritten to include hermetic seals and flexible leads installed at the factory. This new specification was not issued. The junction boxes that are used at each heater for termination and connection to power leads are described in a specification [10] and drawings.

10.2.3 Key Problems and Resolutions

10.2.3.1 Design Heaters for Operational Availability over a Plant Life of 30 Years.

Resolution of this problem was to employ a conservative design. Watt densities were also reduced by derating heaters to at least half of rated power. Factory installed hermetic seals to meet humidity requirements and moisture absorption were specified. Factory installed flexible lead wires were used to eliminate splicing or brazing in the field. Heaters were to be generally mounted in a hairpin configuration to reduce bends, spirals or serpentine installations. Hot-to-cold junctions of the heaters were to be made internal to the sheath and surrounded by compacted insulation. One hundred percent redundant heaters are installed in inaccessible areas with capability to connect in a terminal box in an accessible area.

10.2.3.2 Difficulty of Trace Heating Small (8-inch or Less) Sodium Pipes with Normal Tubular Heaters.

This problem was resolved by using MI Cable for pipe runs of 20 to 30 feet and 3/8-inch diameter tubular heaters for short lengths of pipe.

10.2.3.3 Provide Acceptable Locations for Several Thousand Local Heater Connection Boxes.

This problem was resolved by designing three size boxes with four electrical configurations. The boxes were to be mounted on the outside of the thermal insulation by flanges for flat or semiflat surfaces or by reversible runners for various sizes of pipes. The boxes accommodated either the tubular heaters or MI cable.

10.2.3.4 Schematic Type Electrical Diagrams Unacceptable to Show Cable Type and Number, Heater Connections, etc.

Resolution of this problem was as follows: A priority task to update all existing heater arrangement drawings (in excess of 50) with connection diagrams was implemented. Input sheets were generated by Engineering to show details such as series/parallel connections, all heaters and wiring, junction and terminal boxes, type and sizes of cables, control panel connections and thermocouple routing. These input sheets were used as input to a program for a Computer Aided Drafting (CAD) system to generate the drawings.

10.2.4 Final System Status

There was considerable work in progress at the time of Project termination. There were approximately one hundred heater arrangement drawings. Some of the later piping heater arrangement drawings were not yet baselined. A few drawings had not been started due to lack of isometric piping drawings from interfacing systems. Parts lists associated with the heater arrangement drawings required updating.

The top sheets (of the heater arrangement drawings) that show heater and thermocouple placement on the piping and components were almost completed, except for interfacing systems lacking isometric drawings information. In 1983 the instrument tag number was changed to correct multiple existing System 68 tag number formats to comply with the format of WARD-D-0059, Rev. 16. All the drawings were revised to this new instrument tag numbering system.

About mid-1983 the last evolution of the heater connection diagrams took place. All drawings were retrofitted with this methodology to show connections, terminals, wire sizes and types, junction boxes, terminal boxes and grounding schemes. As of time of CRBRP termination, there was a program underway for Burns & Roe to identify number and location of terminal boxes for heaters and thermocouples. These terminal boxes are the enclosures in an accessible area and should not be confused with junction boxes near the heater terminals. Some packages of drawings remained to be completed.

About fifty individual heater drawings exist, mostly issued in 1979 and 1980 for large components. Since that time the wiring methods and margins have changed. The heater shapes and configurations are correct, but the electrical data in the charts require revisions.

The Equipment Specifications were in various stages of modifications at the time of Project termination as noted below:

- o The tubular heater specification [9] was completely rewritten; the new document 23A3020, Rev. D-1 was in the Project review cycle (ECP G1501).
- o ICD CG52168 "Trace Heater Penetration" was redrawn and was in the Project review cycle (ECP G1441).
- o Specification 23A2494 "System 68 Thermocouple Holders" was in the process of being revised (ECP G1500).

- o Specification 23A3053, "Heater Connection Junction Box Assemblies" had been issued; however, the specification control drawing 972E147, "Junction Box Heater Connection Assembly" was not completed.
- o Additional work was planned for Specification 23A2475, "Heater Power and Control Panel" and the associated ordering date 23A2475AA. This specification would be updated to reflect ICD changes, revised heat rates and other system changes. This update would have required modifications of the heater power and control panels such as rearrangement/replacement of some breakers and output modules and expanded thermocouple handling capability.

Other engineering change proposals being developed at time of Project termination were as follows:

- o ECP G1503 "SDD Update"--This entailed extensive changes for Sections 1 and 2 and Appendices B, D, E, F and G of SDD-68.
- o ECP G1396 and ECP G1502--These ECPs reflect latest MI cable configuration and installation information respectively.
- o ECP G1499 "Terminal Boxes on IHX and Cold Traps"--This ECP defined terminal box arrangement on the outside of shielding or structures where the heater terminals were not available for normal junction box use.

10.2.5 References

1. D. E. Hafer and J.H. Skomp, "CRBRP Piping and Equipment Electrical Heating and Control System - Final Report", CRBRP-GEFR-00713, November 1984.
2. System Design Description, SDD-68, "Piping & Equipment Electrical Heating & Control System", GE Specification 22A3526, Revision 38, July 10, 1983.
3. Development Requirements Specification (DRS) 68.01, "MI Cable Use Tests", GE Specification 23A2458, August 20, 1979.
4. GE Specification 23A2655, "Equipment Specification MI Cable", Revision 0, August 16, 1983.
5. GE Specification 23A2475, "Equipment Specification Heater Power and Control Panels", Revision 15, September 16, 1983.
6. GE Specification 23A2475AA, Specification Heater Power and Control Panels Ordering Data", Revision 10, April 15, 1983.
7. Contract 78115, General Electric to Research Incorporated for Heater Power and Control Panels, February 1980.
8. ECP G1383, "System 68 Instrument Tag Number", September 30, 1983.
9. GE Specification 23A2353, "Equipment Specification System 68 Trace Heaters", Revision 2, May 8, 1980.
10. GE Specification 23A3053, "Equipment Specification Heater Connection Junction Box Assemblies", Revision 0, October 25, 1983

10.2.6 Major ECPs Affecting System 68 Design

Listed below are the major ECPs which affected System 68 design:

G0147	General Update of SDD-68
G0388	System 68 - Delete System 91 Interfaces
G0525	System 68 - Redefinition of Electric Heaters to Include MI Cable
G0536	System 68 - Single Phase Heater Power
G0664	System 68 - Use of MI Cable
G0704	System 68 - PHTS Trace Heater Redesign
G1022	System 68 - Delete Use of Heater Standoffs for Small Pipe and Valves
G1145	System 68 - Heater Failure Detection, Ground Fault Interruption Test and Update of Control Requirements
G1383	System 68 - Instrument Tag Number
G1448	System 68 - Update Heater Connection Diagrams
G1464	System 68 - Heater Arrangement Drawings

11.0 AUXILIARY LIQUID METAL SYSTEMS - SYSTEM 81

11.1 Overall System Scope

The Auxiliary Liquid Metal System, System 81, is composed of five inter-related, but separate, subsystems, each with its independent functions. Although all are part of System 81, each is referred to both in the detailed design and in this report as a "system." The five systems are listed below.

- Sodium and Sodium-Potassium Alloy (NaK) Receiving System
- Primary Sodium Storage and Processing System
- Ex-Vessel Storage (EVS) Processing System
- Primary Cold Trap NaK Cooling System
- Intermediate Sodium Processing System

All systems are exclusively for liquid metal service. With the exception of NaK used in the EVS processing and primary cold trap NaK cooling systems, sodium is the liquid metal used throughout the plant. The NaK used is the eutectic sodium-potassium alloy containing 22.2% sodium and 77.8% potassium. Only a small amount of NaK is used in CRBRP (all in System 81); of the total plant liquid metal inventory of approximately 439,000 gal, only approximately 4,000 gal is NaK, the balance being sodium.

11.2 Functions and Description

11.2.1 Functions

The principal functions of each of the systems within System 81 are listed in the following table.

System	Functions
Na/NaK Receiving	Transfer incoming Na/NaK to plant storage vessels
Primary Na Storage and Processing	Provide primary sodium purification and thermal expansion compensation Establish normal reactor vessel Na level Provide Na fill, drain, storage for primary and EVST Na Backup reactor decay heat removal (DHRS)
Primary Cold Trap Cooling	Cool primary cold traps
Ex-Vessel (Spent Fuel) Processing	Cool and purify ex-vessel storage tank (EVST) sodium
Intermediate Na Processing	Purify intermediate heat transport system (IHTS) sodium. Provide capability to fill IHTS loops.

11.2.2 Description

11.2.2.1 Overall System Configuration and General Features

11.2.2.1.1 Configuration

The functional arrangement of System 81 components is shown in Figure 11-1. This figure shows the relationship of each of the five systems and the principal physical interfaces with other Clinch River Breeder Reactor Plant (CRBRP) systems.

11.2.2.1.2 General Features

Some of the general features applicable to the overall System 81 design are listed below:

All liquid metal free surfaces are maintained under argon cover gas.

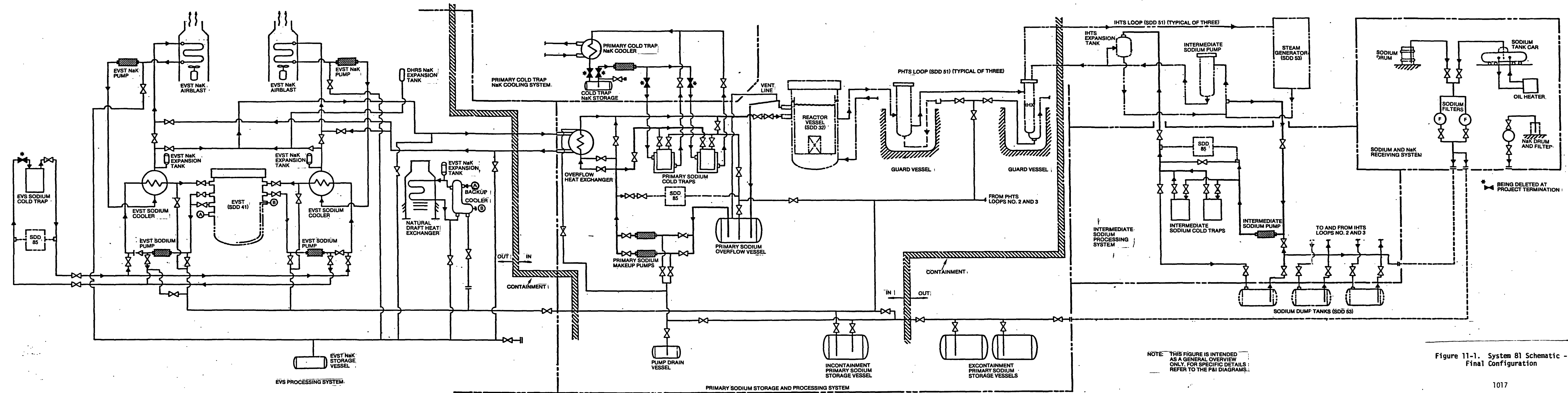


Figure 11-1. System 81 Schematic - Final Configuration

All radioactive components (those containing primary or EVST sodium) are located in shielded, nitrogen-inerted cells.

All sodium components are electrically preheated. Although not required to maintain NaK in a liquid state (NaK freezing point is 9°F) some of the NaK components are electrically preheated for system purposes--either transient mitigation or to protect against sodium freezing in a sodium/NaK heat exchanger.

With the exception of the components in the Na/NaK receiving system, all components are designed to Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code (see Section 11.2.2.1.3). All safety-related (Class 1, 2, or 3) components are seismic Category 1, are tornado and missile protected, and, as applicable, are on emergency power.

Continuous leak detection monitoring is provided by System 66 for all components. The principal means of leak detection is the monitoring (aerosol detection) of cell atmosphere for all equipment cells (rooms). Individual "spark plug" or cable detectors are also provided for major components: storage tanks, heat exchangers (both liquid/liquid and liquid/air), and each valve.

The overall system is designed and arranged to meet the inservice inspection (ISI) requirements of Westinghouse Advance Reactor Division report WARD-D-0244, "CRBRP Preservice and Inservice Inspection Requirements for Sodium and Cover Gas Systems" (Ref. 1) and the ALARA* criteria of WARD-D-0177, "CRBRP ALARA Criteria and Guidelines" (Ref. 2).

Operation of all systems is from local panels. All safety-related components are also operable from the main control room.

*As low as reasonably achievable.

11.2.2.1.3 Safety and ASME Code Classifications

The final safety and ASME Code classifications of System 81 components are listed in Table 11-1. The safety classes are defined in Section 1.3.1.6 of Overall Plant Design Description (OPDD-10) (Ref. 3). Additional discussion of the evolution and basis for these classifications is given in Section 11.3.1.1.6.

11.2.2.2 Sodium and NaK Receiving System

The sodium and NaK receiving system is a small system consisting of components used to transfer incoming sodium and NaK from the shipping containers to the various liquid metal storage tanks in the plant. Bulk frozen sodium is received in 10,000-gal railroad tank cars. An oil-fired tank car heater is provided to heat and circulate oil through an outer jacket on the tank cars to melt the sodium. Temporary drain lines are connected to drain the melted sodium by gravity into the storage tanks. A sodium drum melt station is provided to permit meltout of a sodium drum, using electrical clamshell heaters, for small makeup requirements following initial plant fill. The melt station includes filters which are used for all incoming sodium.

A NaK drum unloading station, also including a filter, is used to transfer NaK, which is shipped (in liquid state) in 30-gal drums. The system is not safety related, and all components are commercial-grade hardware. Following plant fill, the components are disconnected and stored onsite. The components are essentially the same as those used for the Fast Flux Test Facility (FFTF).

During the evolution of the CRBRP design, this system remained unchanged except that, based on FFTF's recommendations, the size of the oil-fired tank car melt station was increased to approximately double the FFTF size in order to expedite tank car meltout; a 24-h melt time, to approximately 350°, was a design objective.

TABLE 11-1
COMPONENT SAFETY AND ASME CLASSIFICATION - SYSTEM 81

Subsystem or Component	Safety Classifi- cation	ASME Code Class
Sodium and NaK Receiving System		
All Components	None	Sect. VIII ANSI B31.1
Primary Sodium Storage and Processing System		
Overflow Heat Exchanger	SC-1	III-1
Overflow Vessel	SC-1	III-1
Primary Sodium Makeup Pumps	SC-1	III-1
Primary Sodium Cold Traps	SC-3	III-3
Makeup Pump Drain Vessel	None	III-3
In-Containment Primary Sodium Storage Vessel	SC-3	III-3
Ex-Containment Primary Sodium Storage Vessel	SC-3	III-3
Overflow Line/Makeup-Return Piping	SC-1	III-1
PHTS Drain Lines to Second Isolation Valve	SC-1	III-1
PHTS Drain Lines Downstream of Isolation Valve	SC-2	III-2
Piping Outside Containment	None	III-3
EVS Sodium Processing System		
Sodium and NaK Drain Lines Downstream of Isolation Valves and Cold Trap Piping	SC-3	III-3
EVST NaK Storage Vessel	None	III-3
Natural Draft Heat Exchanger	SC-3	III-3
EVS Cold Trap Crystallizer and Economizer	SC-3	III-3
NaK Piping Third Loop	SC-3	III-3
All Other Sodium Components	SC-2	III-2
All Other NaK Components	SC-2	III-2
Primary Cold Trap NaK Cooling System		
All Components	None	III-3
Intermediate Sodium Processing System		
All Components	SC-3	III-3

11.2.2.3 Primary Sodium Storage and Processing System

This system interfaces principally with the reactor vessel and primary heat transport system (PHTS) loops. With the exception of the ex-containment storage vessels, the system is entirely within containment (see Figure 11-1). Functionally, the system can be separated into two parts--the overflow/makeup circuit and the drain and storage components.

11.2.2.3.1 Overflow/Makeup Circuit

The primary sodium overflow vessel, makeup pumps, cold traps, overflow heat exchanger, and associated piping constitute the overflow and makeup circuit. With the exception of the overflow heat exchanger, which is normally bypassed, this circuit operates continuously during reactor operation. During operation, sodium overflows from the reactor vessel by gravity to the primary sodium overflow vessel. The primary sodium makeup pumps continuously pump sodium from the overflow vessel at a rate of 160 gpm. Of the 160 gpm, 60 gpm is returned (through a cold trap) to the reactor vessel, 10 gpm is pumped through impurity monitoring components (System 85) and returned to the overflow vessel, and 90 gpm flows directly to the reactor vessel. The reactor (gravity) overflow rate varies during primary system transients. During steady-state operation, the rate matches the makeup rate of 150 gpm. Nominal steady-state overflow temperature is 855°F. The overflow vessel functions as an expansion tank for the reactor and PHTS systems. The level varies with the primary system temperature.

The overflow vessel (gross capacity of approximately 36,322 gal) is sized to accommodate primary sodium volume expansion from 400°F isothermal to PHTS system design temperatures, plus the primary pump drawdown volume. During normal 100% plant power operation, the vessel is approximately one-half full. The overflow vessel must also accommodate the expansion of primary sodium from 400°F up to the peak temperature reached during the direct heat removal service (DHRS) operation discussed in Section 11.2.2.5. During DHRS, the sodium volume in the overflow vessel increases to a maximum of approximately 28,000 gal.

The overflow line is sized and sloped to permit a gravity flow of at least 650 gpm from the reactor vessel. During the DHRS function, the maximum overflow rate is approximately 610 gpm. The overflow line contains a section of 8-in. piping adjacent to the reactor and includes a raised loop. The peak elevation of the loop is at a height to control the sodium level in the reactor consistent with maintaining the minimum operating pool level at elevation 794 ft 5 in. The line is 6 in. in diameter from the midpoint in the downstream leg of the loop to the overflow vessel. The overflow design is discussed further in Section 11.3.1.2.2.

Two primary sodium makeup pumps are provided. Each pump is rated at 280 gpm, providing a total of 560 gpm for duty during reactor direct heat removal service (DHRS) operation. During normal plant operation, both pumps operate at 80 gpm to provide the required 160-gpm flow.

Two NaK-cooled sodium cold traps, arranged in parallel, are included in the reactor makeup return line to provide sodium purification. Each trap is rated at approximately 60 gpm at normal system operating temperature (approximately 850°F). During normal plant operation, one trap is in use and the second is on standby. Although both traps can be operated, a single trap is sufficient to maintain the primary sodium within established impurity limits of 2.0 ppm oxygen and 0.2 ppm hydrogen.

11.2.2.3.2 Sodium Storage and Draining

Permanent onsite primary sodium storage capacity is sized to permit complete drainage of the reactor vessel. To drain the reactor requires storage of 140,900 gal of sodium at 400°F. This capacity is provided by the total capacity of the overflow vessel, in-containment storage vessel, and two ex-containment storage vessels. The capacity of these vessels are:

	Gross	Net Usable
Overflow Vessel	36,322	31,902
In-Containment Vessel	36,000	34,530
Ex-Containment Vessel (2)	<u>105,000</u>	<u>100,560</u>
Total	177,322	166,992

The in-containment storage vessel is provided to permit drainage within the containment building of a single PHTS loop. Drainage of the PHTS loops is accomplished through drain lines connected to each of the loops at the low point of the primary pump and intermediate heat exchanger (IHX). All drain lines are routed to a common header which is connected to all storage vessels and to the overflow vessel. The drain lines are sized to drain a PHTS by gravity in 8 h. All drain lines connected to the PHTS loops are isolated by two locked-closed valves in series.

11.2.2.4 EVS Sodium Processing System

The principal function of this system is to cool the spent fuel in the EVST. A secondary function is to purify the EVST sodium. Working in conjunction with the primary storage and processing system (Section 11.2.2.3), this system also functions to provide a backup means of reactor decay heat removal. This is the DHRS function discussed in Section 11.2.2.5.

Three independent, physically separated, full-size cooling circuits are provided for EVST cooling. Two of these are identical forced circulation circuits, which are used as the normal cooling systems. The third circuit is a natural circulation circuit, which is used as a backup cooling system. Each of the three circuits is designed to remove the EVST design heat load of 1800 kW. At 1800 kW, either of the forced circulation circuits will limit EVST sodium temperature to 515°F; the natural circulation circuit (alone) will limit EVST sodium temperature to 610°F.

11.2.2.4.1 Normal EVST Cooling

That portion of the EVS Processing System provided for normal ex-vessel storage tank (EVST) sodium cooling and purification consists of two independent cooling circuits, either of which can remove the required EVST heat loads of up to 1800 kW. During normal operation, one circuit is used for EVST cooling and the other is on standby. Each of the circuits is composed of two loops, one a sodium loop and the other a NaK loop. The sodium loop circulates sodium from the EVST through a sodium-to-NaK heat exchanger (EVST sodium cooler) and back to the EVST. The NaK loop circulates NaK through the exchanger, where it picks up EVST heat, to a forced-draft airblast heat exchanger (ABHX) for dissipation of heat to the atmosphere, and back to the sodium-to-NaK heat exchanger. The system also includes a sodium cold trap which can be valved into either of the sodium loops to provide sodium purification. A diffusion cold trap is provided in each of the NaK loops for NaK purification.

During cooling of the EVST, simultaneous failure of both normal cooling circuits (by a single component failure) is precluded by operating only one of the circuits at a time and by the physical arrangement of the loop connections to the EVST. In the event of a circuit malfunction, the standby normal circuit can be brought into operation within 30 min.

The normal cooling circuit which is on standby is maintained full of sodium and NaK and, by the piping and equipment electrical heat control system (System 68), at temperatures of 400 to 485°F.

EVST NaK storage vessel is provided for filling or draining and storing the NaK. It has sufficient capacity to accommodate the entire inventory from either normal NaK loop plus the DHRS crossover piping (refer to Section 11.2.2.5) or the NaK from the third (backup) cooling loop.

11.2.2.4.2 Backup EVST Cooling

In the event of a double failure in the normal EVST cooling circuits, a third backup EVST cooling circuit is provided. This circuit is composed of a natural thermal convection sodium loop, which circulates sodium from the EVST through a sodium-to-NaK backup sodium cooler to a natural thermal convection NaK loop, which circulates NaK to a natural draft NaK-to-air heat exchanger (NDHX) and back through the backup sodium cooler.

The backup circuit is maintained in a preheat condition of 400 to 510°F by allowing a small induced flow to circulate in both the sodium and NaK loops. This small induced flow is caused by heat loss from the NDHX to the surrounding cell. Trace heating is provided to prevent sodium freezing when there are no heat-producing elements within the EVST. On a periodic basis, EVST cooling is manually transferred from a normal cooling circuit to the backup circuit to physically test its operability with a significant heat load of several hundred kW in the EVST.

11.2.2.5 Direct Heat Removal Service

One of the functions of System 81 is to provide a backup means of reactor decay heat removal in the extremely unlikely event that heat removal via the main heat transport system is lost. This System 81 operating mode is referred to as the direct heat removal service (DHRS). The design-governing requirement for System 81 is plant (faulted) event F-2 as defined in OPDD-10 (Ref. 3). This event involves a reactor trip from full power with simultaneous loss of all heat transfer through the Intermediate Heat Transport System 51B intermediate heat exchangers. Thirty minutes after trip, System 81 is to assume the reactor decay heat load and limit bulk primary sodium temperature to approximately 1140°F. DHRS is accomplished by using the combined heat removal capability of both of the normal NaK loops in the EVS Processing System and the overflow/ makeup circuit of the Primary Sodium Storage and Processing System. During DHRS, the EVST heat load is conservatively assumed

to be 1800 kW, with the EVST being cooled by one of the normal cooling circuits. In the DHRS configuration, the NaK ("crossover") piping connecting the EVST NaK loops with the overflow heat exchanger (OHX) is opened and both NaK pumps and airblast heat exchangers are turned full on. A combined total of 800 gpm (400 gpm each pump) is routed through the shell side of the OHX. On the primary sodium side, the makeup pumps are increased to a total flow of 560 gpm (280 gpm per pump), all of which is routed through the OHX and back to the reactor. The primary sodium cold traps are valved out of service. The coolant flow paths during normal plant/EVST cooling operation and during DHRS operation are shown in Figures 11-2 and 11-3, respectively. DHRS is manually initiated by operator action from the main control room. Following manual "start," the sequencing and operation of all valves and equipment is automatic.

11.2.2.6 Primary Cold Trap NaK Cooling System

The sole purpose of this system is to cool the primary sodium cold traps. The traps are cooled by circulation of NaK through a jacket on the trap crystallizer. The system is a copy of the same FFTF system. The combined storage/expansion tank and the NaK cooler, used to transfer heat to the chilled water System 23, are identical to the FFTF components. NaK is used in this system because it is liquid at system operating temperatures and is compatible with sodium. Heat is rejected to a small Dowtherm J loop (System 23) which provides a mutually compatible buffer between the NaK and the cooling water system used for ultimate heat rejection.

The system consists of one cooling loop which provides NaK coolant to both primary sodium cold traps. During normal plant operation, when one primary sodium cold trap is in use, NaK is circulated through the cooling loop at a flow rate of 30 gpm and a cold leg temperature of approximately 125°F. The NaK flows from the cold trap NaK pump and up through the jacket of the primary sodium cold trap where the NaK temperature rises to approximately 270°F. From there, it passes down through the tubes of the primary cold trap NaK cooler where it is cooled by Dowtherm J back to 125°F, into the storage vessel, and then back to the pump.

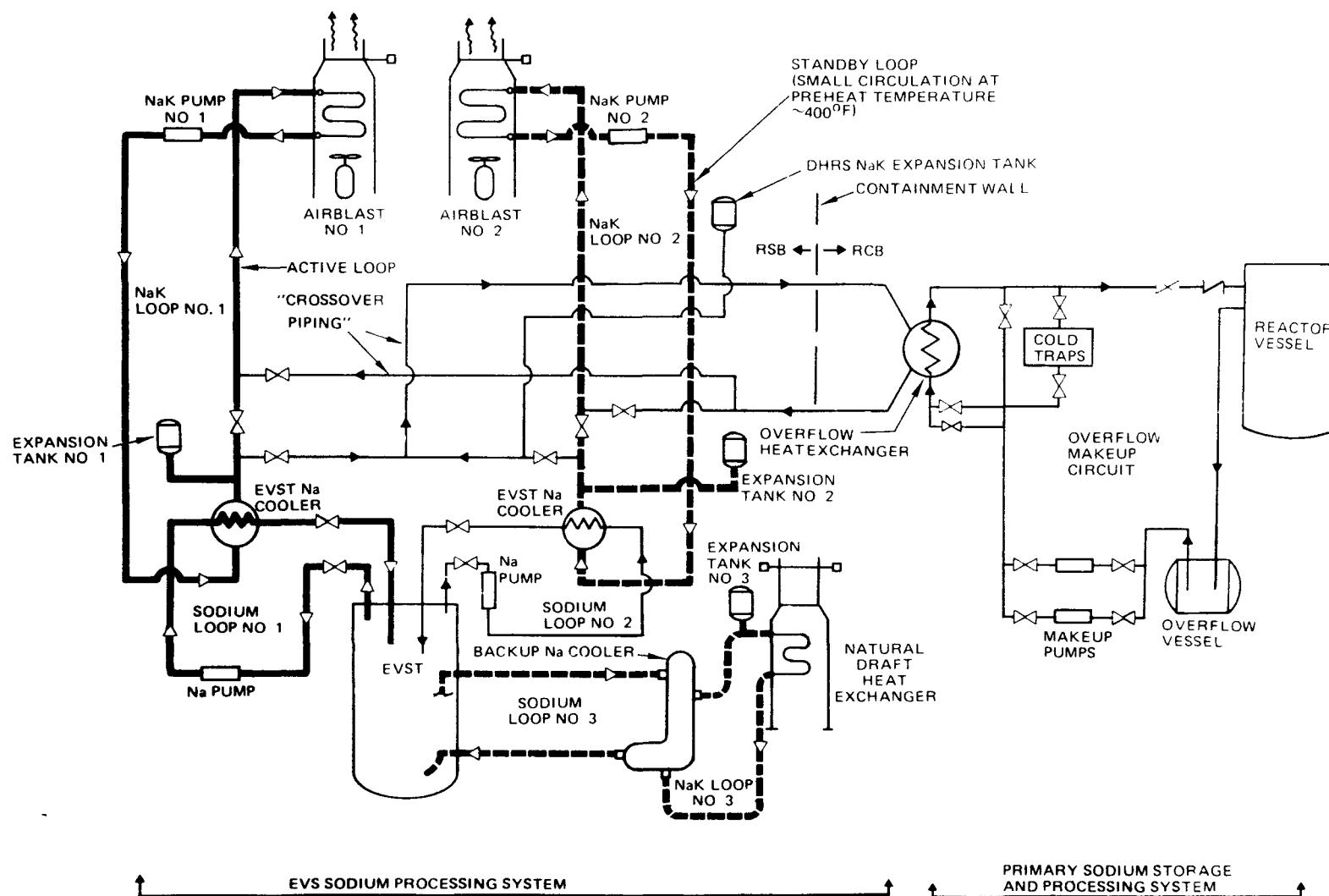


Figure 11-2. Sodium-NaK Flow During EVST Cooling Operation

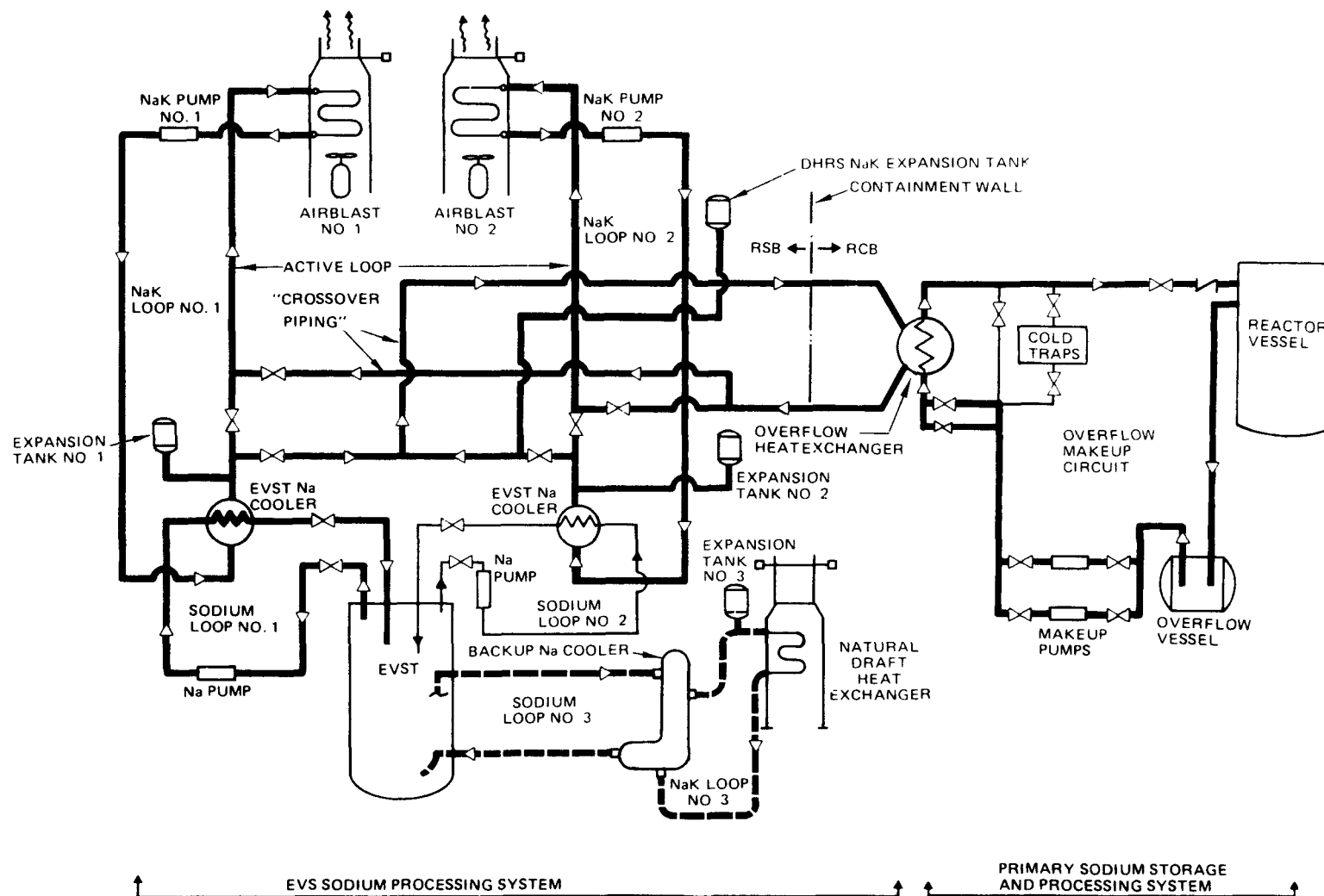


Figure 11-3. Sodium-NaK Flow During EVST-Plus-Reactor Decay Heat Removal

The primary cold trap NaK cooling system is sized to provide the capacity to cool two primary sodium cold traps simultaneously (total flow rate of approximately 60 gpm). The maximum total heat transferred from two traps is approximately 256 kW. The NaK flow through each trap is regulated by a control valve and a temperature-sensing element in each cold trap crystallizer.

11.2.2.7 Intermediate Sodium Processing System

The principal function of the intermediate sodium processing system (ISPS) is to purify the sodium in the System 51B IHTS heat transport loops. A secondary function is to fill the IHTS loops from the steam generator System 53 sodium dump tanks using the System 81 pumps and piping. The principal burden on System 81 is removing hydrogen that diffuses into the sodium through the walls of the steam generator tubes and is collected in the cold traps.

The system consists of three identical configurations, one for each IHTS loop, consisting of an EM pump, two 60-gpm air-cooled cold traps, and piping connecting these components to one another and to the associated IHTS loop. A typical loop is shown schematically in Figure 11.1

During normal operation, sodium from each IHTS loop (System 51B) is cold trapped by an independent ISPS circuit (System 81). A total of 123 gpm is withdrawn from the IHTS loop near the IHTS pump suction, of which 60 gpm is pumped by the process EM pump through each of the loop's two cold traps and 3 gpm through the sodium sampling station (System 85). The three flows recombine and return to the IHTS loop by way of the IHTS loop expansion tank. The sodium of all three IHTS loops is processed independently in the same manner.

Each of the two cold traps in a loop is capable of processing 60 gpm at normal IHTS cold leg temperature of 650°F. The EM pump is rated at 125 gpm to permit simultaneous operation of both traps. The operation of one trap in each IHTS loop is sufficient to remove anticipated oxygen and hydrogen in-flux and maintain the level of these impurities below the maximum permissible values of

2.0 ppm and 0.2 ppm, respectively. However, both traps in each loop are operated to minimize the hydrogen concentration and diffusion into the primary heat transport system (PHTS), which maximizes the life of the primary cold traps. Operation of these cold traps is discussed more fully in Section 11.3.3.3.

11.3 Evolution of Design

The design of System 81 was initiated in March 1973, as was that of all other major nuclear island systems and buildings. The immediate emphasis was on the systems and components having significant impact on plant arrangement, particularly the containment building. System 81 design proceeded through conceptual and preliminary design with completion of preliminary design in November 1977 for all of System 81 except for the EVST natural circulation loop. This loop was being added at this time, and a separate preliminary design review for this loop only was held in October 1978. The overall design then proceeded into final design, with the final system design review held in May 1983. There was no formal conceptual design review; conceptual design was considered complete following the initial release of Rev. 0 of the system design description, SDD 81.

11.3.1 Conceptual Design

11.3.1.1 Initial System Concept

The initial conceptual design of CRBRP System 81 was based directly on the auxiliary liquid metal system design developed for the Fast Flux Test Facility (FFTF). The first cut at the overall design is shown in Figure 11-4. The name and general function of each of the five individual systems were taken from FFTF. Key features of the system are discussed below; some of the significant changes during the conceptual design are discussed in Section 11.3.1.2.

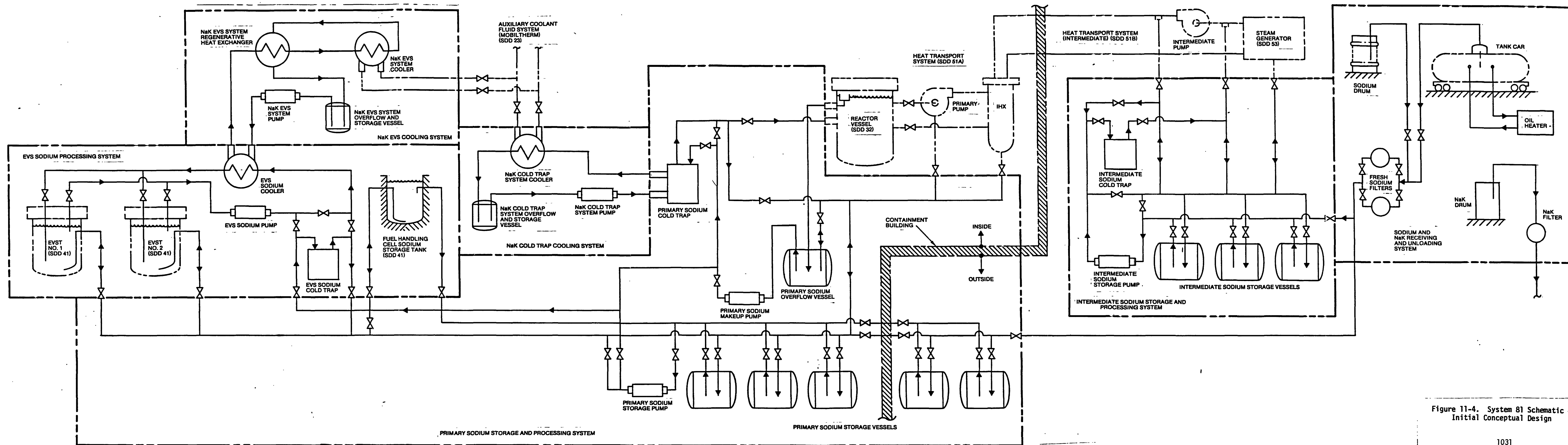


Figure 11-4. System 81 Schematic - Initial Conceptual Design

11.3.1.1.1 Overflow/Makeup Circuit

One of the first design decisions made was to use a reactor vessel the same diameter as FFTF. It was also decided not to lengthen the vessel enough to permit accommodation of sodium volumetric changes within the vessel because of impact on building structure and on the required additional length of fuel handling components. Because an external expansion tank was needed, the FFTF overflow/makeup circuit, including cold traps, was adopted. Since the CRBR's thermal conditions were no more severe than those of FFTF and since the gross capacity was considered adequate, the FFTF overflow vessel design was used, as is, for CRBRP. The EM pumps and primary sodium cold traps were initially assumed to be identical to FFTF components. The makeup rate was not firmly established, but the original thinking was a rate in the order of 325 gpm. This rate was based on an attempt to maintain at least some small gravity overflow immediately following a reactor trip in which a rapid contraction of sodium occurs in the reactor vessel. This thinking was driven by the FFTF overflow concept (originally adopted) which withdraws overflow sodium directly from the reactor upper pool. This in turn results in high steady-state operating temperature, severe thermal transients on the overflow line and vessel during a reactor trip, and very substantial design and thermal/stress analysis problems.

11.3.1.1.2 Sodium Storage Capacity

The initial concept regarding sodium storage was to provide permanent storage capacity to permit drainage of all primary sodium (reactor plus all PHTS loops) or the EVST. The existing overflow vessel design was to be used as the drain tank design as discussed in Section 11.3.1.2. This concept was revised during conceptual design.

11.3.1.1.3 NaK Cold Trap Cooling System

This system was taken, without change, from FFTF. With the exception of the detailed pump design, this system has not changed.

11.3.1.1.4 Intermediate Sodium Storage and Processing System

This system, consisting of an independent purification circuit for each of the intermediate heat transport loops, has not changed in concept since design inception. The only significant changes involved the operation of the cold traps (discussed in Section 11.3.3.3) and the function of the drain tanks.

The drain tanks in each processing circuit, sized to drain an IHTS loop, were initially part of System 81. When the decision was made to use these tanks for rapid drain in the event of sodium/water reaction due to steam generator failure, the tank requirements became more complex and the tanks became part of the steam generator system (System 53); drainage was deleted as a function of System 81.

11.3.1.1.5 EVS Sodium Processing and Cooling Systems

The principal feature of the initial ex-vessel storage (EVS) system was the provision of two ex-vessel storage tanks (EVST). The combined maximum heat load at that time was 1300 kW, and EVST cooling was accomplished by the NaK EVS cooling system, in which EVST sodium was cooled by a NaK system, with heat rejection to Mobiltherm (System 23). The cooling system was a copy of the FFTF system. A small sodium storage vessel (for spent fuel) was located within the fuel handling cell.

11.3.1.1.6 ASME Code Classes

During the later stages of conceptual design, considerable discussion was held with the Nuclear Regulatory Commission (NRC) on the subject of codes and standards. With the exception of the Class 2 primary drain lines, the final result shown in Table 11-1 generally reflects NRC's position (Ref. 4). The Atomics International/project position was less conservative; Code Class 2 was proposed for the overflow/makeup circuit and Code Class 3 for the EVS sodium circuits. NRC disagreed. Following the introduction of the DHRS (see Section 11.3.1.2), the EVS NaK circuits became Code Class 2 because of their DHRS function, otherwise the NaK circuit would probably have been Code Class 3.

The NaK cold trap cooling system, a non-safety-class system, was upgraded by project choice to Section III Class 3 because the cost penalty was not significant and also as a hedge against NRC ultimately not agreeing to the "non-safety-class" system category. (NRC had serious concerns [which the project did not share] with the safety implication of loss of primary cold trap cooling, regarding the reintroduction of large amounts of soluble impurities into the primary sodium system.) The primary drain lines inside containment were initially (until well into final design) ASME Code Class 3, but were revised to Class 2 after a review of containment isolation concluded that these lines could be considered a "closed system" inside containment and therefore, per NUREG 800 SRP 6.2.4, should be Class 2. Although the intermediate sodium purification circuits perform no safety function per se, these circuits are considered safety Class 3 (and therefore ASME Class 3) because of the importance of pressure boundary integrity.

11.3.1.2 Significant Changes During Conceptual Design

During the duration of conceptual design, several significant changes were made to the overall system design and are discussed in this section. At the conclusion of conceptual design, the system design had evolved schematically from that shown in Figure 11-4 to that shown in Figure 11-5.

11.3.1.2.1 Direct Heat Removal Service (DHRS)

The function and operation of DHRS is discussed in Section 11.2.2.5. Prior to its adoption, reactor decay heat removal was accomplished solely via the three main heat transfer loops and associated steam generators. While the project felt strongly that this design, along with the Steam Generator Auxiliary Heat Removal System (SGAHRS), provided adequate protection against a credible loss-of-cooling event, it was considered prudent to provide an independent means of decay heat removal. Considering the commonality of design (particularly of the steam generators), the addition of DHRS was believed to enhance overall plant licensability. Following a major trade study involving various heat removal concepts, the design shown in Figure 11-5 was selected. Of the

alternatives considered, most of which involved the addition of a loop directly connected to the reactor or to the IHTS loops, this configuration was chosen because (1) it had the lowest overall cost, (2) it had minimum impact on the reactor vessel and containment, and (3) the existing FFTF dump heat exchanger design could be used for the EVST airblast heat exchangers. The dependency of DHRS operation on maintenance of reactor sodium level was considered an acceptable disadvantage because of the unlikelihood of complete loss of heat sink coupled with a PHTS leak.

11.3.1.2.2 Reactor Overflow Concept

As noted in Section 11.3.1.1.1, the original reactor overflow concept involved the gravity flow of reactor sodium from the hot, upper pool down to the overflow vessel. During the conceptual design, the concept was revised to withdraw reactor overflow sodium from the annulus between the reactor vessel and thermal liner. This flow, used to cool the reactor vessel shell, is considerably cooler than the upper pool temperature. The initial concept involved a weir within the reactor vessel, with sodium supplied from the annulus. This is depicted in Figure 11-5. Further iteration resulted in the final design, in which the reactor overflow nozzle was lowered and the overflow sodium withdrawn directly from the annulus. The overflow line is configured such that it is full up to a high point, at which point the overflow "free-falls" down to the overflow vessel. The location of the high point establishes the reactor sodium level. This design change was very significant, from a thermal/structural standpoint, for two reasons: (1) it lowered the overflow/makeup circuit design/operating temperature from 1015°/995° (pool temperature) to 900°/approximately 850° (annulus temperature) and (2) it maintained submergence of the overflow nozzle at all times, thus minimizing transient thermal shock at the nozzle.

11.3.1.2.3 EVST and EVST Cooling

During the conceptual period, a single EVST was adopted for spent fuel storage. The cooling system was completely revised due to the adoption of the

DHRS function discussed in Section 11.3.1.2.1. At this time, EVST cooling was provided by the two identical sodium-to-NaK forced circulation circuits shown in Figure 11-5.

11.3.1.2.4 Primary Sodium Storage

The requirement for primary sodium storage was reduced to the capacity required to drain the reactor vessel. In-containment capacity was provided to drain only a single PHTS loop. The design of the storage vessels was changed from the FFTF overflow vessel design to a more conventional (and economical) saddle-mounted design. Carbon steel was adopted for both in-containment storage and ex-containment storage vessels. A single 100,000-gal tank was initially considered for ex-containment storage.

11.3.2 Preliminary Design

This section discusses the significant system and component design changes during the preliminary design phase extending to November 1977 for general system design and to October 1978 for the EVST natural circulation loop.

11.3.2.1 EVST Natural Circulation Loop

The most significant system change during preliminary design was the addition of the natural circulation loop (also referred to as "third loop") for EVST cooling. The previous design included only the two redundant forced circulation circuits. Although reliability studies (Ref. 4) had convinced AI and the project that the existing cooling circuits provided adequate protection against loss of EVST cooling, the third loop was added because (1) the existing loops did not comply with the double failure criteria (passive followed by active failure) and (2) there were general concerns relative to potential common mode failure, particularly in view of the identical component designs for both loops. Also, during this time period, the maximum design EVST heat load was increased to 1800 MW from 1300 kW.

11.3.2.2 Fuel Handling Cell Sodium Storage Tank

The method of handling spent fuel was revised to eliminate the small sodium vessel within the fuel handling cell (FHC). This resulted in elimination of the sodium circuit required for circulation, purification, and cooling the sodium. It is also of interest to note that the periodic small spent fuel heat load (the tank held approximately six elements) within this tank was to have been removed by the EVST cold trap, resulting in an ASME Class 2 category for the cold trap and FHC piping circuit. With the deletion of the tank, the only true cold trap safety function (other than pressure boundary integrity for the radioactive sodium) was eliminated. The ASME Code class was reduced to Class 3.

11.3.2.3 Ex-Containment Primary Sodium Storage Vessel

The single 100,000-gal ex-containment primary sodium storage vessel was replaced by two identical vessels, each of approximately 50,000-gal capacity. The single long (approximately 18 ft diameter by approximately 55 ft long) vessel resulted in a design from a seismic loading standpoint, which was economically impractical.

11.3.3 Final Design

Final design extended from the end of preliminary design up to the project termination in November 1983. Final system status and work remaining is discussed in Section 11-5. During final design, no major physical system design changes were made. Physical changes were generally limited to component design details and isolated piping and valving changes such as the deletion of valves for cost reduction and the simplification of piping around the System 53 sodium dump tanks. The bulk of the effort was devoted to detailed systems analyses, component design and procurement, piping design and stress analysis, and generation of operating and maintenance procedures. Some of the more significant changes are discussed in this section.

11.3.3.1 DHRS Modifications

Most of the DHRS effort during final design involved detailed systems analysis and development of procedures. The peak DHRS total heat load dissipated by the two ABHXs had finalized at 10.2 MW (8.5 MW from the reactor, 1.7 MW from the EVST) from an original value of 8 MW. Two modifications are worth noting--one incorporated, one potential.

The adopted modification involved analyses and changes to ensure DHRS operation despite failure of a single active DHRS component. The project committed to this capability at the request of NRC. Previous sodium tests of the EM pump, prototypical of both the primary sodium makeup and the EVS NaK pumps, had shown that a single pump, as designed, was capable of providing 600 gpm of primary sodium and 500 gpm through one (only) of the EVS NaK circuits. With these flows and the same heat exchangers, thermal analysis indicated satisfactory DHRS operation. Consequently, DHRS capability is maintained despite loss of a makeup pump and/or one of the NaK (ABHX) circuits. Only three physical changes were required to satisfy the single active failure requirement. The first was to add a redundant valve at the primary sodium inlet to the overflow heat exchanger; otherwise, failure of the single existing valve to open would preclude DHRS operation. The second was to add operators to the suction and discharge valves around each of the makeup pumps to prevent backflow through a pump that may have failed during DHRS initiation (one of the DHRS requirements was that all operations be done from the main control room; local manual valve operation is not acceptable). The third modification was to include a physical stop to limit the opening of the cold trap/DHX bypass valve. This valve would be normally throttled and, as long as it was not accidentally opened further during DHRS startup, the bypass was acceptable since the bypass flow was insignificant in relation to total pump flow.

The potential DHRS modification, which was not yet adopted but still under consideration, was to revise the DHRS flow path such that the makeup pumps would be supplied by sodium from the PHTS loops via the drain piping, thus avoiding use of the overflow line. The impetus for this capability was NRC's

continuing, and unresolved, concern (Ref. 6) that DHRS operation was not really independent of the primary loop because a primary sodium leak which lowered the reactor sodium level would preclude DHRS operation (no gravity overflow). A design study showed that with the existing drain piping configuration, this method of DHRS was possible, with qualifications. The existing drain line sizes limit pump suction flow to approximately 200 gpm for single PHTS loop drain and 300 gpm for two PHTS loop drains. Since these flows are considerably less than the 560-gpm DHRS design flow, the resulting primary sodium temperature would be increased unless DHRS initiation could be delayed until approximately 4 h after shutdown. The principal physical impact on system design to implement this capability would have been the addition of operators on various drain valves to permit remote operation from the control room. The adoption of this modification was delayed pending further discussions with NRC.

11.3.3.2 Design for Sodium Spills

A major effort during the final system design was the establishment of design base liquid metal spills for each of the System 81/85 cells containing sodium components and system modifications necessary to accommodate the spills. This effort was concluded and implemented for all inerted cells. For air-filled cells, spills had been established and detailed design revisions were being evaluated at the time of project termination (refer to Section 11.5.3). Design spills from piping were, in all cases, based on the Moderate Energy Fluid System (MEFS) leak rate defined in OPDD-10 (Ref. 3). The leak rates and spill volumes for both inerted and air-filled cells are documented in Ref. 7. With the exception of the small cells containing System 85 impurity monitoring components, the design spill in all inerted cells was based on an unmitigated spill, e.g., no action taken to stop the leak. For the System 85 cells, which contained features such as shielding windows and transfer tunnels not designed to seal against liquid metal, the leak is stopped by automatic valve closure. Additional safety-related valves and operators were required to provide leak mitigation; dual valves were required to satisfy single-active-failure criteria. A major consideration during the inerted cell studies was to locate

all valves required for leak mitigation in cells not subject to sodium aerosols generated by the leak being mitigated. This was to avoid the need for valve operability tests in an aerosol environment. A major revision of the Recirculating Gas Cooling System 28 cell cooling circuitry in the reactor service building was made to accomplish this. Following the inerted cell spill studies and associated design changes, it was assumed that valve aerosol/operability problems had been eliminated; at the time, this was not considered to be a problem in the air-filled cells. As studies in the air-filled steam generator building (SGB) progressed, this assumption appeared false as discussed in Section 11.5.3.

11.3.3.3 IHTS Cold Trap Operation

The initial concept of IHTS sodium purification was to have two 60-gpm cold traps for each of the three IHTS loops. One trap was to be operated with the other on standby. As the design of the steam generator system progressed, the rate of hydrogen diffusion into the sodium via the steam generator was found to be approximately seven times the original estimate. A significant design study discussed in Ref. 8, was undertaken to assess the impact of the higher hydrogen burden on the purification system design. The principal results, documented in Ref. 9, showed that operation of a single trap per IHTS loop could still maintain acceptable hydrogen concentration in the sodium (≤ 0.2 ppm). Because of hydrogen diffusing back into the primary sodium via the IHX, however, operation of a single trap reduced the lifetime of the primary cold trap by a factor of two over the lifetime achieved if both IHTS traps were operated continuously. The estimated 13-year primary trap life noted in Section 11.3.3.5.2 is based on continuous operation of both IHTS traps. The higher in-flux of hydrogen resulted in a reduction of the IHTS trap life to approximately 3 years, based on dual trap operation and the 10% capacity factor discussed in Section 11.3.3.5.2. Even with the admittedly conservative lifetime estimate, a substantial number of replacement traps would be required over the 30-year plant life. As part of the design study noted above, a conceptual design of a portable regeneration skid was developed to permit in-place cold trap regeneration and its adoption as part of the

plant design was recommended. The final decision regarding regeneration, implemented by Engineering Change Proposal (ECP) A81262 (Ref. 10), was to provide for its future installation (space, access, electric power, pipe stubs for connection to the sodium piping) but defer actual design and installation until early in plant life, when a better assessment of actual trap life could be made, and also to gain maximum benefit from on-going Argonne National Laboratory (ANL) cold trap development efforts.

11.3.3.4 Piping Design

Piping design and stress analysis was an on-going activity from the conceptual phase through final system design; the final piping design/analysis was still under way at project termination. The principal effort during the final phase was devoted to the detailed analysis necessary to satisfy ASME code requirements and the design/analysis due to system changes which resulted in valve or piping revisions. One point worthy of note is the project agreement that for small insulated piping, seismic damping values of 3% for Operating Basis Earthquake (OBE) and 6% for Safe-Shutdown Earthquake (SSE) could be used, rather than the previously used values of 1% and 2% suggested (but not mandated) by Regulatory Guide 1.6.1 (Ref. 11). Although not completely implemented, this change was expected to be very beneficial, both in reducing seismic stresses and component nozzle loads (in some cases making the difference in satisfying ASME code or component structural limits) and in potentially reducing the number of piping seismic restraints. The background data used to justify using the higher damping values are given in the attachment to Ref. 12.

11.3.3.5 Component Development

11.3.3.5.1 Electromagnetic (EM) Pumps

The original plan for all System 81 EM pumps was to use the FFTF design. Consequently, the overall pump configuration, i.e., a rectangular box with sodium inlet and outlet at the same end, became the basis for the equipment and piping arrangement; thus, the general pump configuration became fixed early in

the design. The pump design itself was revised to incorporate improvements in the FFTF design. The pump design is discussed in more detail in Section 11.4.2.

11.3.3.5.2 Sodium Cold Traps

11.3.3.5.2.1 Primary Cold Traps

The initial intent was to duplicate the FFTF primary cold trap design for CRBRP. This is a 60-gpm NaK-cooled trap with a separate economizer and crystallizer. The economizer is a toroidal heat exchanger located in a horizontal plane immediately above the crystallizer. Initial testing at FFTF indicated excessive ΔT s within the economizer due to natural circulation flow resulting from very low pressure drop within the unit. In order to avoid this problem, the CRBRP design was revised to incorporate a straight shell and tube economizer mounted vertically above, and attached directly to (without interconnecting piping), the crystallizer. The overall FFTF crystallizer configuration (modified at the top as required to accommodate the integral economizer) was maintained; however, as the detailed design progressed, the inner cylindrical flow baffle was resized to increase the flow area in the outer annulus (sodium downflow) relative to the central section (sodium upflow). This was done to increase the trap impurity capacity and thus design life. The change was based on cold trap studies being done by Argonne National Laboratory (Ref. 13). It is worth noting that the current CRBRP cold trap lifetime calculations (Ref. 9) are based on the assumption that 10% of the trap mesh volume is available for trapping; discussions with ANL indicated that the baffle modifications could potentially double or triple this estimate. Considering that the primary trap life is estimated at approximately 13 years (based on hydrogen/tritium trapping only; initial oxygen cleanup would use on the order of 1/4 to 1/3 of the first trap), it is entirely reasonable to assume that, as modified, the two traps provided would last through the 30-year plant design life.

One final change in the primary trap design occurred late in the final design: an integral shield was added to both the primary and the EVST sodium traps. This change resulted from a Nuclear Island General Purpose Maintenance Equipment System 44 study which concluded that the economics favored an integral shield over the cost of the remote maintenance platform, remote tooling, and dedicated cask required for handling and removal of an unshielded radioactive trap. This change and associated background information was implemented by ECP A81265 (Ref. 14).

11.3.3.5.2.2 EVST and IHTS Cold Traps

Early in the design, the decision was made to use a common trap design for both the EVST and IHTS cold traps. This resulted in an oversized EVST trap; however, this is less costly than generating a different design. A cursory evaluation showed that air cooling the six IHTS traps was much less expensive than liquid (NaK) cooling. After analysis had established that a single 60-gpm trap would maintain a satisfactory hydrogen limit (0.2 ppm), the crystallizer design was generated by using the FFTF design, enlarged as required for air cooling. The economizer design is the same as for the primary traps except that the economizer is separate from the crystallizer. The components are separated for economic reasons--the relatively short IHTS trap life (approximately 3 years assuming the 10% use factor discussed above) and the fact that only the crystallizer need be replaced following trap saturation.

11.3.3.5.3 Pipe Clamps

The original pipe clamp concept chosen was a conventional three-bolt clamp attached directly to the pipe surface. As the design progressed, two problems surfaced resulting in a change in clamp concept. The first was thermal stress; the clamp was not satisfactory for use where repetitive sodium thermal transients exceeded 1 to 2°/s. The second was inservice inspection (ISI); much of the piping is subject to ISI requiring visual inspection of the clamp

and bolts. This leads to costly insulation modules and time-consuming inspection. The bulk of System 81 piping is subject to either, or both, of these problems. Consequently, the concept was revised to use of an insulated clamp, a more detailed discussion of which is given in Section 11.4.1.

11.3.3.5.4 Valves

A single all-purpose valve design was used for all small liquid metal systems throughout the plant. The valve, a Y-pattern, bellows-sealed, globe valve, was based on that designed for FFTF. The valve design and development and modifications adopted are discussed in Section 11.4.3.

11.4 Key Problems and Resolutions

11.4.1 Pipe Clamps

11.4.1.1 Introduction

As originally conceived for the CRBRP program, all small (1-in. through 8-in.) piping supports in the Systems 81, 82, and 85 were to be hot, bare strap-on (HBSO) clamps with the exception of a few lines in the primary overflow system. It was subsequently found that the choice of the HBSO clamp was unacceptable for stainless steel Schedule 40 piping subject to thermal transients in excess of 1°F/s. Data indicated that for a 6-in. Schedule 40 pipe under a clamp loaded condition, at thermal transients in excess of 1°F/s and less than 4°F/s, the pipe experienced severe strain ratchetting beyond its elastic limit. This was due to the stresses induced in the pipe by thermal transients in combination with nominal bending stresses at the clamp. Because there were a number of pipelines in Systems 81 and 85 that experienced thermal transients in excess of 1°F/s, a different pipe clamp design was required. The FFTF pipe clamp design was considered as an alternative. The FFTF clamp test data and analysis were reviewed. From this analysis, it was determined that the loads and stiffness of the FFTF pipe clamp, if used, would impose a significant penalty by requiring rerouting and reanalysis of the System 81 and 85 piping system.

Other options were considered involving different mixes of bare and insulated pipe clamps. These were: (1) a thermally insulated pipe clamp for thermal transient pipelines in excess of 1°F/s where remote inservice inspection was a consideration (all other pipelines would utilize HBSO clamps), (2) a thermally insulated pipe clamp for thermal transient pipelines in excess of 1°F/s, plus all clamps requiring inservice inspection in the reactor containment building (all other pipelines would have the HBSO clamp), and (3) all pipe clamps would be thermally insulated.

A cost analysis was performed on all three approaches, and it appeared that there was little difference in the estimated costs between the options. Option 3 was preferred since it would permit the use of a single clamp design, which would have obvious advantages in procurement, warehousing, and site-related activities. Furthermore, only one development test program, rather than two, would be required to qualify the clamp for the CRBRP program.

A commercially available, thermally insulated pipe clamp was found. The clamp had been designed and was being sold by Pipe Shields Incorporated, San Jose, California. The pipe shield design incorporates a layer of load-bearing insulation (Marinite-P) between the pipe and the clamp, which thermally decouples the pipe from the clamp. Induced local pipe stresses due to the pipe thermal expansion are reduced because the load is distributed over a wider area. In addition to the technical considerations for using the pipe shield clamp, the device provided advantages from an ALARA standpoint. The bare, strap-on clamp required the use of removable insulation modules to permit inspections. The pipe shield clamp, being outside the insulation, provided reduced inspection times for the inservice inspections.

For these reasons, the pipe shield concept offered an attractive alternative in light of the existing problems with the HSB0. Preliminary screening tests of the pipe shield system indicated load capacity and stiffness values equivalent to, or better than, those values used for the bare clamp.

To qualify the pipe shield system for CRBRP, a test program was designed. This program was to determine the performance capabilities of the clamp/insulation concept and qualify the clamp system for CRBRP use. To this end, a Development Requirements Specification (DRS) was prepared, submitted, and approved by the CRBRP Project Office (DRS 81.12).

11.4.1.2 Test Program Objectives

The test program was divided into groups of tests carried out in most cases for 1-in. and 8-in. pipes. These sizes were chosen to umbrella the range of piping system sizes for which the pipe shield clamp would be used. The detailed objectives of the test program were as follows:

- Determine the stiffness and load-carrying capabilities of pipe shield clamp assemblies for 1-in. through 8-in. pipes
- Evaluate the load and stiffness characteristics of the pipe shield clamp when subjected to long-term temperature cycles
- Establish capability of the clamp to withstand dynamic loads at room temperature, at elevated temperature, and after the pipe clamp assembly has been subjected to long-term thermal cycles
- With the clamp attached to a length of pipe, determine any shift in resonance frequency of the pipe after the pipe clamp assembly has been subjected to temperature cycling
- Determine local stresses in the pipe under and in the vicinity of the pipe shield clamp due to pipe and clamp interaction
- Establish the capability of the pipe shield clamp assembly to resist rotation under a skewed compressive load, with the pipe at room temperature, at elevated temperature, and after undergoing thermal cycling

- Determine compressive bearing and flexural strength of Marinite-P material after long-term exposure to elevated temperature
- Perform stress corrosion tests and sodium compatibility tests to determine acceptability of Marinite-P insulation material
- Determine effects of nuclear radiation on pipe clamp insulation and bonding material
- Develop Marinite-P material design properties.

On the basis of the test objectives, the following tests were developed into four broad groups to completely characterize the pipe shield clamp.

- Stiffness and Load-Rating Tests
 - A1 Clamp Stiffness Test
 - A2 Clamp Load-Rating Test
 - A3 Compressive Load-Rating Test
 - A4 Axial Load-Carrying Capacity Test
- Clamp Performance Tests
 - B1 Antirotation Test
 - B2 Vibration and/or Seismic Test
 - B3 Thermal Cycling Tests

- Pipe Clamp Interaction Tests

- C1 Strain Profiles

- Pipe Clamp Insulation Tests

- D1 Sodium Compatibility Test of Marinite-P
- D2 Stress Corrosion Test and Chemical Analysis Test of Marinite-P
- D3 Radiation Exposure Test of Marinite-P, including Complete Pipe Shield Assembly
- D4 Rupture Strength in Bending of Marinite-P
- D5 Material Design Properties of Marinite-P

11.4.1.3 Description of the Test Articles

The test articles consisted of a series of insulated pipe clamps for use on CRBRP small-diameter sodium and sodium vapor pipes 1 in. through 8 in. Figure 11-6 illustrates a typical clamp assembly (1-in. pipe) with the major components identified.

The Marinite-P insulation, a nonasbestos thermal insulation, is manufactured in 1.5-in.-thick sheets by Johns-Manville Corporation. The pipe shield assembly was fabricated by Pipe Shields Incorporated, San Jose, California. A typical pipe shield contains 16 segments of insulation assembled into 4 quadrants bonded into 2 half-shells to facilitate ease of assembly. Slots were provide in the insulation ID for pipe heaters, thermocouples, and other instrumentation.

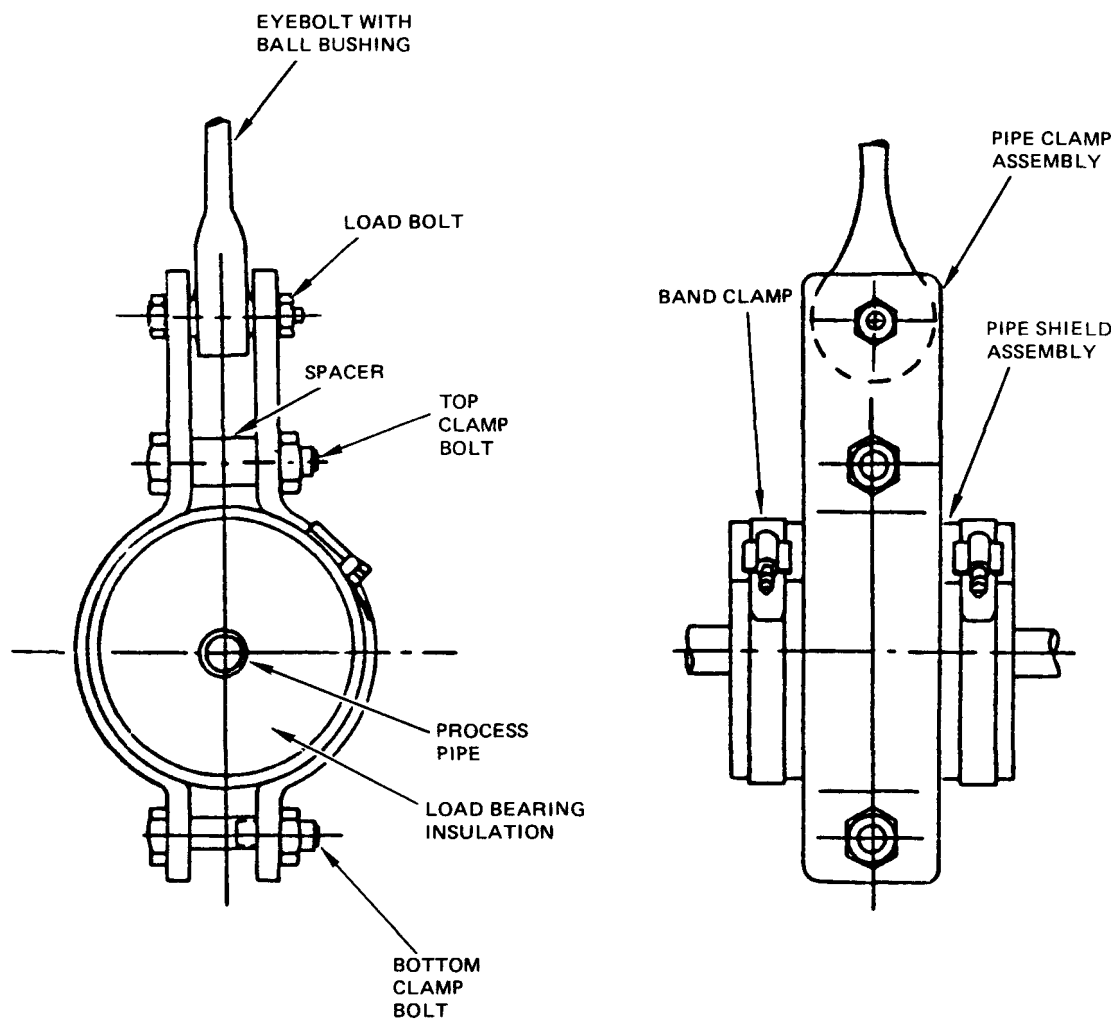


Figure 11-6. Small Insulated Pipe Clamp Assembly

A project rule established for the clamp/insulation system was that this device be procured to "commercial standards" in order to reduce costs. Tolerances were allowed to be "loose" without extra precautions for fit between pipe and insulation and clamp radius.

All tests were conducted on heat-treated material from several lots. Heat treatment of Marinite-P material was considered necessary to stabilize its mechanical properties and drive out any residual water in the material from the fabrication process.

The test series provided vital information for use in the design of an insulated pipe clamp support system. It is apparent that the test series also raised some questions that should be addressed by further design and testing before a clamp of this configuration is used for future plants. The following is a summary of the test series recommendations for future design and testing. A complete description of the test and test results is described in report ESG-DOE-13427 (Ref. 15).

11.4.1.3.1 A1 Test Series

These tests did provide stiffness values that met the CRBRP piping design criteria. However, it is felt that better control of the pipe shield and clamp fitup would provide increased stiffness values and reduce the unit loading on the Marinite-P insulation material. This would reduce loss of preload.

It is also recommended that consideration be given to the use of Belleville washers to the clamping bolt to take up minor preload loss due to the misfit of the pipe shield and some insulation material consolidation under preload. Additionally, a tolerance study should be made of the pipe shield/clamp assembly. Clamps and shields encompassing the extremes of a tolerance should be tested to demonstrate and verify the expected results. Especially critical are the uniformity of the bore diameter and the flatness of the clamp strap.

11.4.1.3.2 A2, A3 Test Series

These tests indicated that, for the 1-in. and 8-in. clamps, the design was adequate to provide the necessary margins required by Subsection NF of the ASME Code. These tests should be repeated with newly toleranced clamps, as in the A1 series, using clamps that have been used in the new A1 series and performed on 1-, 4-, and 8-in. clamp systems for both the A2 and A3 series. The 8-in. clamp slipped consistently during the skewed test (A3). A better fit of the clamp and increased sustained preload would provide improved results.

11.4.1.3.3 A4 Test Series

These tests did provide data on the resistance of axial slip of the clamp. Again, because of fitup of the large clamp (8 in.), this test should be repeated for the 8-in. pipe size to determine the expected improvement provided by a better fitup.

11.4.1.3.4 B1, B2, B3 Test Series

The purpose of the test series was to determine the effect of thermal cycling and vibration of the pipe support system, namely:

- Would the insulation "pound out" under vibration?
- Would there be a loss of preload from the thermal cycling and vibration?
- Would there be rotation of the clamp during vibration/skewed tests?

The results indicated that there is no "pounding out" of the insulation material for either size. It was found that the clamps do lose preload (up to 40%), but this had no effect on the 1-in. pipe clamp system regarding slipage. However, the 8-in. clamp system did slip even though the preload had

been increased to 100 ft-lb. Preload loss was felt to be the result of fitup of the clamp to the pipe shield and material consolidation. The future tests should be modified to eliminate the interim vibration portion and merely perform some thermal cycling and skewed load tests on a clamp with better fitup. A vibration test should be conducted at the end of the thermal cycling series to verify no slipping of the clamp after temperature service.

11.4.1.3 5 C1 Test Series

Results of the 1-in. pipe clamp tests indicated negligible pipe strains. 4since the 8-in. test was not performed, it is believed it should be included in future testing. It is known that the pipe strain would have been greater since the piping has relatively thinner walls (Schedule 10).

11.4.1.3.6 D1, D2, D3, D4, D5 Test Series

The D1 through D3 series did not reveal any results that would disqualify the material from use. The D4 series provided spurious data due to test techniques. However, the data obtained in the D5 series (material properties) provided sufficient data to characterize the material.

The D5 test results did show some slight deviation from the published properties, with the most significant deviations being found in the mechanical strength properties.

Results of the heat-treated compressive strength indicate a reduction of approximately 20 to 50% from J-M published data for 1/2-in.- and 1-in.-thick material. Tests conducted on 1-1/2-in.-thick material showed that at 1% strain, a stress value of 400 psi is obtained for the five lots, whereas J-M data indicated a stress value of approximately 500 to 800 psi average for the two thicknesses. Test results would indicate that loading capacity of 50% less than advertised can be expected and should be factored in the design analysis of the clamp.

Aging tests conducted up to 500 h at 1050 and 1200°F indicated a loss of approximately 10% to 875 psi in ultimate strength for material aged at 1050°F for 250 h. Material aged at 1200°F for 500 h showed a loss in ultimate compression strength of 40% to 672 psi. These results were obtained from three samples each at the two temperature exposures. These samples were tested in plane, which is the direction of loading for the clamp. Values given are at approximately 1% strain and conducted at room temperature. These results indicate a potential long-term degradation of the material properties. It is recommended that further long-term aging tests be carried out on the material to understand the rupture strength and compression strength properties more fully. Samples should be aged at 1000°F for periods up to 1 year and tested at intervals of 250, 500, 1000, 5000, and 9000 h. The results would determine if degradation continues or levels off. If degradation of the material does level off, then the insulating material might still be suitable provided the expected loading under service conditions could be accommodated by the material.

A survey of additional candidate materials should be performed. Samples of these materials should be obtained and screening tests performed. These tests should include the same material properties tests performed in the D5 series. In this way, should the Marinite-P continue to degrade under long-term aging tests, an alternative would be available.

The basic design concept can provide an adequate insulated pipe support system meeting stringent requirements. With a closer fitting clamp/pipe shield and a better knowledge of the material's long-term properties, this clamp design could provide a satisfactory support system.

11.4.2 EM Pumps

As stated earlier, the original plan was to utilize the FFTF pump design for System 81. This decision was based on the fact that tooling was on hand for the L-200 pumping section and pumps had been fabricated to a design generated for FFTF by Mine Safety Appliance.

However, testing results at ETEC of the FFTF L-100 (100 gpm) pump indicated two problem areas that created concern regarding the use of this pump design. These were (1) severe vibrational modes at high flows and low head conditions and (2) a high pump duct ΔT at low flows and high head.

AI-ESG began a study to determine if these problems could be predicted analytically. It was determined that it was possible to explain the induced vibration by both the acoustic impact of the jet stream header on the header wall and by the energy deposition into the downstream header by flow discharge. The two approaches provided consistent results. It was also determined that the high thermal gradient problem was probably due to the low cross-sectional area of the side bars, which caused high current densities and therefore high heating rates in the side bars. Since the side bars are thermally insulated from the coolant air flow, the heat generated must be conducted through the tubing in the duct and dumped to the passing sodium, thus causing the high ΔT s.

The L-100 pump did perform as designed. Since the problems described above were shown to be analytically predictable, AI-ESG recommended that with proper redesign of the headers and decreasing the heating rate and/or cooling, the duct side bars, the basic pump design concept, could provide a reliable pump for CRBRP.

It was also recommended that water flow tests be conducted of a new header/throat section design to demonstrate performance, namely, minimization of vortex generation in the header, low pressure drop, low flow-induced vibration, and bubble formation. Additionally, it was proposed that an engineering model of the new pump design be fabricated and tested in sodium to verify the design prior to the start of fabrication of the plant units. This test program was to be performed early enough so that the test results could be factored into the design if necessary in sufficient time to preclude delay in manufacturing of the plant units.

Project approval was given to proceed with a new design of the header and throat section. Figure 11-7 illustrates the design change for the header. Figure 11-8 illustrates the design change for the pump throat section. In the new design, the throat was of one-piece construction.

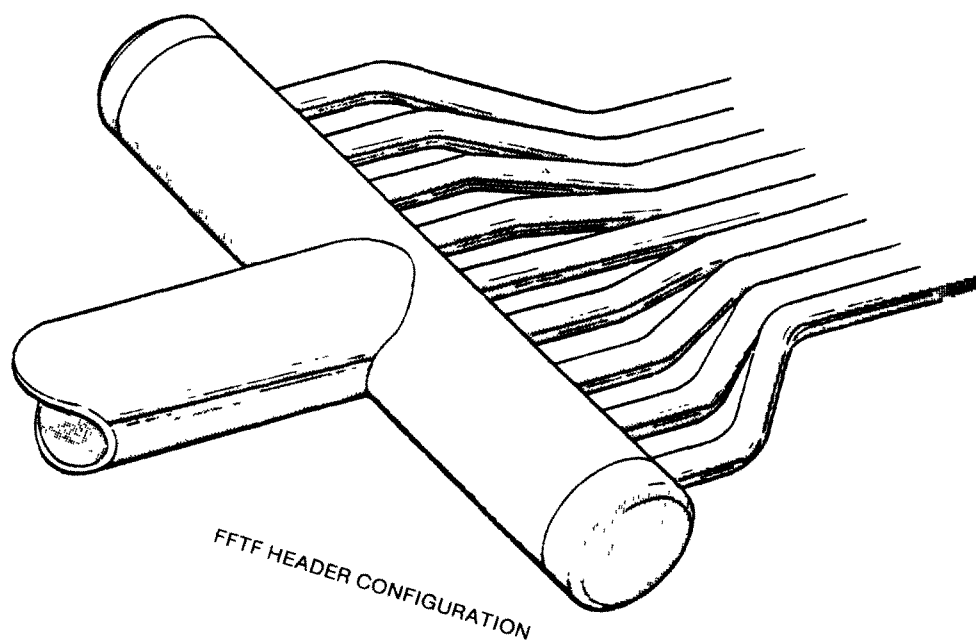
A clear plastic water test model was fabricated and tested to the flow conditions required to meet the CRBRP requirements. Test results substantiated the analytical predictions.

Fabrication of a 400-gpm prototype test pump was initiated. In order to gain experience in the fabrication of an ASME Section 3 Class 1 pump, all pressure boundary materials were purchased as Code material. All welding was performed to Code requirements. The stator assembly was fabricated in the same manner required for the plant units. Figure 11-9 provides a cutaway view of the completed pump assembly.

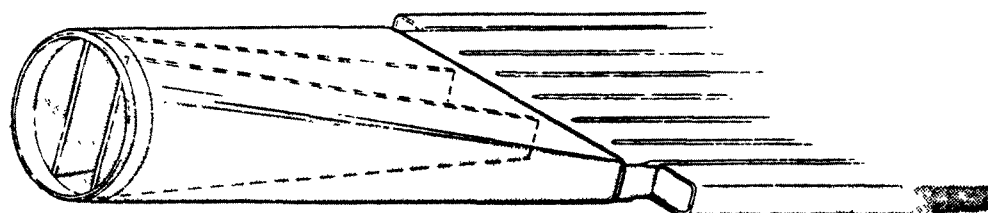
A test program was prepared and tests were conducted at ETEC to verify flow and electrical performance characteristics of the new pump electrical and mechanical design.

The functional pump was welded into a 4-in. sodium loop at ETEC where it was subjected to a series of tests that enveloped the CRBRP requirements. Excellent results were obtained at all planned sodium temperatures and flow rates with ample excess capacity being exhibited. The total running time for the pump during the testing exceeded 1000 h, including a 200-h continuous run at 300 gpm with 900°F sodium.

Because of the number of system variables (such as loop flow impedance, sodium temperature, and pump operating voltage), many curves were required to fully characterize the pump's performance. Flow-mapping curves at 276 V and 368 V and 800°F sodium are given in Figures 11-10 and 11-11 to illustrate typical pump performance. These data were obtained by setting the pump at the pre-selected voltage and adjusting the loop throttle valve for the desired flow

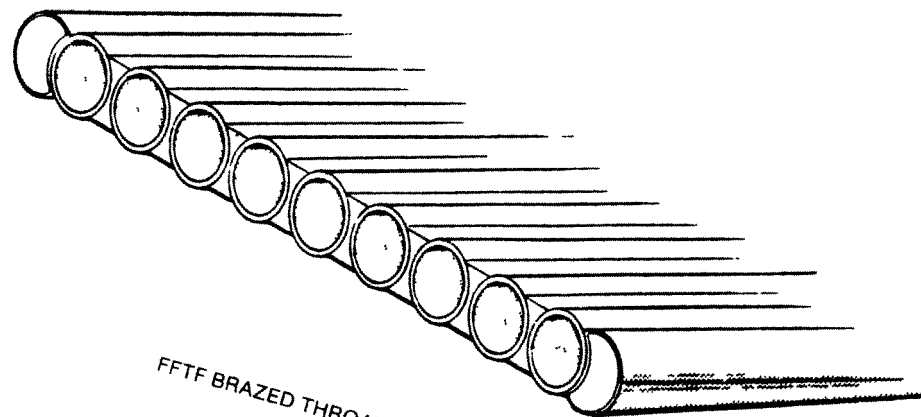


FFTF HEADER CONFIGURATION

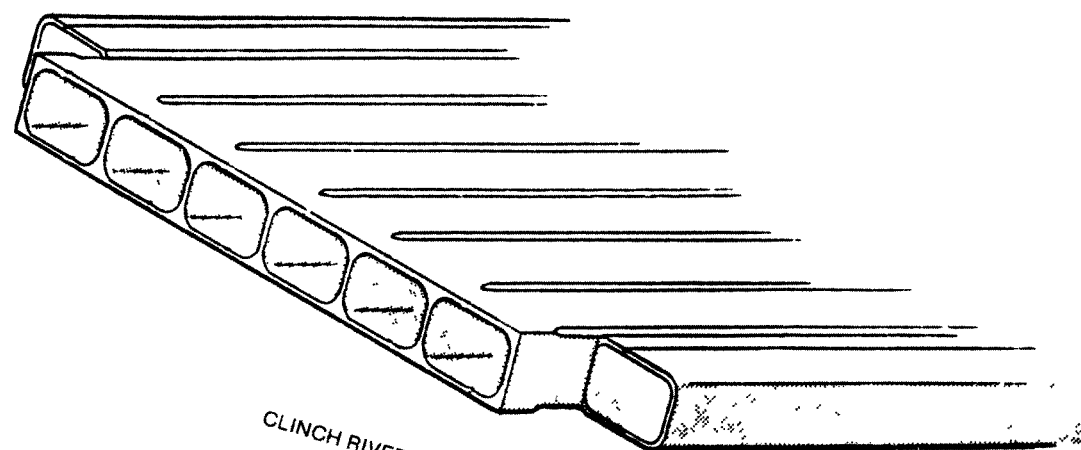


CLINCH RIVER HEADER CONFIGURATION

Figure 11-7. EM Pump Header Configurations



FFTF BRAZED THROAT CONFIGURATION



CLINCH RIVER THROAT CONFIGURATION

Figure 11-8. EM Pump Throat Configurations

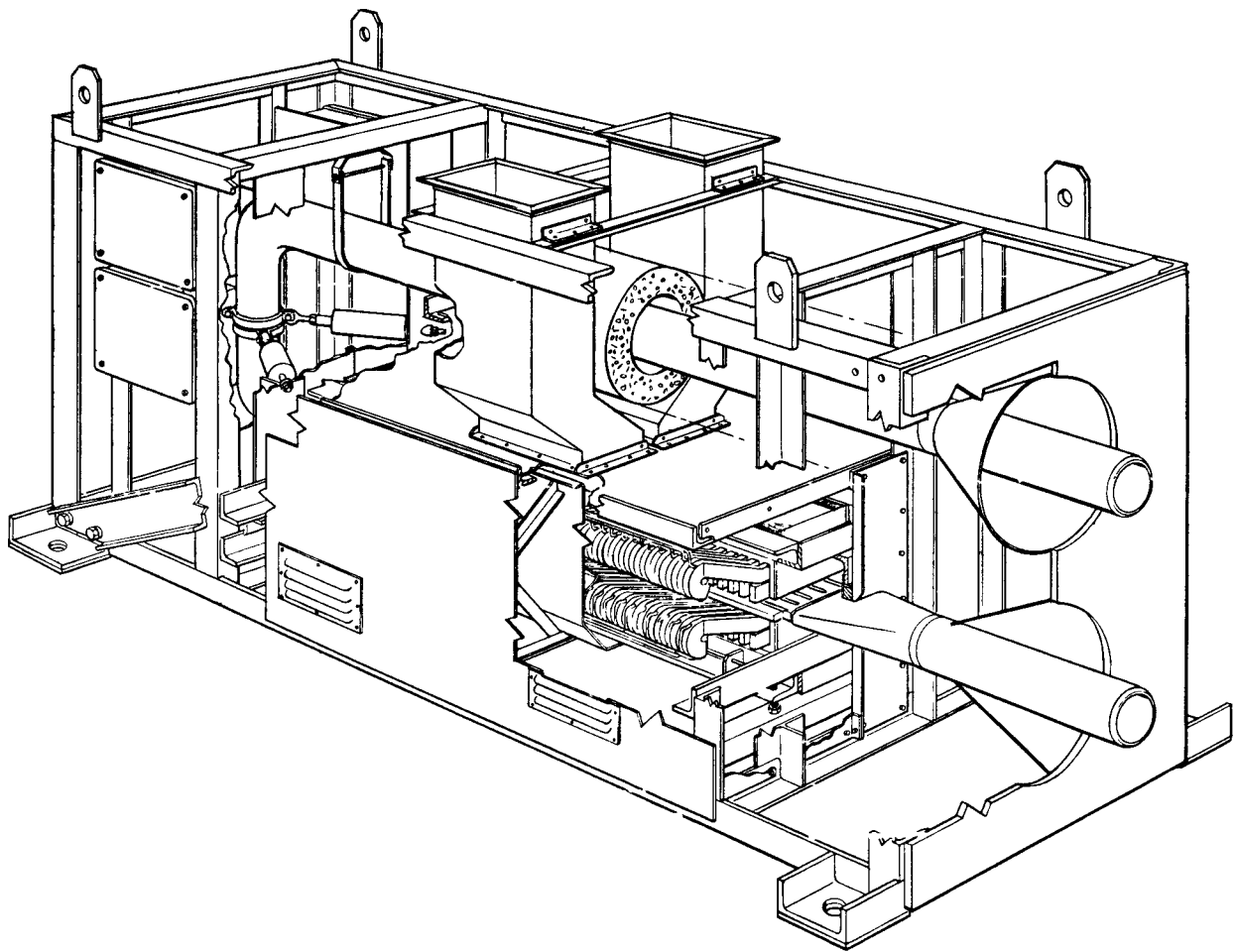
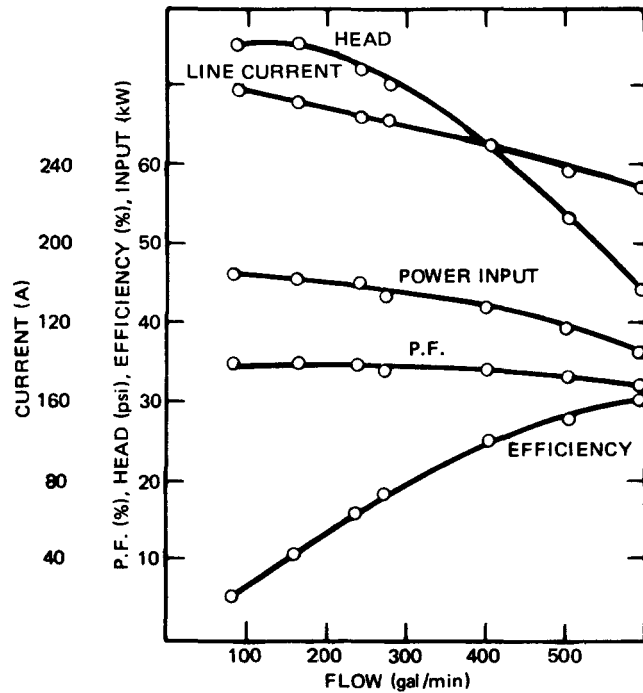


Figure 11-9. EM Pump Assembly



9212-52

Figure 11-10. Flow Mapping at 276 V and 800°F

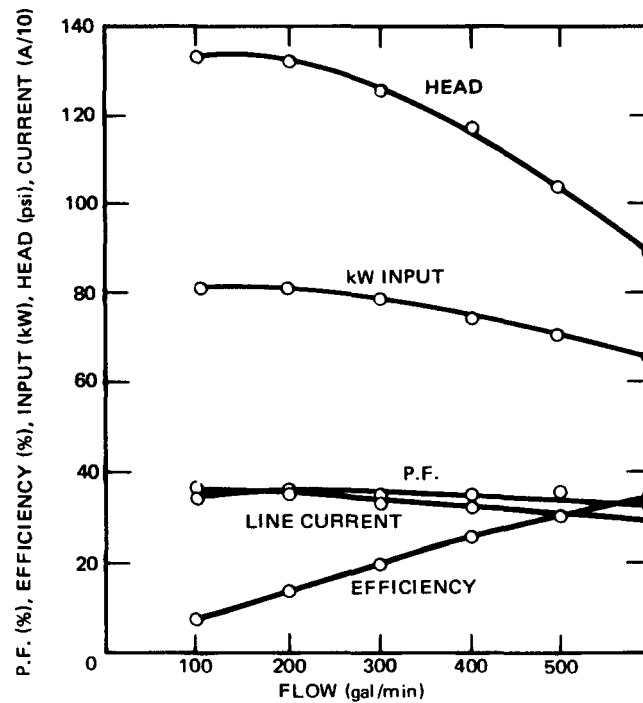


Figure 11-11. Flow Mapping at 368 V and 800°F

point. A performance-design-limits test was also performed to establish the pump capabilities at off-design operation. The test parameter limitations were established by the test loop flow rate limit of 800 gpm and the pump pressure limitation of 200 psi rather than by pump performance limitations. These data are shown in Figure 11-12.

Throughout this testing, no evidence of pump overheating or flow instability was observed. Cavitation testing was also performed by reducing pump inlet pressure while maintaining a given flow rate. The cavitation point of the pumping section, which is evidenced by an abrupt decrease in flow and some noise at the pump, was quite low. An inlet pressure of 7.5 to 12 psia was adequate to prevent cavitation at flows up to 500 gpm.

The testing results were most satisfactory and demonstrated the EM pump to be a simple, rugged device capable of meeting or exceeding the requirements for which it was designed. Fabrication of the functional test pump confirmed the efficacy of the basic design and demonstrated that the manufacturing of the stators and flow duct assembly could be achieved with normal manufacturing methods and processes.

11.4.3 Valves

The small liquid metal thermal transient valve design selected for CRBR plant valves were of the basic design used for FFTF. Common design features were the body shape/dimensions and the bellows systems for the primary stem seal. In fact, the forging dies used for the FFTF 1-, 2-, 3-, and 4-in. valves were the same as used for CRBRP. Figure 11-13 illustrates a typical CRBRP valve design. Figure 11-14 illustrates a typical CRBRP valve with operator. Due to the design and operating requirements established for CRBRP, some minor design changes were required. Some of these changes were dictated by design requirements, others in an attempt to reduce cost.

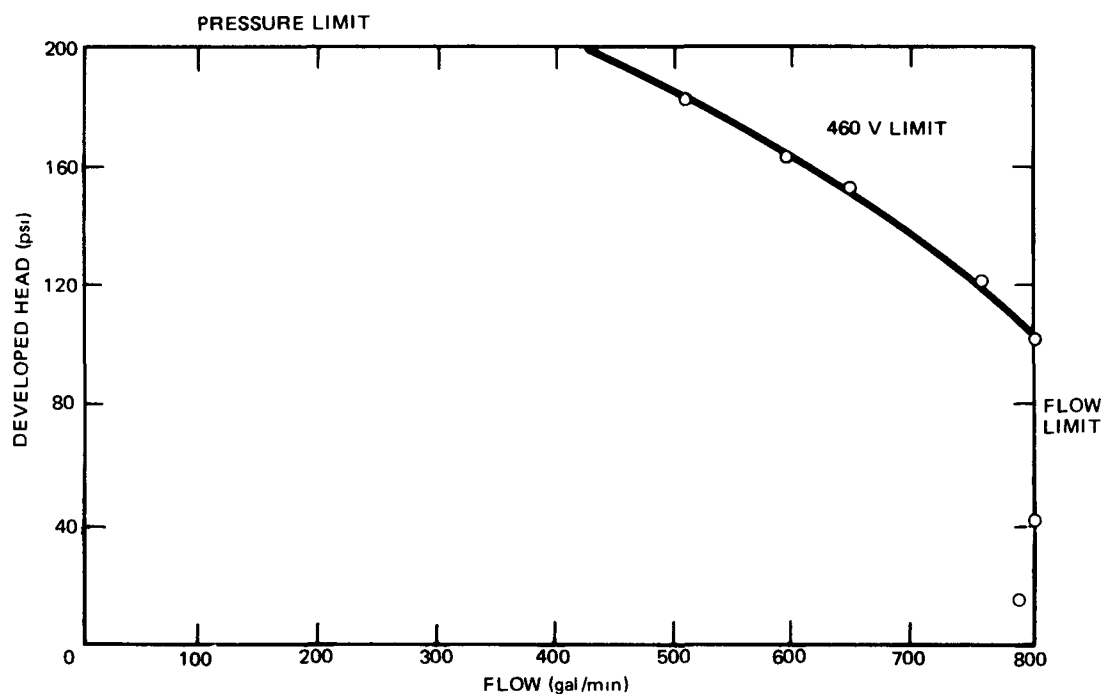


Figure 11-12. EM Pump Design-Limits Test at 600°F Showing Test Points

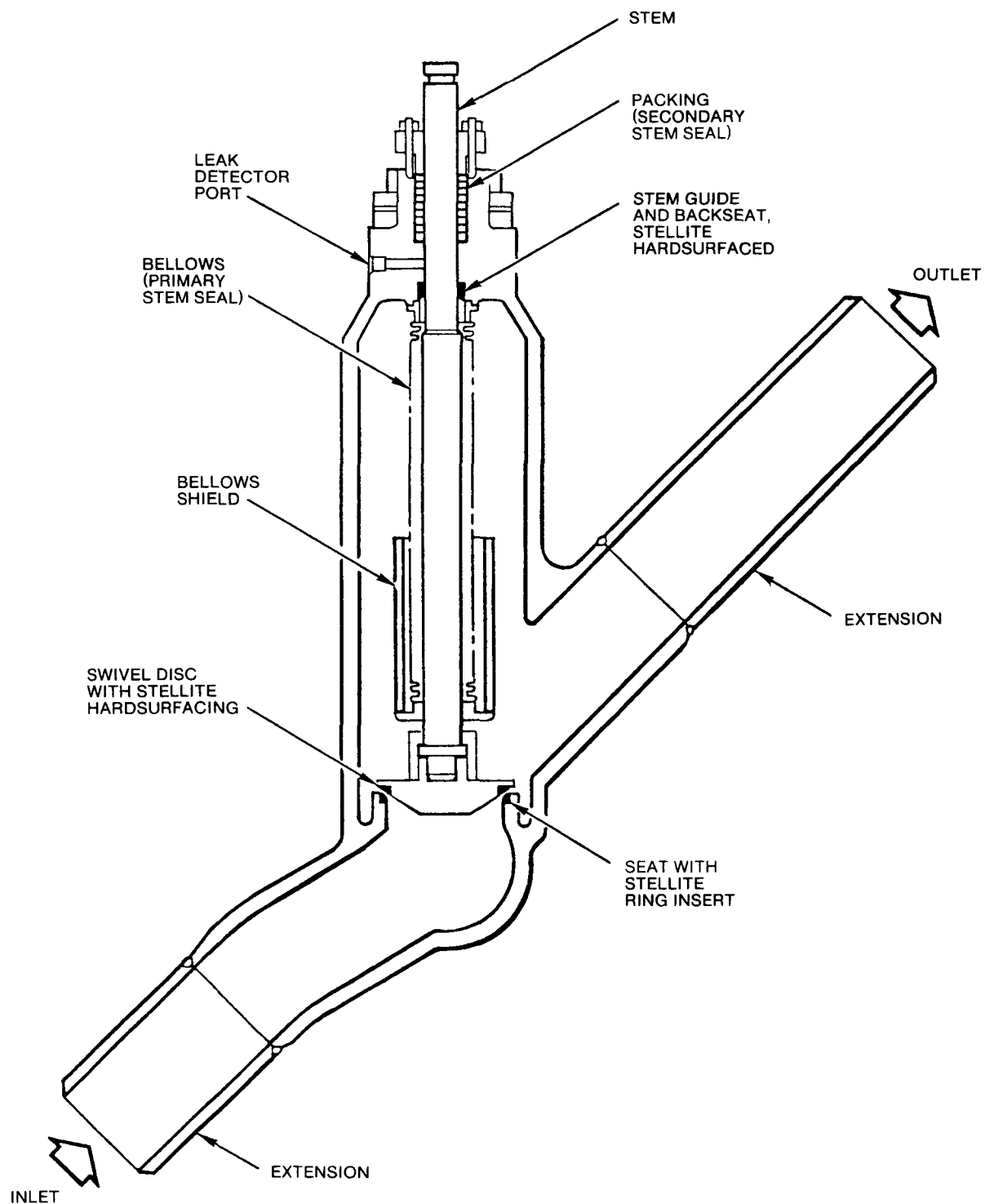


Figure 11-13. Typical Thermal Transient Valve

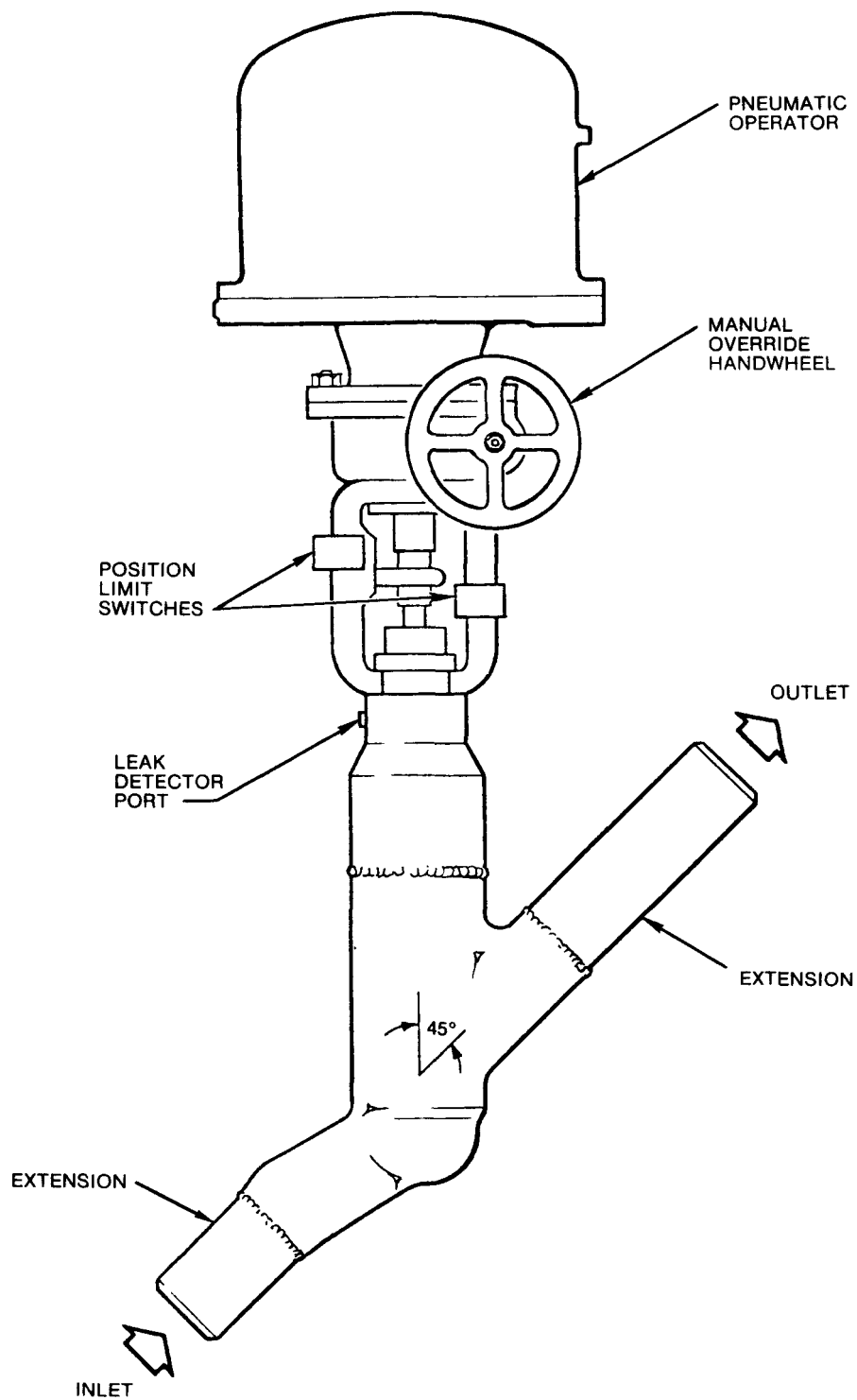


Figure 11-14. Typical Thermal Transient Valve Operator Assembly

To ensure that these design changes would not affect the valve performance, a test program was established to provide test data on prototype valves of the sizes needed for CRBR. The contract with the valve manufacturer stipulated that prior to the production of the plant valves, testing of the prototypes had to be completed.

A Development Requirements specification was prepared, approved, and implemented (DRS 81.07). Tests to be performed included: (1) functional tests, (2) valve seat leakage tests, (3) flow characteristics, (4) life cycling tests, (5) thermal cycling tests, (6) seismic tests, (7) pneumatic operator performance tests, and (8) valve operability during and/or following a seismic event. Included in the test program were the 1-, 2-, 3-, and 4-in. valves.

In general, the valve testing indicated that the valve performed as specified except for a stem problem detected during the fabrication and/or testing of the valves.

A 1-in. valve was found to have a bent stem which was discovered during the fabrication cycle. A 2-in. valve, after undergoing some thermal cycling tests successfully, was found to have a bent stem, as exhibited by seat leakage in excess of requirements. Another 2-in. valve was found to have a bent stem even though it had yet to be tested. Both 4-in. prototype valves were found to have bent stems--one admittedly after excessive loadings during a seismic test, the other after a successful sodium thermal testing program.

Considerable effort was expended to determine the cause of this common failure mode. The valve design was scrutinized in detail, and a probable cause fault analysis was created. Table 11-2 describes the elements that could cause the failures. An in-depth study of the manufacturing processes and the testing history log books was made. Additionally, material properties of the valve components parts were again reviewed in detail. No specific normal operational mode postulated could be expected to cause the failure. However, if

TABLE 11-2
POTENTIAL CAUSES OF STEM DISTORTION AND VALVE LEAKAGE
(Sheet 1 of 2)

Potential Cause of Failure	Mode of Failure	Test or Analysis to Demonstrate Potential for Failure	Potential Solutions
Combination of excessive load and misalignment	Progressive ratchetting resulting from exceeding proportional limit	Plastic analysis including misalignment	. Higher strength material . Reduced loads . Overload protection device . Increased stem diameter . Better control on misalignment
Nonuniform, eccentric reaction of loads at seat	Progressive ratchetting	Test to determine load distribution at maximum offset	Flat seat
Insufficient swivel in the disc causing eccentric loads on disc	Progressive ratchetting	Compare minimum clearances with potential misalignment	Increased swivel angle tolerances (axial and/or radial gaps)
Improper setting of spring preload; excessive stem load	Progressive ratchetting	Examination of 2-in. valve operator	Accurate calibration of spring load in closed position
Property (yield) degradation due to braze cycle	Plastic loading which can lead to ratchetting	Run pull test specimens. Establish limits on property degradation	Reduce load or increase stem diameter. Hardness indications are that it is lower strength

TABLE 11-2
POTENTIAL CAUSES OF STEM DISTORTION AND VALVE LEAKAGE
(Sheet 2 of 2)

Potential Cause of Failure	Mode of Failure	Test or Analysis to Demonstrate Potential for Failure	Potential Solutions
Dynamic overload due to "slamming shut"	Plastic buckling	Check possibility of "slamming shut"	Design of twice nominal load
Overload by manual override	Plastic bending (buckling) of the stem	Service history: torque/axial load relations	. Do not use override to close on this valve . Calibration . Variability
Excessive loads induced by handling or during installation	Plastic bending of the stem	Check handling and service history	Installation requirements revision
Manual override in a closed position - ΔT	Plastic bending of the stem	Service history	Manual override should not be used to close
Conical seating surface does not seat uniformly	Bending loads on stem - axial and bending causes plastic distortion	Check stability and self-centering assumptions	Stabilizing springs for the disk (Belleville washer). Redesign of seat/disk interface
Inappropriate design limits for compressive loads	Plastic ratchetting or buckling		Design to fraction of proportional limit. Include bending compressive stresses.

the manual override was used excessively, it could create a stem-bending condition. Excessive loads induced by handling during manufacturing or testing could also create a condition that would cause the stem to bend. No determination could be made indicating that the above conditions existed.

A major redesign effort was proposed to develop a design of the stem elements that would preclude the possibility of any further stem bending. The changes included (1) a redesign of the stem-to-disc attachment, which eliminated a braze operation that could reduce material properties, (2) an increase in stem diameter, (3) using higher strength A286 alloy steel instead of 316 stainless steel as the stem material, (4) incorporating a lockout device for the manual override, (5) incorporating a clutch assembly to the manual override to limit the amount of torque that would be applied, (6) a redesign of the stem guide bushing assembly to ensure proper alignment of the stem/disc to the seat, (7) the incorporation of a spring pack device which could allow for thermal growth of the stem when attached to the valve operator, (8) redesign of the operator to ensure absolute alignment of the operator shaft in relationship to the valve stem, (9) a redesign of the stem/disc attachment to ensure that the disc would articulate so that it was self-centering to the seat, and (10) a change in the seat angle from 120° to 90° to reduce the seating loads required for no seat leakage. Analysis indicated that the stem loads required to seal the valve could be reduced significantly.

Several of the design changes described above were experimentally evaluated before any modifications were incorporated into the valve design. Tests were conducted on a valve body mockup which simulated the stem/disc, seat, and guide bushing arrangements. Objectives of the test program were to:

- Evaluate initial misalignment between valve articulating disc and seat with the disc not touching the seat. This test was to establish the effects on alignment of various guide bushing-to-stem clearances and stem-to-disc clearances.

- Demonstrate the ability of the articulating disc to align itself to the valve seat with a seating load on the valve stem. This test would evaluate the effects on final disc/seat alignment using various guide bushing-to-stem clearances, stem-to-disc clearances, and various angles on the valve disc.
- Determine the effect of stem-to-packing clearance on initial and final alignment while using various guide bushing-to-stem clearances and stem-to-disc clearances.
- Perform valve leak tests using 325 psi water pressure below the disc with various stem loads to optimize valve stem loads required to meet the equipment specification leak rate of 10 cc/hr·in. in diameter of seat.
- Determine the effect of various disc angles (i.e., 120°, 90°, and 80°) on final disc/seat alignment and stem load versus leak rate.

A detailed description of the test equipment and the results are described in Valve Stem and Disc Mockup (VSDM) Test Report N099TR810005 dated June 3, 1983.

At the conclusion of the test program, the results indicated that the proposed changes would improve the valve performance and eliminate or substantially reduce the potential of a bent stem. The following is a description of the original design and the changes effected.

11.4.3.1 Stem-to-Disc Attachment

The original design of the stem-to-disc arrangement (Figure 11-15) required that the disc be brazed to the retaining collar. This design feature necessitated that the stem would be subjected to a high-temperature (greater than 1800°F) braze cycle. The thermal treatment reduced mechanical properties of the 316 stainless steel stem material. Additionally, it was found during the disassembly of two valves that there was no stem articulation. The clearance

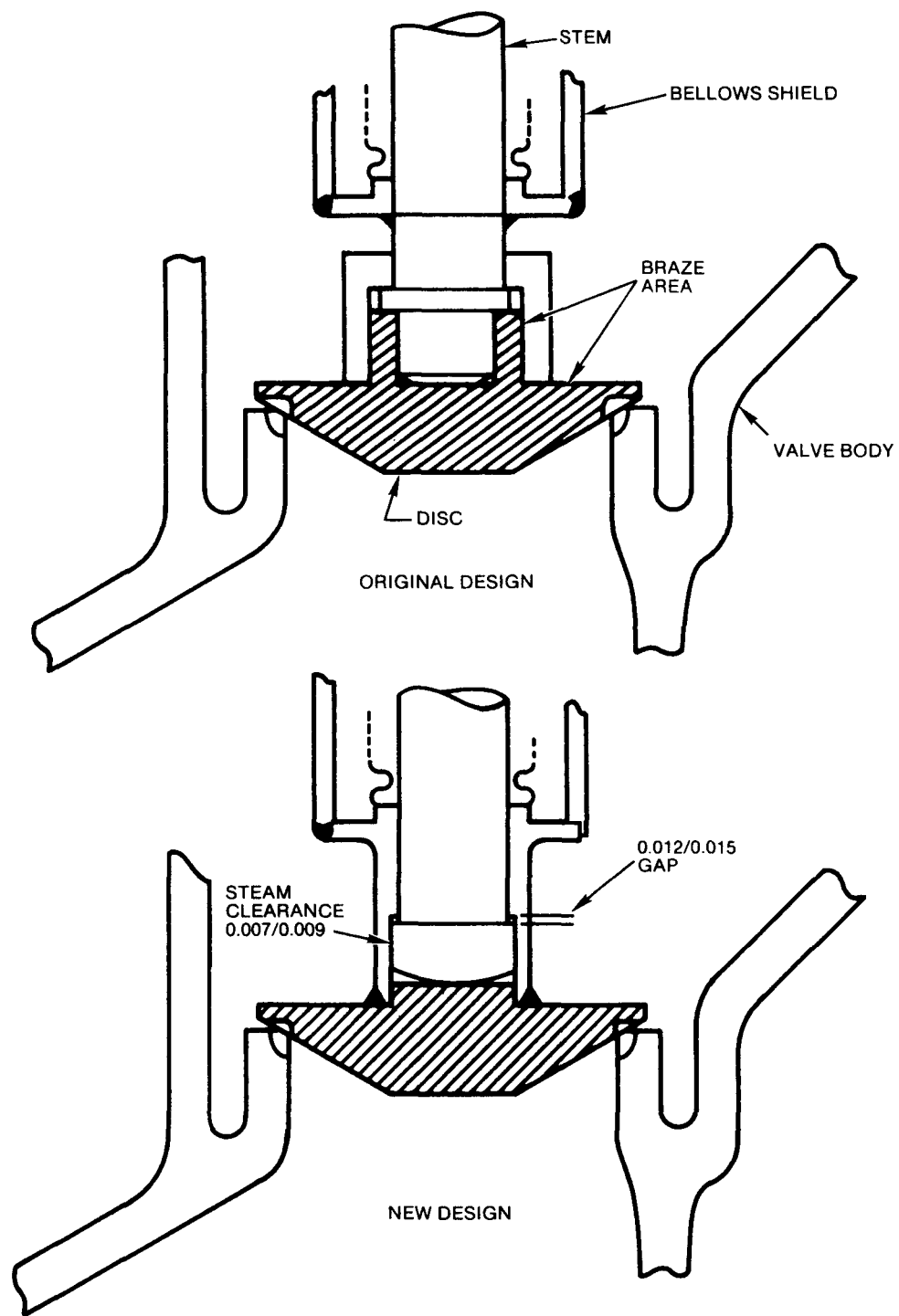


Figure 11-15. Valve Disc-to-Stem Design

between the stem and the disc socket was less than 0.001 in. Vertical clearance between the stem and the disc socket was also found to be minimal.

The new design provided for exact clearances to ensure articulation of the disc assembly and the elimination of the braze joint. As the stem material was A286 alloy steel, the brazing cycle would degrade the heat-treated properties of this material. Further, the bellows shield was redesigned to eliminate the weld to the stem. This was done to eliminate a dissimilar material weld as the shield was 316 SS and the stem A286 alloy steel. The new design also provided exact alignment between the bellows assembly and the stem.

11.4.3.2 Stem Design

The material for the stem was changed from 316 stainless steel to A286 alloy steel. This change provided for higher material strength. Additionally, the stem diameter was increased to provide additional strength. This design change was made without disturbing the bellows assembly design since the bellows had been thoroughly analyzed and tested for the FFTF valves.

To protect the stem further, a spring pack was incorporated between the stem and the operator. This device provides for thermal growth of the stem during operation if the valve is in a closed position during operation. The device was similar to that used by FFTF on their plant valves. It also provided for small misalignments between the valve stem and the operator, should any exist.

11.4.3.3 Bonnet Assembly Redesign

Several design details were modified on the bonnet assembly (Figure 11-16).

(1) The guide bushing was relocated and extended to provide better guidance of the stem. This bushing was faced with Stellite 6 as the inner diameter to provide a hard rubbing surface. (2) The Stellite 6 back seat, which had to be an overlay in the original design, was moved to the bellows adapter. This change permitted the overlay to be performed as a detail part. (3) The bonnet was cut back to within an inch of the bonnet reduction zone to permit better

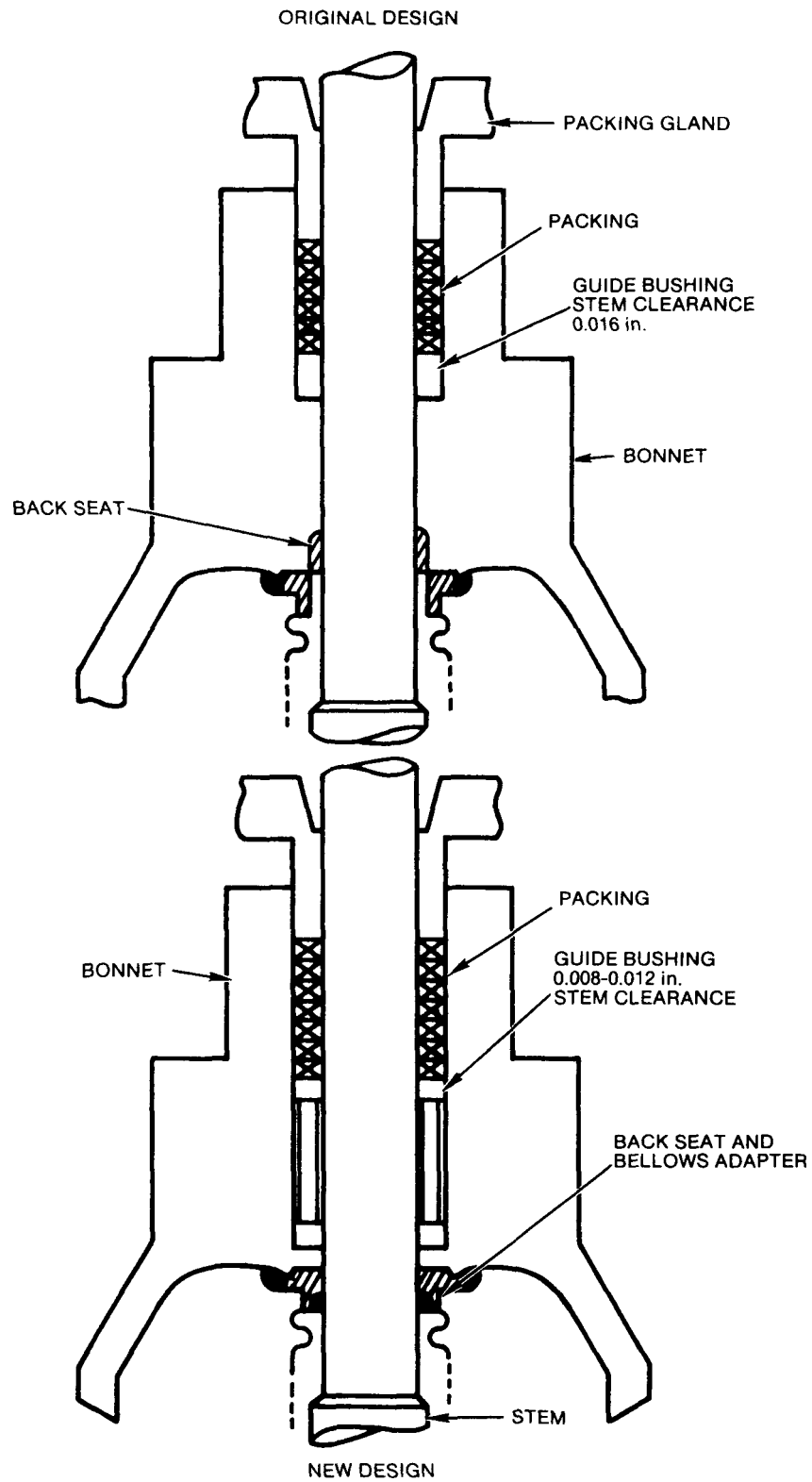


Figure 11-16. Bonnet Modifications

accessibility for welding of the upper bellows adapter to the bonnet. Originally, this weld was more than 10 in. away and was difficult to perform and inspect.

11.4.3.4 Disc Design

As a result of the tests, it was determined that the stem loads could be reduced as much as 39% if the disc angle was changed from 120° to 90°. This change was incorporated into the design. Additionally, the Stellite 6 seating face, which had been deposited by welding, was changed to a plasma spray application. This type of application eliminates the potential for porosity generally present by a welded overlay.

11.4.3.5 Operator System

To ensure that the manual override could not impose excessive loads on the valve stem, a slip clutch was incorporated into the manual override design. Additionally, the manual override system could be locked out to ensure against inadvertent operation. The design of the operator interface with the valve was modified to ensure concentric placement of the operator operating shaft and the valve stem so that eccentric loading of the stem was impossible. The reduced seating loads required also permitted the use of a smaller operator. The operator design was modified to permit the use of a smaller, lighter unit.

All of the above design changes were made to the valve design. A 3-in. prototype valve was modified to incorporate these changes. A commercial valve operator was modified to ensure concentricity of the piston movement with the valve stem. This valve was then subjected to ambient water seat leakage tests at the reduced stem loads. At ETEC, it was subjected to seat leakage tests with air. The results of these tests were positive. The valve was then subjected to a repeat of the sodium testing undergone by other valves. The seat leakage rates were well within the specification requirements. Operation of the valve was found to be as expected. With the lower stem loads and design

features, which diminished and/or eliminated potential stem failure, it is expected that this valve design would provide CRBRP with an acceptable valve.

11.5 Final System Status

11.5.1 Status at Project Termination

11.5.1.1 General System Status

At the time of project termination, the basic design was complete except for the final stress analysis of the piping. The outstanding changes and activities discussed in the following sections would not have had a major impact on the general physical design of System 81. Two final system design reviews were held. The first, documented by Ref. 16, was a formal in-house review meeting the requirements of Reactor Development and Technology (RDT) Standard F2-2. Most of the comments from this meeting involved corrections to, and updating of, the System Design Description (SDD) and other documentation to correctly reflect final system and component design. These changes had not been incorporated at project termination. The second review, on May 16, was a lead reactor manufacturer (LRM)/Project Office final working system design review. The review comments and their responses are given in the Attachment to Ref. 17. Most of these comments involved changes and additions to Sections 3 through 6 of SDD 81. The comments to SDD Sections 3, 4, and 5 were incorporated into the latest published draft listed in the following section.

11.5.1.2 Documentation Status

At project termination, the published status of the System Design Description, SDD 81, was as follows:

Sections 1, 2, and Appendices	Rev. 33
Section 3	Draft 6
Section 4	Draft 7

Section 5	Draft 6
Section 6	Draft 6
Maintenance Outline Procedures (Appendix H)	Draft 4

The principal system and component documentation covering System 81 is listed in documentation diagrams N099007539 (Figure 11-17) and N099005513 (Figure 11-18). The drawings have been annotated for this report to indicate those planned documents that had not been released at least in draft form. As a group, only the component operation and maintenance (O&M) manuals and installation requirements documents had not been prepared.

The status of the System 81 piping and instrument diagrams (P&IDs) diagrams (included as part of the SDD) is given below:

Na/NaK Receiving	Rev. 11
Primary Na Storage and Processing	Rev. 30
EVS Processing	Rev. 27
Primary Cold Trap NaK Cooling	Rev. 17
Intermediate Na Processing	Rev. 21

11.5.1.3 Supporting Documentation and Reference Sources

The general design bases for the System 81 systems and components are contained in the system design description and final system design reports, Ref. 18 through 23. These documents, particularly the design reports, provide all requirements and their verification, the overall rationale, and experience on which the designs are based; they also reference the supporting system and component documentation generated during the design. The design reports provide the principal source of detailed references for System 81.

11.5.1.4 Component Status

The component status at project termination is summarized in Table 11-3.

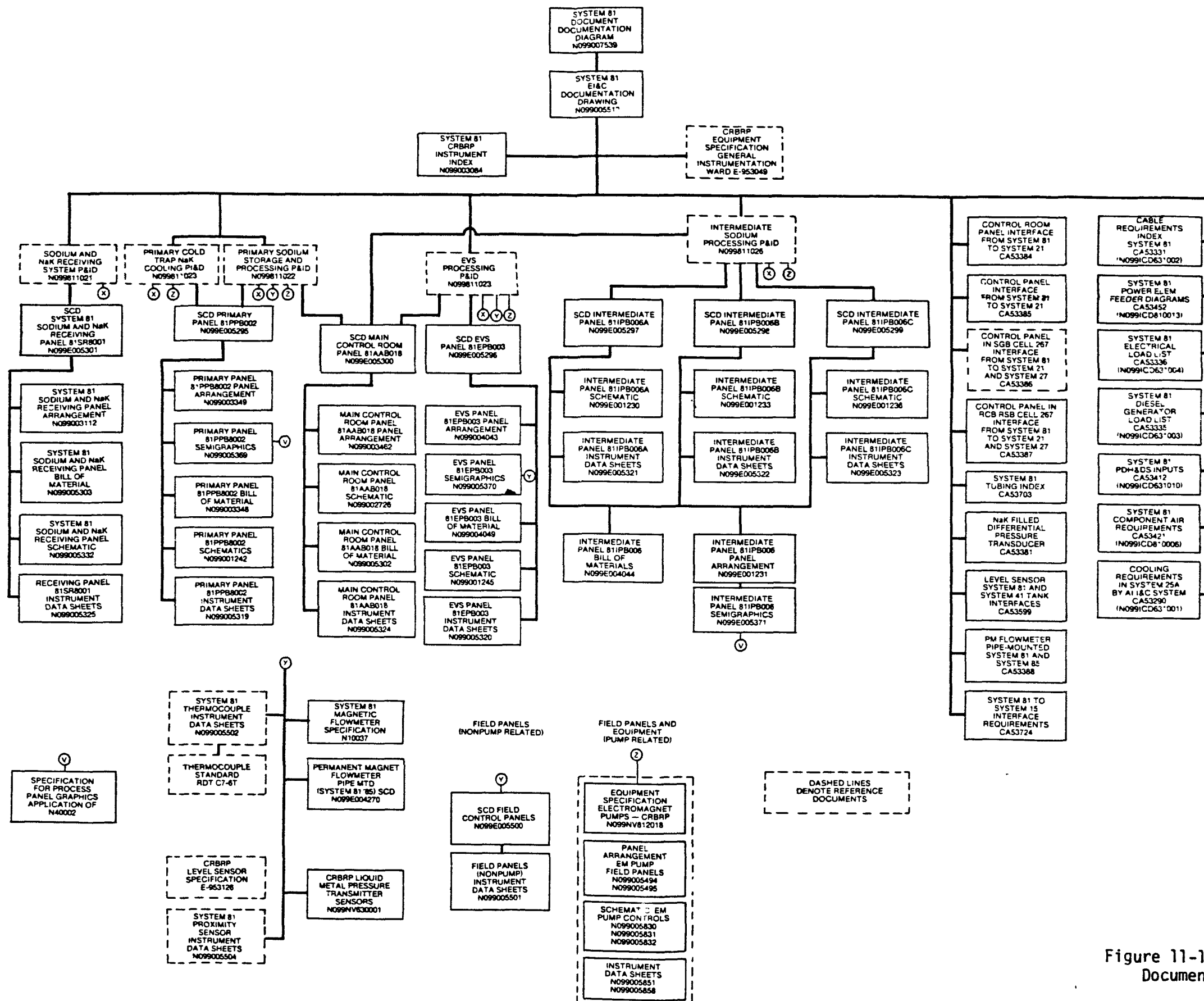


Figure 11-18. System 81 EI&C Documentation Diagram

TABLE 11-3
COMPONENT STATUS

<u>Delivered Components (in storage)</u>	
● Overflow vessel ● In-containment/ex-containment storage vessels ● EM pump prototype ● T.T. valve^a prototypes (1 in., 2 in., 3 in., 4 in.) ● NDHX^b ● 1 in. and smaller valves	
<u>Components on Order</u>	
● Check valve^c ● ABHX^d ● EM pump throats^c ● Penetrations^d ● Small vessels^c ● Na pressure sensors^c ● Coolers and heat exchangers^c	
<u>Components Not Yet Ordered</u>	
● Na cold traps	Design complete
● Na T.T. valves	Design in-work (AI)
● EM pump stators	Design complete
● EM pump Phase II (make)	Design complete
● EM pump panels and cabinets	Design deferred
● Flowmeters	Vendor design (not done)
● Panels	Common procurement by W - Requirements have been provided by AI
● Level sensors	
● Thermocouples	
● Proximity switches	
● Na/NaK drum stations	Design being initiated

^aThermal transient (Y-pattern) valves

^bCurrently installed at ETEC following completion of component sodium test.

^cDesign complete

^dVendor final design in work

11.5.2 Outstanding Approved Changes

With few minor exceptions, outstanding changes that had been approved at project termination had been incorporated into the principal documentation (SDD, E-Specs, Interface Control Documents and Drawings). No significant approved changes remained except incorporation into the piping design/stress analysis of changes that affected the piping, such as the valve changes discussed in Sections 11.3.3.1 and 11.3.3.2. A list of approved ECPs that had not been completely incorporated into the design is given in Table 11-4.

11.5.3 Outstanding Activities to Complete

11.5.3.1 General Activities

In addition to completion of component procurement, a number of design activities remain to be completed. The more significant of these are discussed below.

- Documentation Update - Revise and update documentation to incorporate design review comments and reflect current design. Principal documentation involved is the SDD, the design reports, maintenance, construction and pre-op test procedures, and instrumentation and control documents such as the instrument and cable indices, and panel layouts. Baseline all SDD sections.
- Prepare component O&M manuals and installation requirements documents.
- Establish and implement overpressure protection requirements and prepare overpressure protection report (refer to Section 11.5.3.2).
- Prepare qualification test plans for 1E electrical equipment, and update or clarify qualification requirements in applicable E-Specs.
- Completion of the final piping design and stress analysis.

TABLE 11-4
APPROVED ENGINEERING CHANGE PROPOSALS
NOT COMPLETELY INCORPORATED AT TERMINATION

A81383 - System 81 to 82 Interface Requirements Revision
A81395 - Piping Penetration Specification and ICD Update
A81265 - Primary Cold Trap Redesign and Cleanup Facilities
A81397 - Sodium Valve Specification Revision
A81437 - Liquid Metal Valve Seismic Support Relocation
L10202 - Revise Decay Heat Removal Requirements
L10165 - Clarification of Severe and Nonsevere Sodium Aerosol Definitions
in WARD-D-0165
L10193 - IEEE Standards in OPDD-10
B25056 - Addition of Tornado Dampers
A82354 - Hold Removal from Equipment Specification N099NV812009
A81339 - Flowmeter Specification Update
W1803 - PDH&DS/PAS Interface Philosophy
W1857 - Control Room Communication and Annunciator Interface Requirements
A81373 - EVS Cold Trap Gas Ducting
A81416 - Update P&IDs and EI&C Documents
A81392 - Valve Equipment Numbers and Preheat ICD Update
A81393 - Intermediate Na Processing System Isolation
A81276 - Update and Baseline Liquid Metal Spills in Inerted Cells
A81346 - ABHX Equipment Specification Changes

- Formalize and implement in-work changes:
 - Revise valve/operator weights consistent with final valve design. The final (increased) operator weight for 1-in. and 2-in. valves would require separate deadweight and seismic support on the operator. The valves were previously supported by the piping. Principal effects of this change are to the pipe support design and to auxiliary steel.
 - Revise piping insulation weights and add insulation support clamps as determined by the final insulation design. The final insulation weights were generally equal to, or less than, previously used weights. This change would not significantly affect the piping design, but the additional insulation clamps would affect the System 68 heater arrangement drawings.
 - Incorporate into component and piping E-Specs the detailed cell sodium spill environments. The impact of this change is anticipated to be minor, principally involving changes to or relocation of pipe supports.
 - Update pipe penetration, anchor and guide details to reflect final piping design.
 - Delete five valves for cost reduction and system simplification. Operating procedures would be revised to permit this modification. The valves to be removed are noted on Figure 11-1.

11.5.3.2 Open Issues

Table 11-5 lists the unapproved ECPs outstanding at termination.

There were three significant unresolved issues at project termination: overpressure protection, DHRS flow path, and spills in air-filled cells.

TABLE 11-5

UNAPPROVED ENGINEERING CHANGE PROPOSALS OUTSTANDING AT TERMINATION

A81415 - Revision to Valve Weight/Center of Gravity
A81427 - System 81 IHTS Isolation Valves - 1E Power
A81438 - Revision to Valve Equipment Specification
A26036 - Na/NaK Leaks in Air-Filled Cells

OVERPRESSURE PROTECTION

Most of the System 81 components and much of the piping is protected against overpressure by relief valves provided by System 82 in the cover gas piping. The drain piping is protected by providing proximity gages to ensure that the lines are at least partially void during normal plant operation. The problem areas are those electrically preheated components and pipe sections that can be isolated by valves and are normally full. The initial plan was that administrative procedures were adequate to ensure against overpressure. This is the FFTF approach. During final design, informal discussions with the ASME Code inspector indicated that this approach would not satisfy Code requirements. Resolution of this problem involves, for the most part, a combination of electrical interlocks among various valves and heater circuits to protect against overpressure of potentially isolatable pipe segments. Another solution, applicable in some locations, is to provide physical stops on valves to prevent complete closure; this approach had been planned, for example, for the 4-in. valves on the EVST sodium pump discharge valves, which are used only to provide a pressure drop for cold trap operation. This problem had not been resolved at project termination. The detailed solution was not seen as having a significant effect on System 81; the major effect would be on the instrumentation and control of the electrical heater circuitry.

DHRS FLOW PATH

As discussed in Section 11.3.3.1, the requirement for DHRS operation using the PHTS drain piping as a flow path remained unresolved. The principal concern was the potential requirement for remote operators on many drain valves, not only because of the added cost and complexity, but also because of the increased potential for inadvertent PHTS drain.

SPIILLS IN AIR-FILLED CELLS (ECP A26036)

At the time of project termination, a change proposal was in work to establish detailed design changes necessary to protect against liquid metal spills in

air-filled cells. The major locations of concern are the three steam generator bays in the SGB. The approach adopted for the large IHTS piping is to prevent a spray fire with a combination of piping insulation (providing confinement) and short, open sections of pipe located between pipe clamps to drain the pipe/insulation annulus in such a way as to prevent a spray. This approach was not considered practical for the System 81/85 piping because of the extensive amount of piping involved, and also because the design and analysis of the smaller piping would be significantly affected by the sections of drain piping required. The only approach considered practical for this piping is to stop a leak by immediate closure of isolation valves actuated by smoke detector signals. Because of the single-active-failure criteria, double isolation valves were being considered (the existing design has single valves only). Also, this is the one location in the plant where the isolation valves are located in the same cell as the fire. A protective cover over the operator and exposed valve shaft was being considered for protection against direct sodium spray impingement and gross aerosol deposition. It was hoped that this approach would avoid the need for valve operability testing in aerosol/spray environments. The acceptability of this approach and the detailed design changes required had not been established.

11.5.4 Recommendations for Future Designs

COLD TRAP COOLING

For future designs, it is strongly recommended that all sodium cold traps be air or gas cooled. Liquid-cooled traps are not recommended because of the increased cost and complexity of this type of design and the associated multiple systems required for heat rejection. The only disadvantage of the air/gas cooled design is its somewhat larger crystallizer size due to the increased heat transfer surface required. As long as the size is provided for in the initial plant arrangement (which was not the case in CRBRP due to the initial plan to use the FFTF design), all factors favor the air-cooled design.

COLD TRAP REGENERATION

From a cost, arrangement, and handling standpoint, the CRBRP IHTS cold traps probably represent a reasonable practical limit from a flow and capacity standpoint. Considering the fact that hydrogen in-flux is directly related to steam generator area and therefore plant size, the large cold trap replacement costs envisioned for CRBRP can only increase with larger plants. Consequently, future designs should include cold trap regeneration as a part of the purification system design.

PRIMARY SODIUM STORAGE

The need for permanent storage capacity for primary sodium should be carefully evaluated, particularly for larger plants, with the objective of minimizing plant capital costs realistically. In retrospect, the CRBRP design to permit reactor vessel drainage was probably overly conservative; designing for drainage of a single PHTS loop would appear acceptable and would provide substantial savings in component and building costs.

GENERAL DESIGN SIMPLIFICATION

Considering the difficulties (and costs) experienced in the design and stress analysis of the piping, particularly in the high-temperature primary sodium overflow and makeup circuit, two general areas stand out in which simplification should be emphasized in future designs. These are valves and ASME Code classes. The valves are bulky, complex, and costly, particularly those with operators and associated support structures, although manual valves with reach rods (those in inerted cells) are also troublesome. The valves themselves and their installation in the piping were the most costly and troublesome of all components from a design and stress analysis standpoint. In addition, the valves can become a major problem area in satisfying Code overpressure protection requirements. The issue of ASME Code classes is somewhat related to valves, at least as related to isolation of one class from another. The main point of emphasis is that selection of Code class should be done to take

advantage of the inherent safety features of sodium systems relative to light water plant design. The Class 1 overflow/makeup circuit is a case in point; the selection of Class 2 is more appropriate for a sodium system, notwithstanding the "primary coolant boundary" function. In the CRBRP design, at least six valves are included only to provide double valve isolation between the Class 1 circuit and components and piping of lower class. Although for CRBRP, Code class disagreements were unproductive, the emphasis on simplification of criteria, and design should still be pursued in the future to achieve lower costs without compromising plant safety or reliability.

11.6 References

1. WARD-D-0244, Rev. 3, "CRBRP Preservice and Inservice Inspection Requirements for Sodium and Cover Gas Systems," dated January 7, 1983
2. WARD-D-0177, Rev. 14, "CRBRP ALARA Criteria and Guidelines," dated July 14, 1983
3. OPDD-10, Rev. 146, "Overall Plant Design Description," dated August 1, 1983
4. NRC Letter, T. Speis to P. Van Nort, "Docket No. 50-537," dated March 5, 1976
5. N099RAR811001, "Reliability Analysis Report, EVST Cooling System," dated May 11, 1976
6. Letter PS:78-299, R. Copeland to J. Nolan and M. Hoyle, "NRC Staff Review of Summary of CRBRP," dated November 16, 1976
7. N099TI810119, "System 81 Liquid Metal Spills," dated November 15, 1983
8. N099TI810038, "CRBRP Cold Trap Study - Impact of Steam Generator Hydrogen Flux on Sodium Purification Systems," dated February 28, 1978
9. N099TI810046, "Revised Evaluation of Hydrogen and Tritium Transport in CRBRP," dated February 20, 1978
10. Engineering Change Proposal A81262, "Cold Trap Regeneration," issued December 12, 1981
11. NRC Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," dated October 1973

12. Letter LR830023, "CRBRP; Transmittal of Meeting Minutes from December 7, 1982, Piping Design Methods Meeting," dated January 13, 1983
13. ANL-83-83, "A Design Study of High Performance Cold Traps for Removal of Hydrogen from Liquid Sodium," November 1983
14. Engineering Change Proposal A81265, "Primary Cold Trap Redesign and Cleanup Facilities," issued November 13, 1981
15. ESG-DOE-13427, "CRBRP Small Pipe Clamp Test Report," June 30, 1984
16. N099DRR810018, "System 81 Final System Design Review (Report)," dated May 25, 1983
17. Letter 83ESG-5801, "System 81 Responses to FWSDR Comments," dated September 30, 1983
18. System 81 System Design Description, SDD 81, REv. 33, dated September 26, 1983
19. N099TI810133, "System 81 Primary Sodium Storage and Processing System and DHRS - Final Design Report," dated March 31, 1983
20. N099TI81034, "System EVS Processing System Final Design Report," dated March 22, 1983
21. N099TI810136, "System 81 Primary Cold Trap NaK Cooling System Final Design Report," dated October 21, 1983
22. N099TI810135, "System 81 Intermediate Sodium Processing System Final Design Report," dated March 23, 1983
23. N099ER810003, "System 81 Final Design Report, Electrical Instrumentation and Control," dated March 15, 1983

12.0 INERT GAS RECEIVING AND PROCESSING SYSTEM - SYSTEM 82

12.1 Functions and Description

12.1.1 Functions

The functions of the inert gas receiving and processing (IGRP) system, also designated as System 82 in the Clinch River Breeder Reactor Plant (CRBRP), initially were as follows:

- Supply and distribute argon and nitrogen to other plant systems as needed.
- Receive certain gases discharged from other plant systems and process them for:
 - reuse (recycled argon),
 - safe release to the environment, or
 - safe storage and disposal.
- Provide for the IGRP system's own safe, reliable, and efficient operation.

These functions are provided by four subsystems: (1) supply and distribute argon (argon distribution subsystem), (2) supply and distribute nitrogen (nitrogen distribution subsystem), (3) receive and process recycled reactor and primary heat transport system (PHTS) cover gas [radioactive argon processing subsystem (RAPS)], and (4) receive other discharged gases and process them for release to the environs [cell atmosphere processing subsystem (CAPS)].

During the later stages of the final design effort, additional functions were added to measure the radioactivity in the reactor cover gas following a large-scale accident in which the fuel cladding is ruptured and to monitor certain nonradioactive impurities in the reactor inlet and outlet cover gas.

12.1.2 Descriptions

12.1.2.1 System Features

The IGRP system maintains liquefied and ambient gas supply storage, delivers inert gases of specified composition and purity at regulated flow rates and pressures to points of usage throughout the CRBRP, and accepts the contaminated gases through its vacuum and compressor facilities for processing and storage before reuse or vent to the heating, ventilating, and air conditioning (HVAC) system. Argon is supplied for sodium cover gas, NaK cover gas, fuel handling cell (FHC) atmosphere including the atmosphere purification unit (APU), ports and floor valve inerting, and seal services. Nitrogen gas is provided for inerted cell atmospheres, valve actuation in inerted cells, cover gas for control and mitigation of sodium-water reactions, and gas for component cleaning and bagging operations. Vacuum vessel, compressor, cryostill, charcoal bed, and storage vessel equipment is used in fill, transfer, and radioactive gas decontamination operations.

Design requirements for System 82 components are defined in Section 1.2 of System Design Description (SDD) 82 (Ref. 1). Specific design bases for the IGRP subsystems are provided in SDD Sections 1.2.3, 1.2.4, and 1.2.5. The detailed design of the system is described in SDD Section 2 and supplemented by the appendices containing the system piping and instrumentation diagrams (P&IDs), interface requirements, and equipment lists.

The overall system is designed and arranged to meet the inservice inspection (ISI) requirements of Westinghouse Advanced Reactors Division (WARD) D-0244, "CRBRP Preservice and Inservice Inspection Requirements in Sodium and Cover Gas System," (Ref. 2) and the "as low as reasonably achievable" (ALARA) criteria of WARD-D-0177, "CRBRP ALARA Criteria and Guidelines," (Ref. 3).

12.1.2.2 Safety and ASME Code Classifications

The final safety and American Society of Mechanical Engineers (ASME) Code classifications of System 82 components are listed in Table 12-1. The safety classes are defined in Section 1.3.1.6 of Overall Plant Design Description (OPDD-10) (Ref. 4)

12.1.2.3 Argon Distribution Subsystem

The argon distribution subsystem includes two sets of liquid argon supply dewars with vaporizers to provide gaseous argon; four auxiliary gaseous argon supplies and one gaseous hydrogen supply (high-pressure bottles); the necessary valves and piping for distribution of the gas to usage points; equipment such as condenser vapor traps, freeze vents, and FHC/APU; and associated instrumentation and controls. The argon gas and liquid are of commercial-grade purity that meets the requirements of Reactor Development and Technology (RDT) Standard A 1-5T (Ref. 5) and RDT Standard M 14-1T (Ref. 6).

This subsystem also receives and accumulates recycle argon from the RAPS subsystem; manages the flow and pressure control of the recycle argon cover gas for the reactor, overflow vessel, and PHTS; and provides pressure equalization among these components. It collects and returns the recycle gas to RAPS for purification. The recycle argon meets the purity requirement established by interfacing systems.

The supplies and services are shown schematically in Figures 12-1, 12-2, and 12-3. The final design reports for the argon distribution components and I&C are Refs. 7, 8, and 9.

TABLE 12-1
SAFETY AND CODE CLASSIFICATIONS OF SYSTEM 82 COMPONENTS
(Sheet 1 of 2)

Compound or Type	Construction Code (ASME B&PVC, Section and Class)	Safety Class
CAPS cold box and components	III-3	SC-3
CAPS compressors and coolers	Industrial	None
CAPS radioactive-gas filters	VIII	None
CAPS tritium-water removal unit	III-3	SC-3
CAPS gas vessels	III-3	SC-3
RAPS cold box and components	III-3	SC-3
RAPS radioactive-gas filters	VIII	None
RAPS process gas compressors and coolers	Industrial	None
RAPS gas vessels	III-3	SC-3
RAPS LN ₂ dewars	VIII	None
RAPS LN ₂ subcooler	III-3	SC-3
RAPS LN ₂ receiver	III-3	SC-3
RAPS LN ₂ vent vaporizer	III-3	SC-3
All other gas supply dewars	VIII	None
Gas supply vaporizers	B31.1	None
Gas supply filters, high pressure	III-3	SC-3
Pump oil supply tank argon accumulator	III-2	SC-3
Gas supply filters	VIII	None
Gas supply filters, high pressure	VIII	None
Ar recycle filters	III-2	SC-2
Auxiliary supply bottle racks	Industrial	None
Radioactive gas filters	III-3	SC-3
FHC atmosphere purification unit	VIII	None
Freeze vents--primary	III-1	SC-1
Freeze vents--EVS Na	III-2	SC-2

TABLE 12-1
SAFETY AND CODE CLASSIFICATIONS OF SYSTEM 82 COMPONENTS
(Sheet 2 of 2)

Compound or Type	Construction Code (ASME B&PVC, Section and Class)	Safety Class
Freeze vents--intermediate	III-1	SC-2
Freeze vents--intermediate cold traps	III-3	SC-3
Freeze vent--EVS bypass Na cooler	III-2	SC-2
Freeze vents--EVS Na cooler	III-2	SC-2
Freeze vent--overflow heat exchanger	III-1	SC-1
Vapor traps, condenser	III-2	SC-2
Vapor traps, filter--reactor vent lines	III-2	SC-2
Vapor traps, filter, pressure instruments	III-2	SC-2
Vapor traps, filter--tanks	III-3	SC-3
Vapor traps, filter--EVS	III-2	SC-2
Vapor traps, filter--IHTS	III-3	SC-3
Vapor traps, oil--primary	III-2	SC-2
Vapor traps, oil--intermediate	VIII	None
Recycle argon storage vessel	III-2	SC-2
Vacuum pumps	Industrial	None
Compressor--accident monitoring	III-2	SC-2
N ₂ sampling and analysis units	Industrial	None

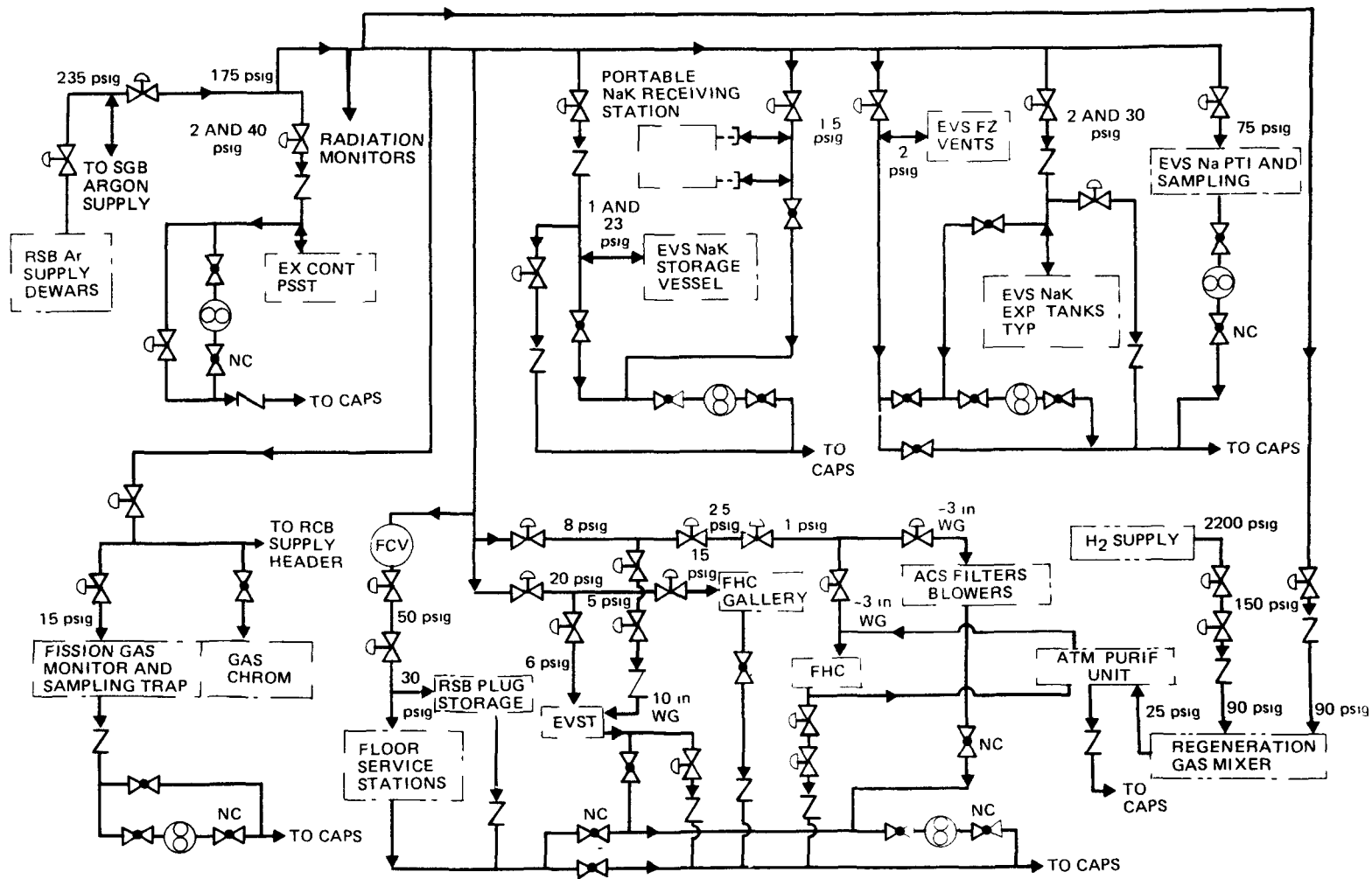


Figure 12-1. Simplified Flow Schematic of Reactor Service Building (RSB) Argon Distribution

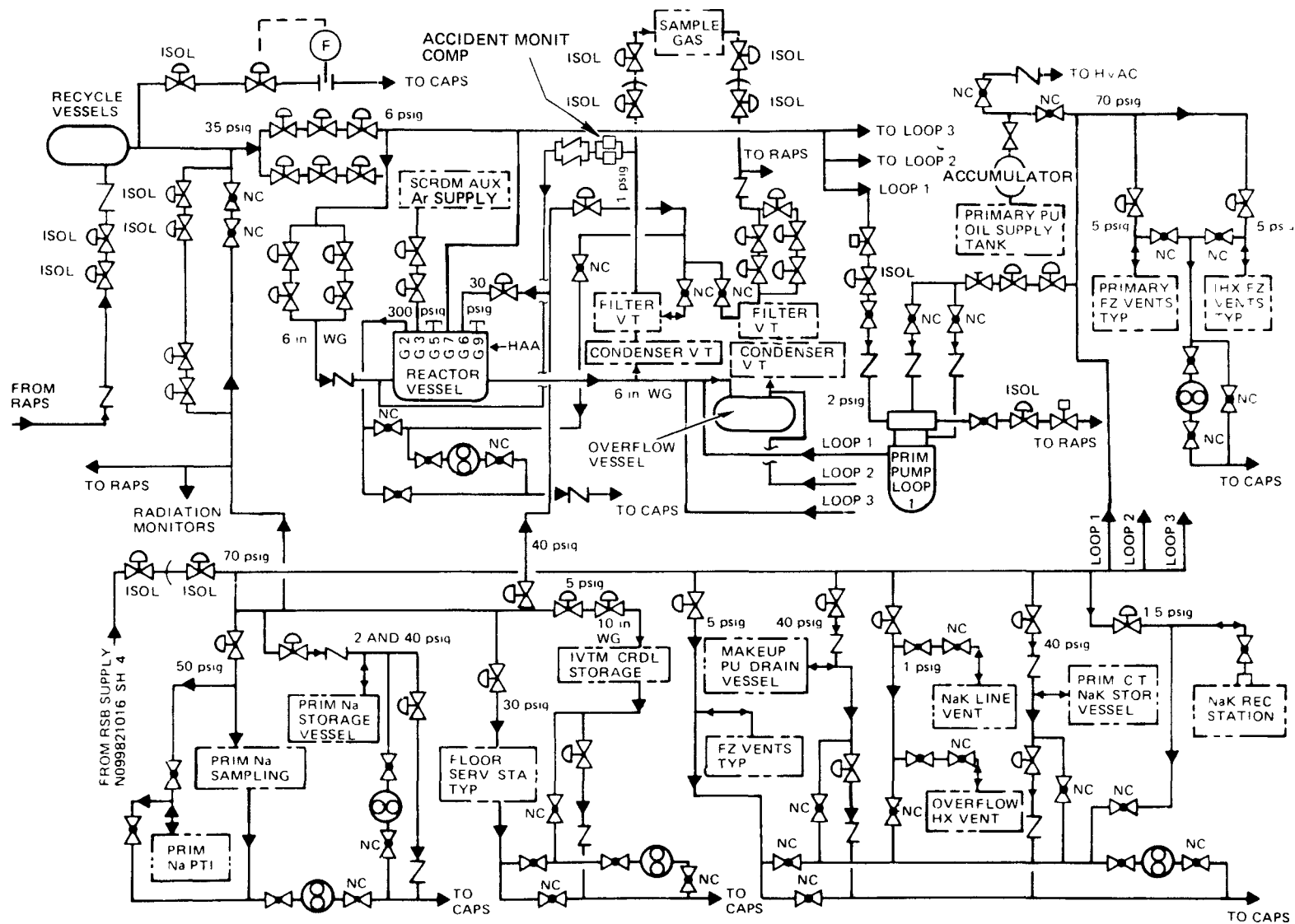


Figure 12-2. Simplified Flow Schematic of Reactor Containment Building (RCB) Argon Distribution

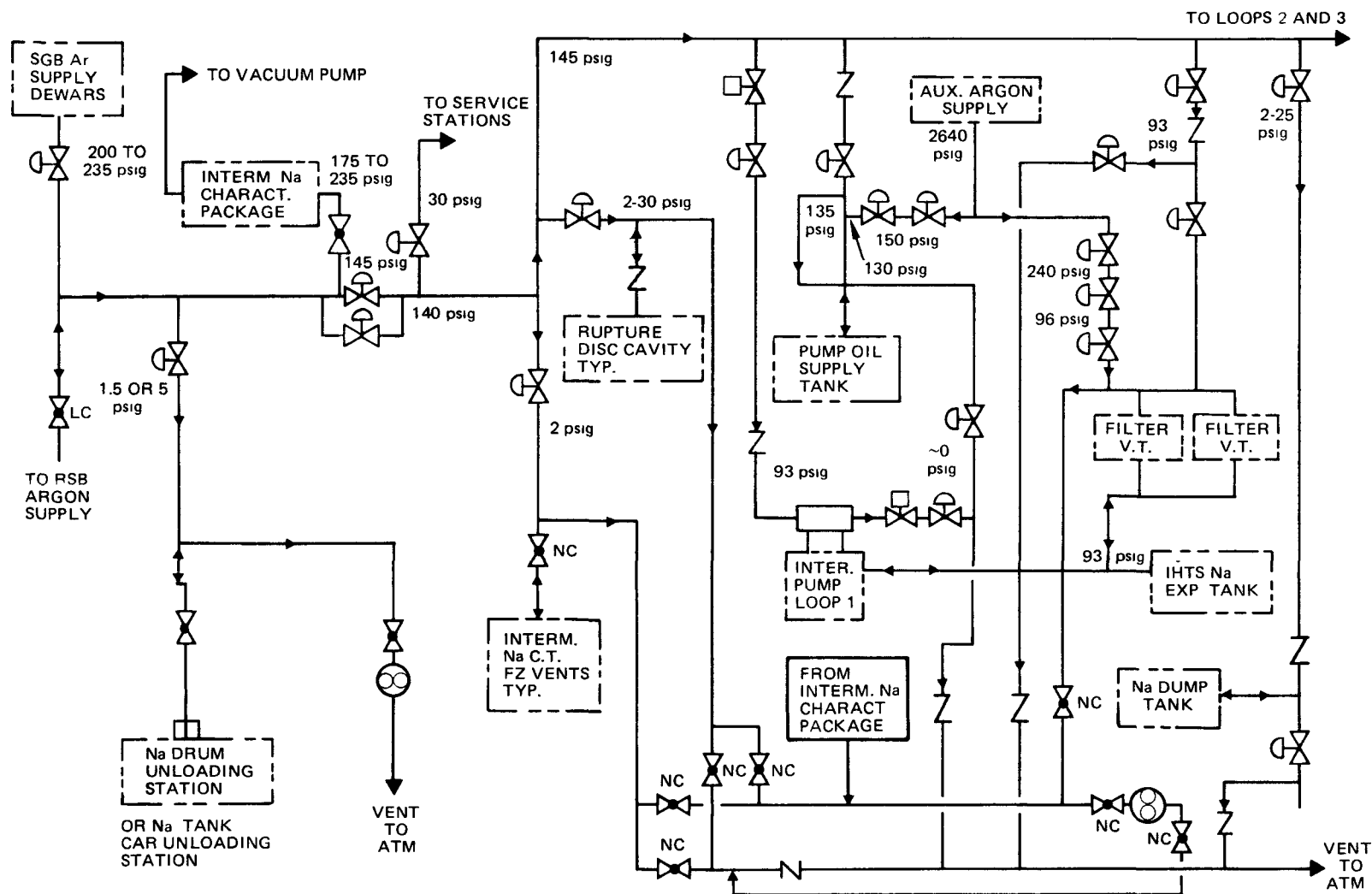


Figure 12-3. Simplified Flow Schematic of Steam Generator Building (SGB) Argon Distribution

12.1.2.4 Nitrogen Distribution Subsystem

The nitrogen distribution subsystem consists of three sets of liquid nitrogen supply dewars with vaporizers to provide gaseous nitrogen, three banks of auxiliary gaseous nitrogen supplies (pressure bottles), and the necessary valves and piping for distribution of the gas to usage points. The nitrogen gas and liquid are of commercial-grade purity. The supplies and their services are shown schematically in Figures 12-4 and 12-5.

The supply at the reactor service building (RSB) pad provides nitrogen to the RSB and reactor containment building (RCB). One of the two supplies at the steam generator building (SGB) provides the normal needs of the SGB, the sodium maintenance area (SMA), and the hot shop; the other supply provides nitrogen in the event of sodium-water reaction accidents.

The final design reports are to be found in Refs. 9, 10, and 11.

12.1.2.5 Radioactive Argon Processing Subsystem

The RAPS subsystem consists of a receiver (vacuum) vessel, two compressors, an accumulator (surge) vessel, a cold box that contains a cryogenic still (cryostill), liquid nitrogen supply components, a noble gas storage vessel, and associated instrumentation, controls, and piping.

The RAPS process removes mainly krypton and xenon from the cover gas, by distillation, thereby reducing the concentration of both the radioactive fission products and the stable isotopes of Kr and Xe that are incorporated in the fuel as tags for fuel-failure location purposes. Radioactive gas input rates to the cover gas are given in WARD-D-0193 (Ref. 12). The radio-nuclides decay in the cryostill; the cryostill contents are off-loaded

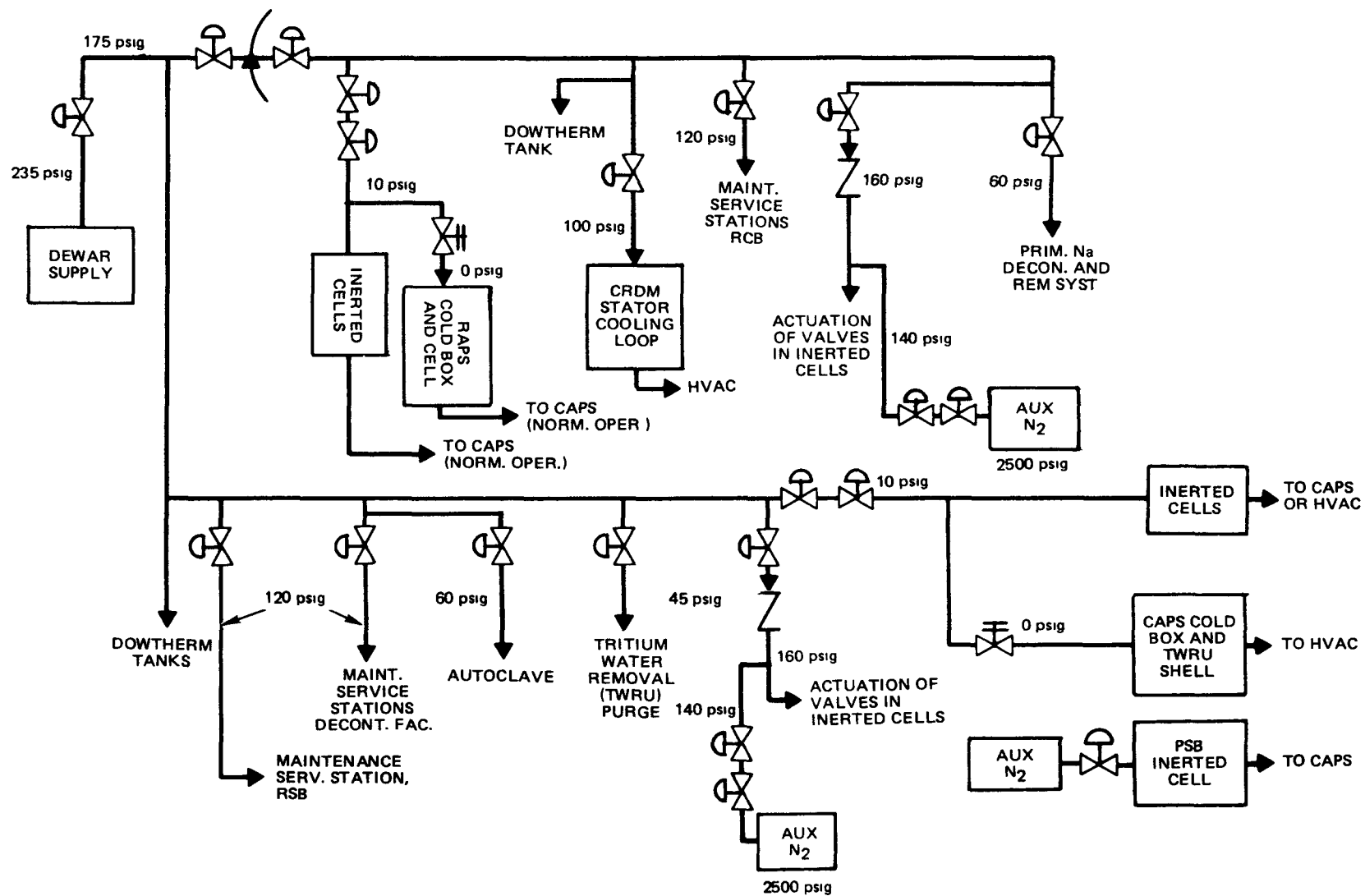


Figure 12-4. Simplified Flow Schematic of RSB/RCB Nitrogen Distribution

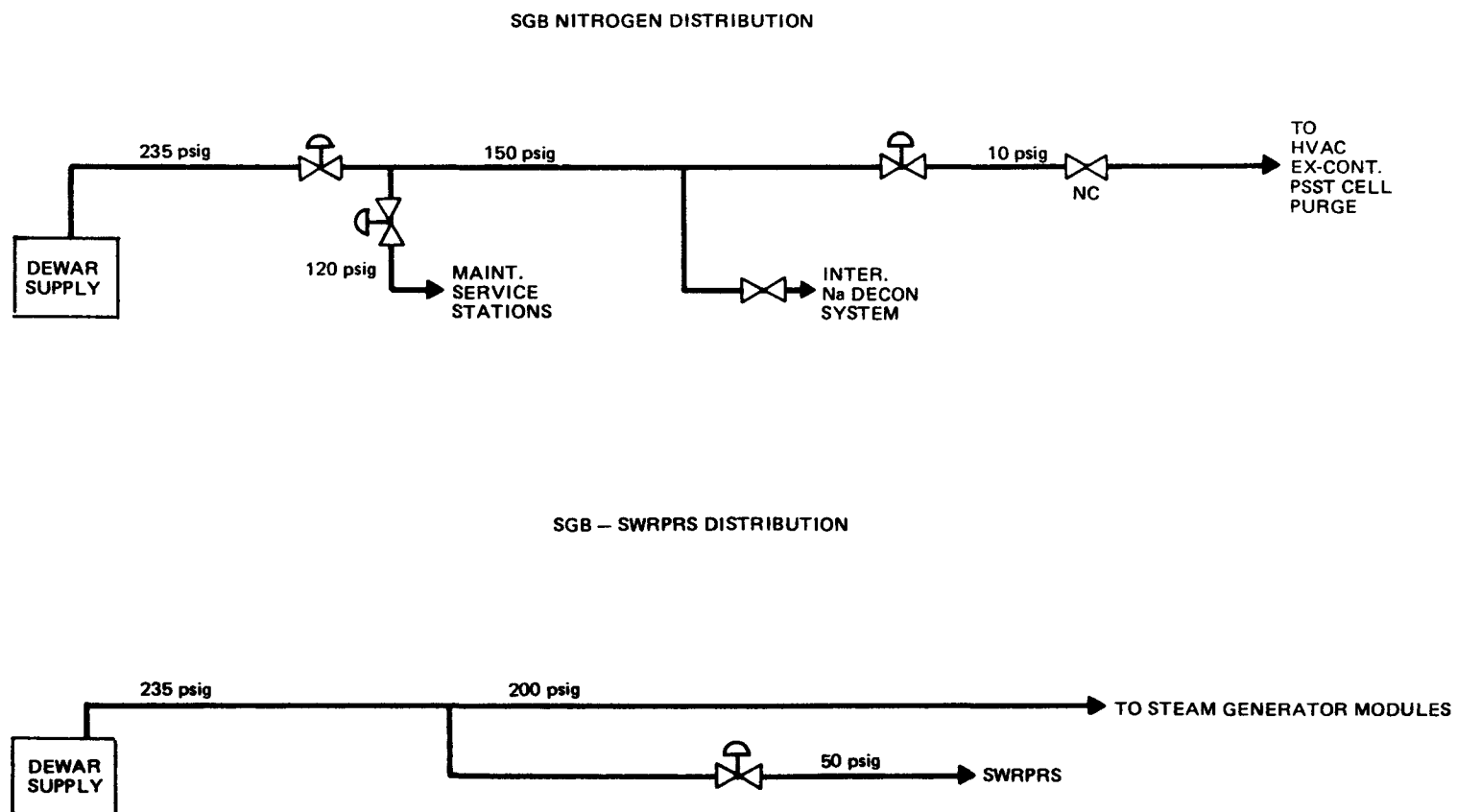


Figure 12-5. Simplified Flow Schematic

annually to the noble gas storage vessel and bled slowly through CAPS. The purification process is continuous. A schematic is shown in Figure 12-6. The recycle argon cover gas loop, which includes RAPS, is "closed" except for leakage and planned withdrawals.

In order to keep the process continuous and to stabilize the separation function of the cryostill, a constant 10-scfm flow is controlled through the cryostill. A recirculation loop is provided, which normally returns to the vacuum vessel 4.85 scfm of the cryostill output flow. Whenever the recycle argon storage vessels (in the argon distribution system) are below normal pressure, the total 10-scfm output of the cryostill is delivered to the recycle vessel, and the recirculation flow to the vacuum vessel is stopped.

The input to RAPS is normally 5.15 scfm, but this flow rate will have both slight continual variations and major perturbations, the latter occurring with any large change in sodium power level. The vacuum vessel and surge vessel are designed to accommodate these variations. The output of RAPS to the recycle argon storage vessel is also at a nominal 5.15 scfm.

A cold-box bypass line within containment permits RAPS to continue to circulate cover gas when the cold box is not operational; this permits maintenance of the cryostill, including an annual drain, without disrupting reactor cover gas flow.

The final design analyses for this subsystem can be found in Refs. 9, 13, and 14.

12.1.2.6 Cell Atmosphere Processing Subsystem

The CAPS subsystem receives and processes argon, nitrogen, and air vented from cells, components, and services wherever these gases might contain significant

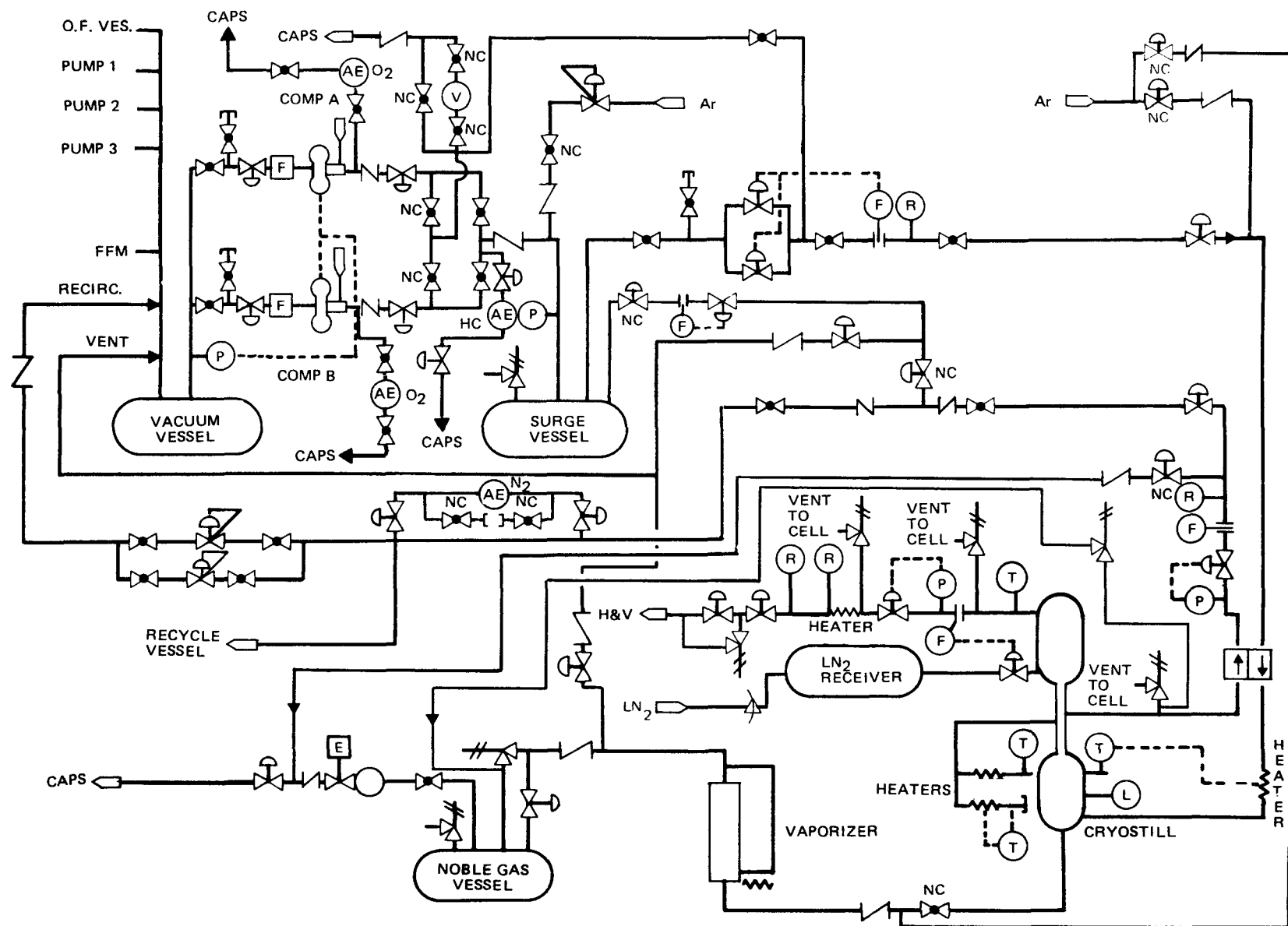


Figure 12-6. Radioactive Argon Processing Subsystem

amounts of radioactivity. The CAPS process reduces the radioactivity below the 10 CFR 20 maximum allowable limits prior to its release from the plant through the heating and ventilating exhaust system. The branch lines for initial collection of the gases are shown in the P&I diagrams of the argon and nitrogen distribution subsystems and of the radioactive argon processing subsystem; these lines, in turn, feed to the CAPS inlet header. The specific sources are discussed individually in Section 2.2.4.4 of the SDD and are indicated by major categories of sources in the diagram of Figure 12-7. The CAPS processing equipment, also indicated in this figure, is designed to delay the gaseous input for a sufficient time period to permit substantial radioactive decay prior to release of the gas to the environment. Radioactivity inputs to CAPS, effective delay times, decay (decontamination) factors, and design radioactive releases are given in the text of the SDD and tables of Appendix I of the SDD.

The final design analyses for this subsystem are documented in Refs. 15 and 16.

12.2 Evolution of System 82 Design

The design of the IGRP system was a continuous effort starting at the beginning of the CRBRP in 1973 and continuing after the Final Working System Design Review of System 82 in 1983. Design changes were in process at the termination of CRBRP. These design changes resulted from design evolution, design improvement, Fast Flux Test Facility (FFTF) experiences, cost reduction, improved safety, changing requirements from interfacing systems, and changing requirements in OPDD-10. Only major design changes will be described in the following sections.

12.2.1 Conceptual Design

At the beginning of the System 82 design effort, every effort was made to utilize the then current design of the FFTF Inert Gas Receiving and Processing System as the reference for the CRBRP design. The FFTF design was modified to

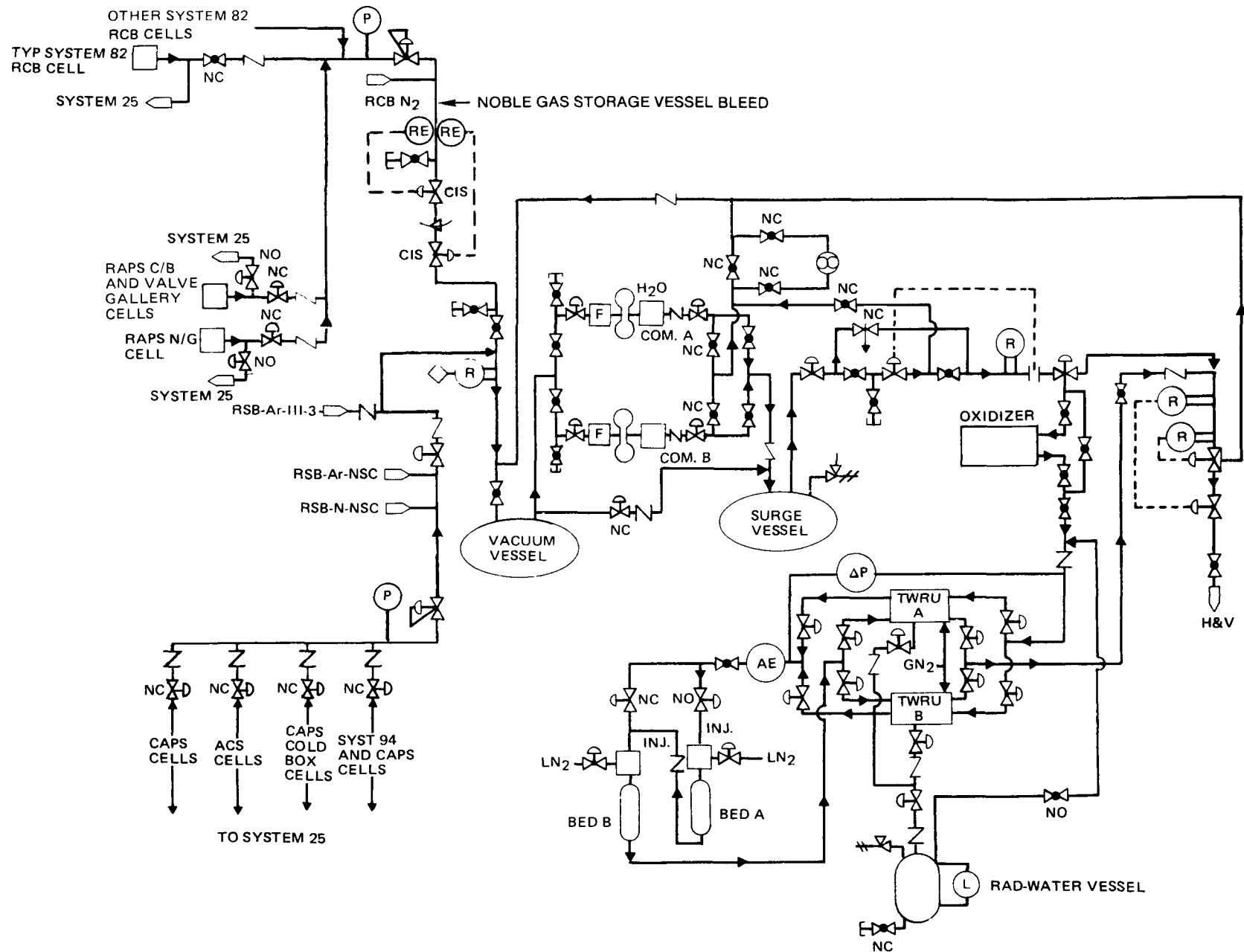


Figure 12-7. Cell Atmosphere Processing Subsystem

interface with the different CRBRP system requirements, but the designs of components were not changed during the initial stages of the conceptual design. At this time, in addition to the four subsystems listed in Section 12.1.1, there was the Fuel Handling Cell Atmospheric Purification Subsystem and the Vacuum Subsystem.

During the conceptual design phase, System 82 responded to the many changing interface requirements from other systems such as the addition of a second ex-containment primary sodium storage vessel by System 81.

At the time of the Conceptual Design Review held on February 3-6, 1976, the Inert Gas Receiving and Processing System consisted of SDD 82, Revision 2, with the subsystems described below.

The argon distribution subsystem was designed to provide and maintain argon cover gas for liquid metal systems and components and supplies argon for purging, filling, and draining the liquid metal systems. It also was to supply argon to the buffered and inflatable seals and to provide argon services for fuel handling, sampling, and maintenance. The subsystem hardware consisted of liquid argon dewars with vaporizers, gaseous argon bottles, piping, valves, vapor traps, freeze vents, filters, pressure relief assemblies, a recycle argon storage vessel, and control systems. These components were located throughout the plant buildings to interface with other systems.

The nitrogen distribution subsystem was designed to distribute and control the composition of the inert gas atmosphere in selected sodium component cells, to provide gas for valve actuation in inerted cells, to provide an inert carrier gas for cleaning operations, and to provide a fire-control blanketing gas flow for intermediate bay cells. The subsystem hardware consisted of liquid nitrogen dewars with vaporizers, gaseous nitrogen bottles, piping, valves, filters, and control systems. These components were located wherever using systems were installed.

The radioactive argon processing subsystem was designed to remove krypton and xenon radioisotopes from the argon reactor cover gas and to deliver the purified argon to the recycle argon storage vessel. The subsystem is to process gas from the reactor cover gas space, the exit gas stream from the PHTS sodium pump seals, and the failed fuel monitoring system excess flow. The components consisted of a vacuum vessel, two compressors, a surge vessel, four cryogenic charcoal beds, a cryogenic distillation column, two repressurization blowers, a noble gas vaporizer, a noble gas storage vessel, a noble gas compressor and cooler, a noble gas bottling station, piping, valves, filters, and control systems. These components were in the reactor service building.

The cell atmosphere processing subsystem was designed to process gases from any source, either in-containment or ex-containment, that contains radioactive gaseous impurities before discharge to the atmosphere. The subsystem hardware consisted of piping, valves, a vacuum vessel, two cryogenic charcoal beds, compressors, a surge vessel, two tritium oxidizer-dehydrator trains, two repressurization blowers, and control systems. CAPS components were all situated in the reactor service building.

In accordance with the comments from the Conceptual Design Review (Ref. 17), Revision 3 of SDD 82 and Revision 0 of all the System 82 P&IDs were released in January 1977 and should be considered as the conceptual design of the IGRP system. During the evolution of the conceptual design, the FHC Atmospheric Purification Subsystem was made a function of the Argon Distribution Subsystem and the Vacuum Subsystem was made part of CAPS so that the four subsystems described in Section 12.1.1 constituted the IGRP system.

12.2.2 Preliminary Design

During the preliminary design phase, most of the design effort was spent in revising System 82 to provide the services required by interfacing systems, as they were revised during this time. Design changes were also implemented on the basis of FFTF experiences and recommendations as shown in Ref. 18. Additional design changes are presented by subsystem.

12.2.2.1 Argon Distribution Subsystem

The recycle argon storage vessel was relocated in the containment vessel; to provide additional reactor scram capability, the one vessel was increased to two of the same volume.

As argon requirements were refined by interfacing systems, the need was reduced so that two liquid argon dewars were eliminated from the design.

Mobile vacuum stations were replaced by permanent vacuum pumps to provide specific service. Similarly, mobile auxiliary gas supplies were modified to permanent bottle racks to safely hold gas bottles at the required location.

To improve plant safety, an analysis was performed to determine the safest position for pneumatic valves in case of failure of these valves. This information was added to the P&IDs. Also, the venting of all dewars was required to be directed to the atmosphere to prevent the exposure of personnel to asphyxiating atmospheres.

The preliminary design review of the argon distribution subsystem was held on February 13 and 14, 1979, with Revision 13 of SDD 82 as the basis. In response to the comments in Ref. 19, the subsystem was revised and the preliminary design was defined in Revision 21 of SDD 82.

12.2.2.2 Nitrogen Distribution Subsystem

As stated in Section 12.2.2, the nitrogen distribution subsystem design was modified to meet additional or eliminated requirements or changing requirements. To provide improved control of the impurity level in nitrogen inerted cells, the nitrogen sampling and analysis units were modified to provide automatic purge on high moisture and/or high oxygen content in the cell atmosphere.

The preliminary design review of the nitrogen distribution subsystem was held on May 16 and 17, 1978, using Revision 9 of SDD 82 as the basis for the review. After responding to the design review comments in Ref. 20, the preliminary design was released in Revision 13 of SDD 82.

After the design review, System 26A, Sodium Fire Protection, deleted the requirement for nitrogen flooding of air-filled cells in the intermediate bay containing sodium components, and other nitrogen requirements were reduced so that System 82 could delete one 3000-gal liquid nitrogen dewar and a total of seven ambient cryogenic fluid vaporizers from the subsystem design. This change was incorporated in the preliminary design provided in Revision 13.

12.2.2.3 Radioactive Argon Processing Subsystem

As part of system simplification, the RAPS vacuum vessel, compressors, and surge vessel were moved to the containment building. As system requirements were changed and as a result of analysis, the single surge vessel was first increased to two vessels of the same volume of the single unit and then the design was modified back to a single vessel.

Analysis of the noble gas bottling concept for cryostill bottoms removal indicated that the cryostill bottoms after decay holdup in the noble gas storage vessel could be slowly vented to CAPS for processing before release to the atmosphere. Therefore, the noble gas bottling station, the noble gas transport cask, and the noble gas compressor with aftercooler could be deleted from the design. An analysis of the RAPS cryostill showed that this component could purify the recycle argon to an acceptable level without the use of charcoal delay beds so the four delay beds in the RAPS cold box were deleted.

As part of the IGRP system change, vacuum stations in RAPS were replaced with permanently installed vacuum pumps.

The preliminary design review of RAPS was held on February 13 and 14, 1979, with Revision 13 of SDD 82 as the basis. The subsystem was revised in response to the comments in Ref. 19 and the preliminary design was released as Revision 21 of SDD 82.

12.2.2.4 Cell Atmosphere Processing Subsystem

As a result of changing requirements, the inputs from the IGRP system and interfacing systems to the CAPS headers were revised.

The design of the Tritium Removal Unit-oxidizer and the Tritium Removal Unit-dryer was changed to a tritium oxidizer with a regenerative heat exchanger and redundant cryogenic tritium water removal units with regeneration capability to eliminate the effect of water freezing out on the cryogenic charcoal beds in the CAPS cold box.

As part of the ongoing effort to simplify design and reduce cost, the CAPS cold box heat exchangers, blowers, and recirculating line were replaced with liquid nitrogen injectors to maintain cryogenic temperatures in the cold box delay beds.

The preliminary design review of CAPS was held on May 16 and 17, 1978, using Revision 9 of SDD 82 as the basis. In addition to design changes made in response to Ref. 20, vacuum stations were replaced with permanent vacuum pumps, and the cell group inputs to CAPS in-containment and ex-containment headers were defined in greater detail. These changes were reflected in Revision 13 of SDD 82 which provided the preliminary design of CAPS.

12.2.3 Final Design

In addition to design revisions in response to changes in requirements from interfacing systems, changes were made from the preliminary design as a result of IGRP system design evolution; in response to recommendations from Key

Systems Task Forces on pump level control, seismic design, radioactive waste, the control room, and inert gas; and as a result of recommendations from FFTF based on their experiences. The Key System Task Force's responses are detailed in Ref. 21, and the FFTF recommendations and responses are detailed in Ref. 18. Subsystem design changes of importance are detailed in the following sections.

12.2.3.1 Argon Distribution Subsystem

In order to resolve cell wall penetration problems from NaK-filled pressure transmitters, these units were replaced by heated pressure sensors, which permitted the use of simpler penetrations. As part of a system-wide design revision, root valves were added to instrument lines. Guard pipes were designed around the argon supply line and the cover gas sample lines to and from System 94, Fuel Failure Monitoring System, penetrating the containment/confinement annulus.

Among a large number of changes due to interfacing system changes, vapor traps were deleted from System 81 NaK systems. Also, freeze vents were deleted from the PHTS intermediate heat exchanger (IHx) outlet line vent.

The notation "Nuclear Safety Related" was added to those P&IDs to which it applied, and IE power channel symbols were added to applicable instrumentation to show the intent for power supply separation.

A plant incident with impact on System 82 freeze vents for liquid metal components was identified as the loss of cell cooling, the result of which is that cell temperatures rise above the melting point of the sodium plug in freeze vents. To prevent the in-leakage of argon cover gas or out-leakage of sodium through overheated freeze vents on the high points of the PHTS and IHx which could cause siphon breaks in the PHTS piping, the freeze vent inlet and outlet piping was redesigned. The design changes involved the use of removable spool pieces with piping end caps and the replacement of packed stem valves with bellows valves.

The System 51A, Primary Heat Transport, interface to System 82 placed a limitation on the nitrogen impurity level in the recycle argon supplied to the PHTS pump cover gas spaces. Potential System 82 sources of nitrogen were the piping in RAPS which was below atmospheric pressure (compressor inlet) and the condenser in the RAPS cryostill which separated the liquid nitrogen coolant from the cover gas. In order to detect leakages from these sources as well as from the reactor, a continuous nitrogen monitor with a cover gas sampling station was installed in the inlet to the recycle argon storage vessels. At the same time, a connection from the recycle storage vessel outlet to CAPS was provided to allow purging with fresh argon to decrease the recycle nitrogen level.

Since all of RAPS was now in containment, the only source of primary cover gas out of containment under normal conditions was the cover gas stream provided to System 94, Failed Fuel Monitoring. To prevent flow of radioactivity out of containment, which if released to the environment would exceed allowable site boundary doses, a flow restrictor and radiation monitors installed in the System 82 piping to System 94 were used to control the position of the existing containment isolation valves, which were also under the normal containment isolation signal control.

Since radiation monitors had been located in the piping from the 1-cfm vapor trap to System 94, as discussed above, an additional requirement to monitor the cover gas radioactivity level in the reactor following massive fuel cladding failures was imposed. To provide this service, two redundant compressors contained in ASME Section III Class 2 pressure vessels and powered from the 1E supply were installed in a sampling loop from the System 94 line, which discharged to the reactor inlet. These components were located in the cell previously assigned to System 85 for its auxiliary plugging temperature indicator. This design allowed for the continuous monitoring of the reactor cover gas after the System 94 sample inlet containment isolation valves were closed.

The System 82 gas services to the PHTS pumps were modified to provide automatic isolation of the gas inlet to the pump and the gas outlet from the lower seal oil leakage tank upon the receipt of high or low pump tank sodium level signals. Additionally, argon accumulator tanks were installed at the primary pump oil tanks to provide the equivalent of 90 days gas usage without any additional gas supply.

12.2.3.2 Nitrogen Distribution Subsystem

In response to overall IGRP system design changes, root valves were added to all instrument lines. As discussed in Section 12.2.2.1, the safe failure positions for pneumatic valves were added to the nitrogen subsystem P&IDs.

System interfaces were clarified and revised to meet changing requirements. A nitrogen supply header was provided in the plant service building for the System 85, Impurity Monitoring and Analysis, glove box and analysis cell. The nitrogen supply piping to the steam generator loops was redesigned to provide additional gas services to System 53, Steam Generator, and System 56, Reactor Heat Transport Instrumentation. A guard pipe was designed for the nitrogen supply line penetrating the containment/confinement annulus.

12.2.3.3 Radioactive Argon Processing Subsystem

The safe failure positions for pneumatic valves and instrument line root valves were added as part of the overall IGRP system design change.

In response to a recommendation of the Inert Gas Key System Task Force (KSTF), the RAPS cold box and noble gas storage vessel were moved to the containment building with the deletion of containment isolation valves and penetration. To provide liquid nitrogen (LN_2) to the cold box in its new location, an LN_2 subcooler was added in the reactor service building near the LN_2 dewars to subcool the liquid nitrogen for transport to the LN_2 receiver in the containment building. This added receiver was sized to provide more than

1 day's supply in case of containment isolation. An LN₂ vent vaporizer was added near the receiver to increase the coolant flow whenever the transfer line temperature rose. The added piping, containment isolation valves, and penetration are all designed as cryogenic service components. A guard pipe was included for the liquid nitrogen supply line to RAPS.

The legend, "Nuclear Safety Related," was added to all applicable P&IDs, and IE power channel symbols were added to the appropriate instrumentation.

The RAPS compressors and associated system were redesigned to allow for the use of commercial-grade compressors. In order to use such compressors in RAPS, suitable cell radiation monitors and controls were added to isolate the compressors in case of failure. The hydrocarbon monitor of the gas space between plies of the diaphragm was deleted, and flow switches were added to the cooling water system for the compressor aftercooler to prevent operation without cooling water. Among the concerns of the Inert Gas KSTF was the potential for explosive mixtures being formed in the cryostill by the effect of radiation on mixtures of oxygen and hydrocarbons (see Section 12.3.6). Therefore, continuous oxygen monitors were installed on each compressor outlet to detect and measure oxygen levels due to air in-leakage at the reactor or at the below-atmospheric pressure piping of the compressor inlet. A continuous hydrocarbon monitor was installed at the inlet to the RAPS surge vessel to monitor the hydrocarbon level and to detect failure of the compressor diaphragm or an oil trap in the vent from each PHTS pump.

12.2.3.4 Cell Atmosphere Processing Subsystem

As part of the overall IGRP system design change, root valves were added to all instrument lines and safe failure positions were added to P&IDs for pneumatic valves.

A guard pipe was added for the CAPS pipe which goes through the containment/confinement annulus.

The legend, "Nuclear Safety Related," was added to all pertinent P&IDs, and 1E power channel symbols to show the separation of power supplies were added to the appropriate instrumentation.

As described in Section 12.2.3.3, the CAPS compressors were defined as being commercial grade and controls were added for isolation of the compressors in case of failure. Flow switches were added to the cooling water circuit for the compressor aftercoolers.

12.2.4 Design Changes from the Final Design Review

12.2.4.1 Final System Design Review

A Final System Design Review of System 82 was held on March 23 and 24, 1982, based on Revision 32 of SDD 82, Sections 1 and 2, and on the design reports, equipment specifications, and specification control drawings listed in Ref. 22, which reference also provides a list of action items. Responses to the Design Review Report were provided with the intent that they would be reflected in changes to the SDD. Not all of the responses were completed before termination. The remaining changes were minor in nature and could be resolved in a single document change.

12.2.4.2 Final Working System Design Review

A Final Working System Design Review (FWSDR) was scheduled for the IGRP system on April 4-6, 1983, with Westinghouse-LRM and the CRBRP Project Office. A design review package was transmitted by Ref. 23. This review package was updated for the design review and issued to the design review participants.

As part of the FWSDR presentation, Document N099TI820020, Ref. 24, was presented to show where the System 82 requirements were included in the system design documentation. Verification of the secondary interface requirements were included in the Final Design Reports, Refs. 7, 10, 13, and 15. Documentation was also provided to show the responses to Key Systems Task Force

recommendations (Ref. 21) and to the FFTF experiences (Ref. 18). Of the 185 KSTF recommendations assigned to System 82, five were not completed because they required prior action by other systems. All of the 64 FFTF experience items had been responded to in the System 82 design.

During the FWSDR, draft copies of comments on the design package were received from the LRM and PO participants. These comments consisted of corrections to the existing comments but without major impact on the design. Formal transmission of these comments was by Refs. 25, 26, and 27. System 82 responded in Refs. 28 and 29 to these comments. These System 82 transmittals identified a number of ECPs already prepared to change the system design in response to comments and identified those comments still to be implemented.

After the FWSDR, a consolidation meeting was held between the LRM and System 82 managers on August 25, 1983. During this meeting, the results of which were transmitted by Ref. 30, a number of ECPs were identified which would impact the design to varying degrees and were to be included in the final design of the IGRP system. These in-process ECPs would have the most effect on the system design:

- A82305: Sodium/Water Reaction Pressure Relief Subsystem (SWRPS) Piping Revision
- A82325: Downgrade CAPS to NSC, Seismic II
- A82324: Update SDD 82, Section 1, and Appendix G from FWSDR Comments and F2-2 Review Comments
- A82333: SDD 82, Section 2, and Appendices Revised to Meet FWSDR Comments
- A82353: Revision to Argon Purge Lines for Impurity Control.

12.3 Key Problems and Resolutions

12.3.1 Overpressure Protection

System 82 provided two types of overpressure protection, the design of which identified problem areas to be resolved for the final design. System 82 provided overpressure protection for its own vessels and piping designed to the ASME Boiler and Pressure Vessel Code, Section III; for cryogenic piping to prevent overpressure in isolated piping sections; and for cryogenic liquid storage vessels. System 82 also provided pressure relief systems for the cover gas spaces of liquid metal vessels.

12.3.1.1 System 82 Components

In order to meet the requirements of the ASME Code for relief of component overpressure, the pressure relief device should use the pressure in the cell containing the component to be protected as the pressure-relieving reference pressure and the overpressure should be vented to containment or to a closed vessel. This arrangement was used for the argon recycle storage vessels, the surge vessels in RAPS and CAPS, the noble gas storage vessel, and the CAPS cold box where the overpressure was vented to the same cell as the component and the pressure relief device. The cryostill in the RAPS cold box is protected by two relief valves; the lower set pressure device is vented to the noble gas storage vessel because of the large amount of radioactivity which would be in the cryostill and which should be safely stored in the storage vessel. The backup device is vented to the cell at a higher set pressure.

The argon gas accumulators for the primary pump oil supply tanks are located in accessible areas, and therefore the pressure relief devices on the accumulators are vented to a convenient H&V duct to eliminate the hazard to personnel from this asphyxiating gas.

In several instances, System 82 has provided pressure relief devices on gas supply lines to limit the pressure delivered to components at the request of other systems. These devices are arranged to vent to the cell containing the relief device. Overpressure protection analysis reports have been released for the recycle storage vessels and vapor traps.

When the RAPs cold box and noble gas storage vessel were relocated in the reactor containment building, cryogenic piping was added to the system design to transfer liquid nitrogen from the dewars located in the reactor service building to the RAPs cold box, and LN₂ receiver, LN₂ vent, and LN₂ sub-cooler components were added to the system design for this line. Each component and each section of pipe that is isolatable is provided with a pressure relief device. Since nitrogen can be an asphyxiant and the cells containing the cryogenic components are normally accessible to personnel, these pressure relief devices are vented to outside the building rather than the component cell. Additionally, the LN₂ supply dewars for this service are located on the operating floor of the reactor service building, and their pressure relief devices are also vented to outside the building for the same reason.

The argon and nitrogen dewars that furnish the plant gaseous needs are all located outside the buildings and so present no venting or reference pressure problems.

12.3.1.2 Liquid Metal Components

System 82 provides overpressure protection devices on gas lines connected to vessels containing liquid metal for Systems 41 and 81 as defined in the ICDs between these systems and System 82. Since the piping between the protected vessel and the relief device would contain sodium vapor, which can condense on the seat of the relief valve and affect the valve operability, all relief systems for liquid metal vessels were designed to consist of a rupture disc in series with a relief valve. The disc protects the valve seat from condensed sodium. Except for the overflow vessel, all liquid metal vessels are vented to the component cell and are referenced to the cell pressure.

At the time of termination, two rupture disc/relief valve assemblies provided overpressure protection for the System 81 overflow vessel--one connected to the gas outlet nozzle and one connected to the equalization line inlet to the vessel. Since the reactor is connected to the equalization line and thereby to the overflow vessel, these relief systems could provide overpressure protection for the reactor.

Two problems that had not been resolved at termination still remain in the design. One, based on conflicting sections of the ASME Code, is whether the total flow capacity is required for each relief system or for both. The second problem was the location of the vent discharge that also provided the reference pressure to the valve and rupture disc. In the current design, the relief systems both discharge to Cell 107B, while the overflow vessel is located in Cell 102. As stated above, neither of these problems had been resolved at termination.

12.3.2 Automatic Line Isolation Effects on Plant Operation

A problem came about where System 82 was revised in response to recommendations of the Inert Gas KSTF to limit the transport of radioactivity from the reactor containment building. The recommendations were to relocate the RAPS cold box and noble gas storage vessel to containment; to add safety-related radiation monitors and containment isolation logic on the System 82 inlet to System 94 to limit the radioactive inventory in System 94; and to add cell radiation monitors and automatic line isolation valves on the component in a given cell or other impacted components to limit leakage to the cell and to isolate the cell upon the detection of a component leak. In addition, the detection of radioactivity in the cell would also automatically close System 25A duct isolation valves for that cell, and the detection of excessive cell pressure, which would increase the cell leakage, would terminate the nitrogen or air supply to pneumatic valves in the cell. To provide additional control of the radioactive transfer from the reactor containment building,

safety-related radiation monitors were added to the CAPS header at the containment boundary to provide information to the operator in the MCR for closure of the containment isolation valves on the CAPS header to limit the amount of radioactivity in CAPS.

Such automatic source isolation, while protecting the public from the release of radioactivity, could impact plant operations due to the loss of a subsystem or component by a monitor failure or false alarm. The Inert Gas KSTF evaluated the effect of false closures on the plant.

The isolation of the RAPS cold box would cause the system to operate in the bypass mode, and the plant would function until the operator could verify the false alarm.

Monitor failure in either the RAPS vacuum or surge vessel cells would shut down RAPS and cause a lack of vent capability of about 1 h/year during verification of a false alarm.

Monitor failure in the recycle argon storage vessel cell would shut down all of RAPS and would isolate the recycle argon supply. While fresh argon could be supplied to the reactor and pumps, there would be no vent capability for about 0.3 h/year until the false alarm is rectified. Monitor failure in the recycle argon storage vessel cell or in the operating area of RAPS while the RAPS cold box is on bypass would isolate the recycle argon supply while the false alarm is verified. During this time fresh argon could be supplied to the reactor and pump as needed, but there would be no vent capability. Lost time due to this false alarm is expected to be about 0.5 h/year.

Monitor failure in either of the RAPS compressor cells would have no effect since this would cause a switch to the standby compressor with no loss of service. No loss of service is also true for a false alarm in the noble gas storage vessel cell.

A monitor failure in any of the CAPS component cells, except the compressor cells, would shut down the entire CAPS with the ensuing loss of cell venting among other functions. The downtime is about 2.5 h/year to verify the false alarm condition. Since cell venting and other CAPS services are not on a continuous basis in normal operation it does not seem that the loss of CAPS due to a false alarm would have any impact on plant operation.

While there is loss of vent capability when RAPS or CAPS components are isolated due to monitor failure, there is no effect on plant operation. If the components are isolated due to a component failure, it may be necessary to shut down the reactor until the component is repaired, especially if all of RAPS is lost.

12.3.3 Design of CAPS as a Nonsafety Class System

During the final design of CAPS, it became evident that the design requirements of the CAPS compressors could be reduced if certain precautions were taken to protect the public from radioactive releases from the reactor service building. Detailed analysis of the radioactive inventory in a compressor and the piping in the same compressor cell showed that the release of this inventory would be acceptable. Cell radiation monitors and isolation valves for the inlet and outlet piping and for the ventilation duct are used to isolate the failed compressor when radioactive gas is detected in the cell atmosphere. Based on these studies, the design requirements for the CAPS compressors were identified as being limited to commercial standards and are included in the current CAPS design.

In response to recommendations of the Inert Gas KSTF and as discussed in Section 12.3.2, the radioactive inventory of gas flowing to CAPS from the reactor containment building was limited to an amount which, if released by a failure in CAPS, would result in a radiation dose of less than 0.5 rem at the site boundary. As a result of this change, a study was made to evaluate changing all of CAPS to safety Class D or nonsafety class (NSC). This would allow the

design of the subsystem components to be made to standards other than those of the ASME B&PV Code, Section III.

A proposed design change was submitted for approval to classify all of CAPS as nonsafety class, even though many of the components had already been designed to Section III rules of the ASME code. In this proposed change, additional source isolation valves and cell monitors were added to components in the reactor service building, which vented to CAPS, to limit the inventory in CAPS to the equivalent of less than a 0.5-rem site boundary dose. The program was terminated before this proposed change could be approved.

In the current and proposed configurations, CAPS is limited in its usefulness for recovery from accident conditions. However, under suitable operator control, cover gas spaces with high radioactive inventories could be slowly vented to CAPS without exceeding isolation trigger levels. This will increase the time needed to clean up accidents but would still be shorter than the time required if the inventory were reduced by radioactive decay alone.

12.3.4 Deletion of Cryogenic Charcoal Beds from RAPS Cold Box

Since System 82 for CRBRP was based on the comparable system design for FFTF, the conceptual design of the RAPS cold box included four cryogenic charcoal beds upstream of the cryostill. The system was to be operated such that the short-lived radioactive xenons and kryptons were removed in the charcoal beds and long-lived radioisotopes and stable isotopes of xenon and krypton were removed in the cryostill to satisfy the impurity level requirements from the Failed Fuel Monitoring System (SDD 94).

During the preliminary design phase of RAPS, a trade study was made of various configurations of purification components in the cold box for simplification and cost reduction. The results of these studies showed that the cryostill by itself could remove all of the radioactive and stable isotopes except those of argon to such levels that would satisfy the requirements of System 94. These

analytical conclusions were verified by tests performed by a potential vendor of the RAPS cold box, the Japanese in their breeder program, and EBR-II personnel on an existing cryostill who found xenon and krypton levels in the cryostill effluent stream to be below the limits of detection.

As a result of these studies, the cold box design was simplified and the deletion of four cryogenic charcoal beds and related equipment lowered the cold box cost by about one million dollars. These changes were made without reducing the efficiency of the cold box.

12.3.5 Separation of Functionally Redundant Components

During the Inert Gas KSTF evaluations it became evident that the redundancy of components and the separation of components connected to PHTS loops had been lost by the routing of piping through common cells. Following the recommendations of this task force, the problems had been resolved or had been identified for correction if the program had not been terminated. Before revision, all of the following pipelines were routed through the PHTS Loop 1 cell, Cell 121, which represented a common mode failure for many System 82 services:

- Primary pump oil trap vent lines (all PHTS loops)
- Reactor/overflow vessel vent line to RAPS
- Recycle gas supply to reactor
- Fresh argon supply to reactor
- Nitrogen vent header for all RCB inerted cells.

The redundant RAPS and CAPS compressors were compromised by the isolation valve of one compressor being located in the cell with the backup compressor. These lines were rerouted to provide the required piping separation. When the

RAPS cold box and noble gas storage vessel were relocated to the containment building, the cell and piping arrangement required that the RAPS cold box bypass lines and the RAPS normal process lines be routed through different cells to provide the necessary separation.

At a later time, the gas services to the primary pumps were redesigned to provide the required loop independence and physical separation of pipe and components. At termination, work was under way to invoke similar requirements for loop separation for the IHTS cover gas services.

Even though there is increased cost, it is important that the piping for redundant components and redundant subsystems be routed to provide no common mode failure of the redundant functions.

12.3.6 Potential for Cryogenic Explosion in RAPS Cryostill

The concern about the known tendency of ozone to decompose spontaneously into oxygen and the potential for the detonative reaction of methane with oxygen or ozone caused the Inert Gas KSTF to investigate the problem and to make recommendations to the IGRP system.

It is expected that air could leak into the cover gas system and be carried to the cryostill. The oxygen that could go to the cryostill would have to enter the system downstream of the reactor since oxygen entering upstream of the reactor or at the reactor would react with sodium and be removed from the gas stream. Methane could form by degradation of the PHTS pump seal oil or by leakage of hydraulic fluid through the diaphragm in the RAPS compressors. These methane sources would be added to the argon cover gas flowing to the cryostill.

In the cryostill, the oxygen would be condensed and dissolved in the liquid argon in the still bottom. In a radiation field, such as that from the dissolved krypton and xenon radioisotopes, the oxygen will be converted to ozone,

and an oxygen-ozone equilibration would be established. As the ozone concentration is increased and the ozone forms a free, ozone-rich phase, it is possible for the ozone to convert to oxygen explosively. It has also been postulated, without verification, that a methane-rich solution in argon might react with the contained oxygen or ozone, also explosively.

During the Inert Gas KSTF study, a recommendation was made to limit the oxygen content in the cryostill bottoms to 4 mole percent and the methane to an amount that could not produce a pressure pulse in excess of the cryostill design pressure. Recommendations were made relative to sampling the cryostill bottoms to determine the drain period.

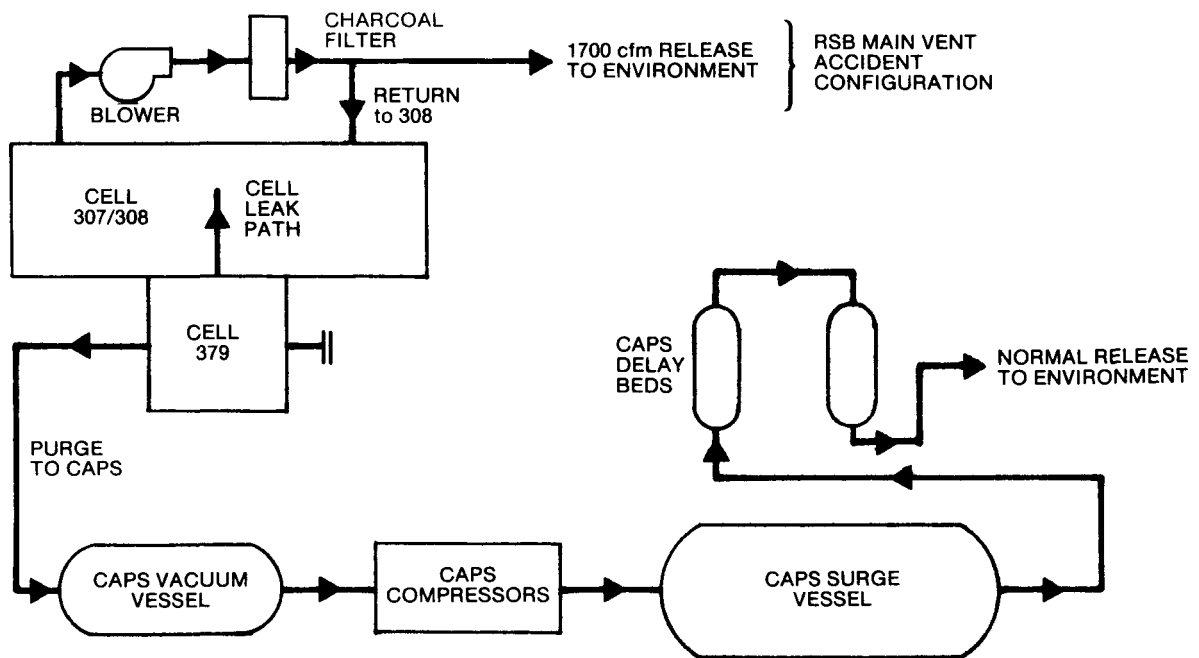
In response to these recommendations, System 82 managers submitted ECP A82177, which was subsequently incorporated into the system design, to provide continuous oxygen monitors on the outlet of each RAPS compressor and a continuous hydrocarbon monitor on the inlet to the RAPS surge vessel. The oxygen monitor signal was conditioned and fed to a time-summing algorithm in the Plant Data Handling and Display System (PDH&DS), which provided the running weight of oxygen entering the cryostill. The alarm set, an indication to drain the cryostill, was the weight of oxygen equivalent to 4 mole percent of the still bottoms. The hydrocarbon monitor provides a methane equivalent mass concentration signal to a similar time-summing algorithm. The alarm set is the weight of methane whose reaction with oxygen would cause the still pressure to exceed its design pressure. These alarm points assume 100% retention in the cryostill and are therefore conservative.

If these limits are not attained between the scheduled annual drains, the cryostill would be drained annually and the alarm sets for the impurity algorithm would be reset to zero before starting up.

12.3.7 Resolution of Conflict Between Design Based on Normal Exposure Levels and the Need to Recover from Accidents

The cell atmosphere processing subsystem (CAPS) is a subsystem of the inert gas receiving and processing system (System 82). CAPS was designed to accept low-level radioactive gas exhaust from inert or air-filled cells or from floor service stations serving the fuel handling and maintenance systems. CAPS treated this exhaust gas by either removing the radioactivity (when in the form of tritiated water vapor or particulates) or delaying it for decay (primarily radioactive krypton or xenon). The primary function of CAPS was to take low-level gaseous radioactivity generated during normal and anticipated operations and treat it so that it would meet the very low-level 10 CRF 20 limit for normal, ALARA releases, i.e., 5 mrem per year.

During a series of key systems reviews in 1979-1980, studies were made which demonstrated that the CAPS system could play a useful part in recovery from accidents involving releases of gaseous radioactivity much greater than those for which CAPS had been designed. There was a heightened sensitivity to providing such a capability in the wake of the TMI 2 accident. Figures 12-8 through 12-10 show the effectiveness of using CAPS for post-accident recovery, using a postulated instantaneous rupture of a tank containing a very large inventory of krypton and xenon isotopes in cell 379 of the reactor service building as an example. Figure 12-8 describes the release paths with and without the use of CAPS. Figure 12-9 shows the very high in-plant dose rates that might occur if such an extremely unlikely event as that postulated actually occurred, as well as the effect of using CAPS. Within minutes of being connected to the cell, CAPS would reduce cell pressure to subatmospheric and stop cell outleakage. If CAPS were connected within 1 to 2 h as shown, the amount of release would be limited to less than the allowable skin dose limit. Figure 12-10 shows similar beneficial results for site boundary doses. The site boundary doses are reduced, and the limiting dose is changed from a whole body to a skin dose. For example, the dose with CAPS is a factor of 2 less (Curve 2 versus Curve 3), and, in addition, the allowable dose limit



RELEASE PATH WITHOUT CAPS

- CELL 379 LEAKS TO OPERATING AREAS (CELLS 307 OR 308)
- CELL 307 HAS HIGHER DOSES BECAUSE OF SMALLER VOLUME ASSUME NO H&V FOR ON-SITE DOSE CALCULATION
- RELEASE TO ENVIRONMENT THROUGH RSB SAFETY RELATED H&V AT 1700 cfm (ACCIDENT CONFIGURATION)

RELEASE PATH WITH CAPS

- CELL 379 GOES TO NEGATIVE PRESSURE — RELEASE TO OPERATING AREAS STOPS WHEN CAPS PURGE BEGINS
- INVENTORY IN CAPS VACUUM VESSEL IS NEGLIGIBLE DUE TO LOW PRESSURE
- INVENTORY PEAKS IN CAPS SURGE VESSEL, THEN DECREASES RAPIDLY
- INVENTORY IS ACCUMULATED ON CAPS DELAY BED AND DECAYS SLOWLY OVER ~2 MONTHS RELEASE TO ENVIRONMENT THROUGH NORMAL RSB H&V ONLY Kr⁸⁵ REMAINS

Figure 12-8. Use of CAPS for Post-Accident Cleanup

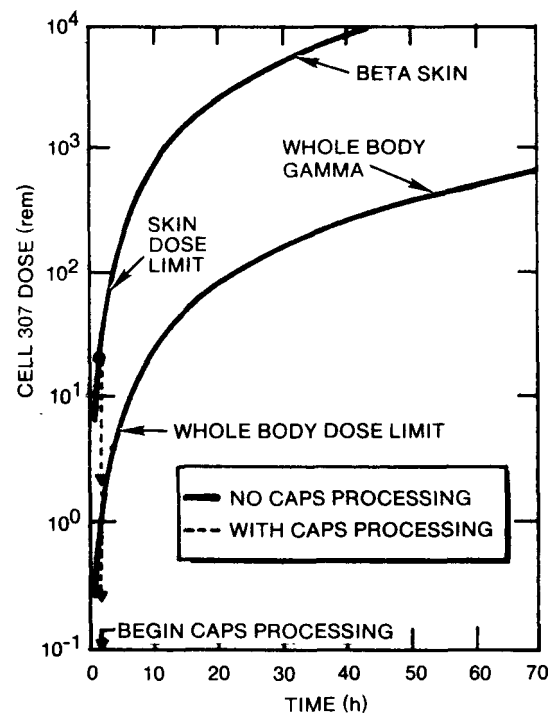


Figure 12-9. Dose in Cell 307 After Postulated Rupture in Cell 379

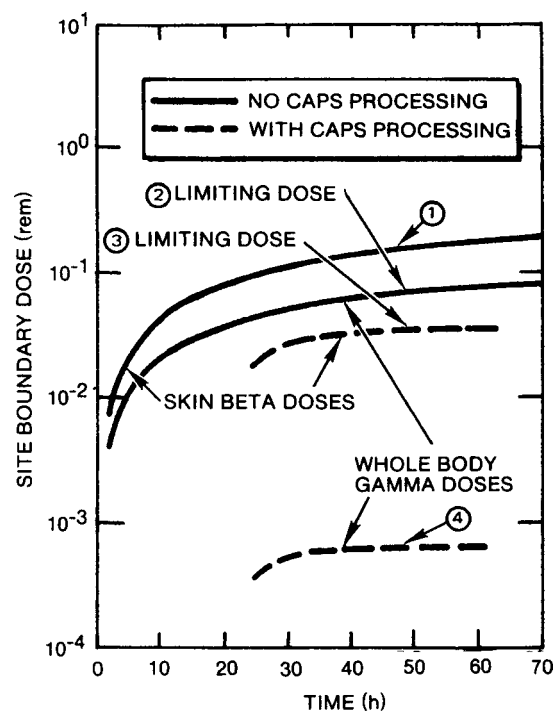


Figure 12-10. Site Boundary Doses After a Postulated Rupture in Cell 379

to the skin is a factor of 6 higher, so the net reduction is a factor of 12. Also, with CAPS the curves are rising less rapidly, and no release at all occurs until after breakthrough from the CAPS delay beds (approximately 1 day after the accident).

Three potential applications of CAPS for post-accident recovery were identified:

- Clean up cell gas following failure or leak in equipment containing gaseous radioactivity (as shown in the above example)
- Clean up ex-vessel fuel storage tank (EVST) or FHC following a fuel handling accident there or in ex-vessel transfer machine (EVTM) (if EVTm, purging to reactor and using RAPS is preferred)
- Assist in RCB cleanup following major incident.

A number of advantages were identified for this nonnormal mode of CAPS operation in these applications.

- There is better containment of radioactivity inside CAPS than at the accident location:
 - CAPS is an all-welded, safety-class system (for containment only)
 - CAPS equipment is or can be housed in relatively low-leakage cells
 - Even the most leak-tight cells are leaky by comparison to CAPS
 - The EVST, FHC, and reactor are more vulnerable to leakage in the fuel handling mode (i.e., port plugs removed, floor valves with inflatable seals in place) than is CAPS

- There are lower onsite and offsite doses when radioactivity is transferred from cells or fuel handling equipment to CAPS:
 - For cells, doses are reduced by up to several orders of magnitude
 - For EVST and FHC, doses may be lower but only if subsequent failure (leakage) occurs in EVST or FHC
- Following transfer of radioactivity to CAPS, there are more lines of defense (LOD) to prevent its release to the environment:
 - Compared to cells with leaky equipment, the initial LOD is restored
 - Compared to the EVST and FHC, the CAPS cells provide an additional LOD (EVST and FHC can leak directly to the RSB operating floor)
- Downtime following an accident is reduced by transferring the radioactivity to CAPS rather than waiting for its diminution by a combination of leakage and decay:
 - Cell access in approximately 1 week versus 1 or 2 months
 - Fuel handling can be resumed in less than 1 week versus 2 months
 - However, reactor containment building (RSB) radioactivity is reduced only slightly faster than with decay only (e.g., at a CAPS flow of 50 scfm, a factor of 10 reduction in gaseous dose occurs in 15 days versus 17 days with decay only).

The value of the reduced downtime using CAPS for post-accident recovery was evaluated quantitatively for later comparison to any cost impacts on CAPS resulting from its use in this nonnormal mode. The cost benefits of reduced downtime are as follows:

	<u>1974 \$/Day</u>	<u>1982 \$/Day</u>
Reactor downtime	100,000	192,000
Refueling extension	60,000	115,000
Delay in spent fuel handling	>4,000	>8,000

For example, if the refueling delay is reduced from 2 months to 1 week one time in 30 years by use of the CAPS, the present worth = $115,000 (60 - 7)/10 = \$600,000$.

Thus, if only modest changes in CAPS are required to permit it to perform post-accident cleanup, such use would be cost beneficial, as well as effective in reducing doses and recovery time. The exception to this is for assisting in RCB cleanup. The volume of the RCB is so large (approximately 1.5×10^6 ft³) compared to CAPS capacity (50 ft³/min), that this application was dropped from further consideration.

Having established the advantages of using CAPS in this manner, several analyses were made to evaluate postulated negative impacts on CAPS design, cost, and complexity as a result of planning to use it to assist in post-accident recovery. The major possible impacts were:

- Would the entire system have to be upgraded in terms of safety classification due to this added function involving much higher levels of radioactivity?
- Would an upgraded classification or designation as an engineered safety feature (ESF) result in additional requirements being imposed by government regulations? For example, redundancy of all active components, environmental qualification for a variety of postulated environments, etc.

- Would the fact that CAPS might contain moderate or high levels of radioactivity, instead of just low level, require changes or additional features from the standpoint of good design practice as contrasted to regulations?
- Could CAPS equipment accumulate sufficient radioactive inventory so that it would present a public safety hazard in case of subsequent rupture, and, if so, would added ESF be required to mitigate such a hypothesized event?

The evaluations to determine the impact on CAPS of a large amount of radioactivity entering CAPS following unlikely, extremely unlikely, or beyond-design-basis (BDB) events are summarized in the following paragraphs.

It is important that the events involved are all low probability accidents not associated with normal operation. From the standpoint of regulations, this means that the higher guidelines of 10 CFR 100 apply rather than the low limits of 10 CFR 20. From the standpoint of CRBRP design practice of providing features commensurate with the frequency of occurrence, this means that relatively few added features might be necessary because of the very low frequencies involved (i.e., less than once in the plant lifetime for large or medium releases).

- Because plant releases are within regulatory guidelines without requiring CAPS to perform this function, CAPS is not an ESF and by regulatory definition does not perform an "active safety function." Therefore, additional regulatory requirements associated with ESF or active components need not be imposed.
- Since CAPS is not required to be used immediately in order to be effective, and since it will not be put into service automatically for post-accident recovery, time will be available to inspect and check out the system to assure that it meets leak tightness requirements and

is operating properly. This will assure good containment and effective processing of any radioactivity transferred to CAPS and will reduce the probability of any subsequent inadvertent release from CAPS.

- Only the CAPS surge vessel and CAPS cold box could have large inventories (when used for post-accident recovery of large releases) as shown in Figure 12-11:
 - The surge vessel contains a large inventory, but only for between less than 1 and 3 days following the start of processing
 - The delay bed contains a large inventory for approximately 2 months following the start of processing
 - The delay bed would hold 99% of any radio-iodine even in case of failure. Similarly, particulate activity would also remain in place so only krypton and xenon would be available for release.
- Combination of (1) the original low probability failure event (unlikely, extremely unlikely, or beyond design basis) followed by (2) a large failure in the surge vessel or delay beds (very conservatively and nonmechanistically assumed to be an unlikely event) during (3) the short time the large inventory is present in the surge vessel or delay beds results in a beyond-design-basis event (frequency less than 10^{-7} year). This falls below the threshold generally used by NRC and industry to determine the need for design features to mitigate or prevent accident events. In other words, the probability of this hypothesized event is so low that it need not be designed for. However, because the source term would be present in the cold box for an extended time, it was recommended that the cold box be housed in a low-leakage, lined cell.

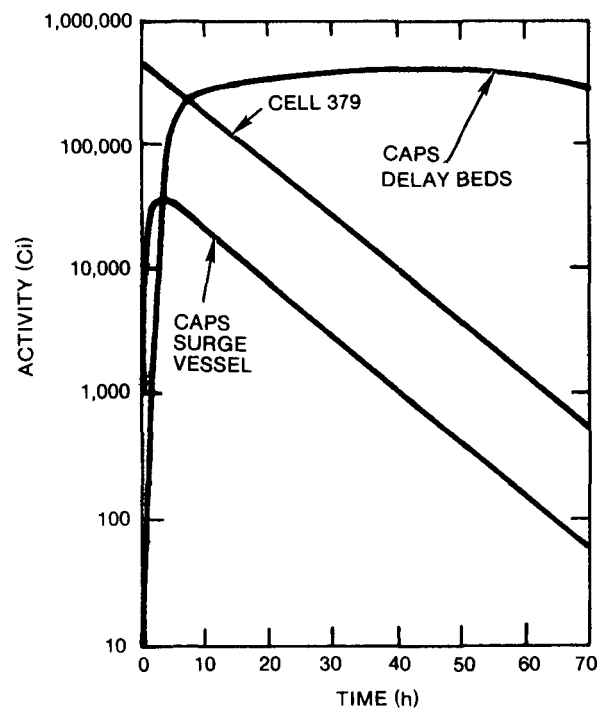


Figure 12-11. Calculated Krypton and Xenon Inventory as a Function of Time Following a Large Postulated Release in Cell 379 Followed by a Purge to CAPS at 15 cfm

- Both releases and evacuation times would be acceptable in case of a major CAPS cold box failure, even if design basis (10 CRF 100) limits are used to evaluate this BDB combined event:

	<u>Offsite (rem in 2 h)</u>		<u>Onsite (hours to limit)*</u>	
	<u>Dose</u>	<u>Limit</u>	<u>Time</u>	<u>Limit</u>
RSB confinement	0.066	150	6.5	30
No RSB confinement	4.6	150	1.5	30

*Evacuation time available. All doses are to the thyroid.

As a result of the above studies, the following two categories of changes were made in CAPS.

- The first category of changes was to extend CAPS capability to include the ability to clean up cells that have experienced a small release of radioactivity and thereby reduce system and plant downtime due to inaccessibility. These changes are generally necessary because the design basis for CAPS originally considered only very low levels of normal radioactivity. With only a modest number of changes, the versatility of CAPS for nonnormal events was improved and doses associated with those events were made as low as reasonably achievable. The use of CAPS for cell cleanup following an accident would increase the level of radioactivity in CAPS temporarily. Most of the items in this category provide the same kind of cell isolation and leak-tightness features for CAPS equipment cells as were provided for cells containing equipment with moderate activity as a matter of good design practice. The cost of all the features in this category totaled several hundred thousand dollars and was judged cost effective.

Several important recommendations were included in this category but have more general relevance:

- CAPS control panels which are useful to mitigate or recover from gaseous release accidents could be made inaccessible by the same accident even though they were not located in the same cell. These control panels were relocated to a location that has a high probability of being habitable in the event of a radiation release in the RCB or RSB.
- A continuing supply of liquid nitrogen to the CAPS cold box (if used for casualty recovery) is important. Interruptions of several hours are acceptable but not several days or longer. (The charcoal bed would approach ambient temperature and release krypton and xenon.) The LN₂ supply pipes were made Seismic Category I to ensure that they would be available following an SSE, and an auxiliary LN₂ fill line was added to permit filling of the dewars from outside the RSB should the RSB become inaccessible following an activity release.
- This category extended the capability of CAPS to recover from large radiation releases as well as small ones. The major changes in this category were: (1) a steel liner for the CAPS cold box cell with a leak-tightness specification was recommended because of the large radioactive inventories and the CAPS cold box could contain following its use for post-accident recovery and (2) procedural changes, including preparation of abnormal operating procedures in advance of their need to assure that all the necessary precautions were observed. To assist in this, the pertinent items from the study were listed in a draft procedure which is summarized in Table 12-2. The cost of the cell lines and other features required for increased leak tightness (e.g., special penetrations) was several hundred thousand dollars. These additions were, strictly speaking, not cost effective. They were made to provide the same

TABLE 12-2
SUMMARY SCENARIO FOR POST-ACCIDENT CLEANUP OF A TYPICAL CELL CONTAINING
HIGH LEVELS OF RADIOACTIVITY

-
- Accident occurs
 - Cell radiation monitors detect high radioactivity in cell atmosphere
 - Alarm sounds in main control room (MCR)
 - Cell isolates automatically
 - Cell slowly leaks to surrounding areas
 - Radioactivity detected in building exhaust
 - Building switches to confinement operating mode
 - MCR operator orders evacuation of plant personnel
 - "Essential personnel" are allowed to remain in building
 - Affected systems may be shut down (e.g., reactor)
 - Operator performs mitigating actions (e.g., if leak is in RAPS cryostill, drain cryostill)
 - Operator determines availability of CAPS and consults SDD 82, Section 4, for operating procedure describing use of CAPS for cleanup of large radioactive releases
 - If CAPS is available (i.e., meeting leak-tightness requirements and operating satisfactorily), operator isolates nonessential input paths to CAPS
 - Operator initiates controlled purge of cell to CAPS
 - After a period of time, CAPS effluent radiation monitors alarm and switch CAPS to the recirculation mode
 - . If this occurs as predicted in SDD 82, Section 4, the operator overrides the recirculation signal since site boundary doses will be low
 - . If this does not occur as predicted, the operator stops purging the cell to CAPS and takes corrective action described in SDD 82, Section 5.
 - Cell is purged to CAPS until there is negligible radioactivity remaining in the cell
 - Nonessential input flows to CAPS are minimized to prolong decay time on CAPS delay beds
-

level of protection for the CAPS cold box cell as for other cells containing large radioactive gas inventories, even though the CAPS cold box would rarely, if ever, contain a large inventory. This extra degree of conservatism added to a demonstration plant would not necessarily be included in a commercial plant.

12.4 Final System Status

12.4.1 Status at Project Termination

The last baselined SDD 82, which included Sections 1 and 2 and Appendices A, B, C, E, F, G, I, Q, V, and X, was Revision 35. Included in this revision were P&IDs for the various subsystems as follows:

- N099821016: Rev. 27 except Sheet 5 - Rev. 26
- N099821017: Rev. 28 except Sheets 2, 3, 4 - Rev. 27
- N099821018: Rev. 21
- N099821034: Rev. 10
- N099821035: Rev. 12
- N099821040: Rev. 25 except Sheet 3 - Rev. 24
- N099821045: Rev. 25
- N099821050: Rev. 18
- N099821051: Sheet 1 - Rev. 22; Sheets 2, 3 - Rev. 21
- N099821052: Rev. 12

- N099821053: Rev. 14
- N099821054: Rev. 18
- N099821055: Rev. 17

The unbaselined sections of the SDD were in the following draft revisions:

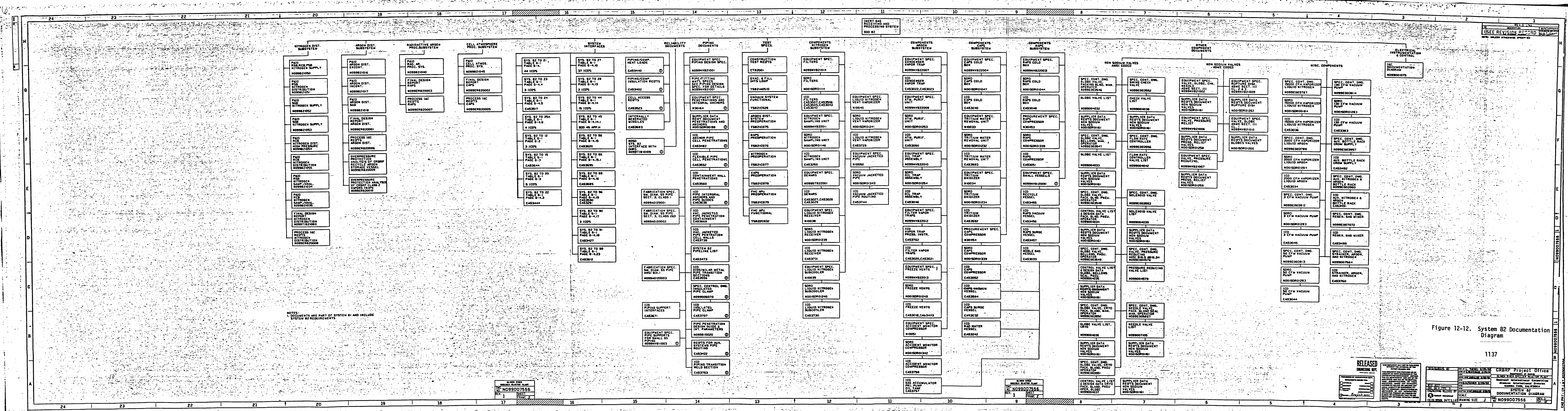
- Appendix H: Maintenance Outline Procedures, D3
- Section 3: System Limitations, Set Prints, and Precautions, D5
- Section 4: Operation, D5
- Section 5: Casualty Events and Recovery Procedures, D3
- Section 6: Maintenance, D4

A complete listing of the applicable documents for System 82 is shown on the System 82 documentation diagrams, Figures 12-12 and 12-13. These documents describe the system design at termination, except for Interface Control Document CA53764, "System 82 to System 85 Interface Requirements," which had not been released at termination.

Design review open items are many and would best be defined in Refs. 20, 26, and 27. All of these open items would have been satisfied if the unapproved ECPs listed in Section 12.2.4 had been implemented.

12.4.2 Outstanding Approved Changes

At the termination, there were 12 approved ECPs that had not been completely incorporated into project documents. They are as follows:



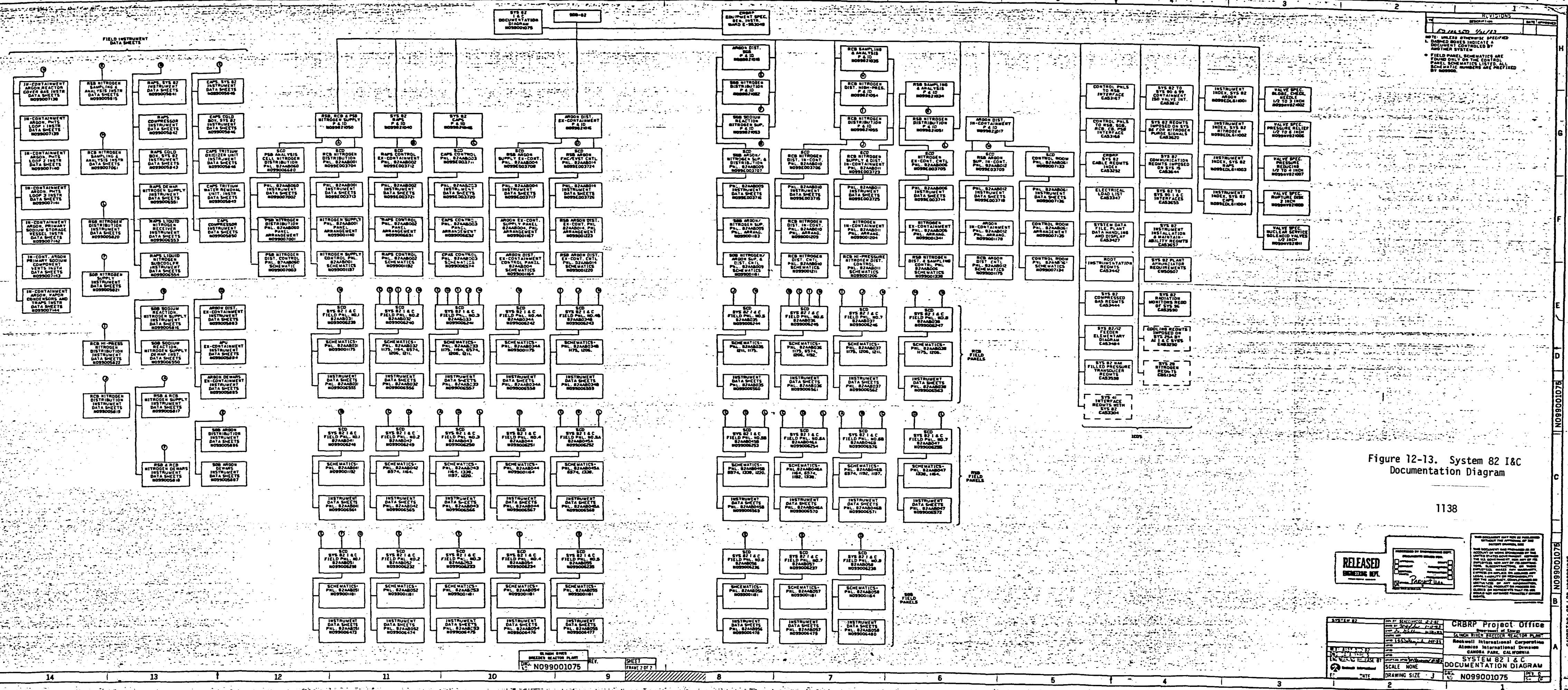


Figure 12-13. System 82 I&C Documentation Diagram

RELEASED
ENGINEERING DEPT.

SYSTEM 82		CRBRP Project Office	
109900175		109900175	
109900176		109900176	
109900177		109900177	
109900178		109900178	
109900179		109900179	
109900180		109900180	
109900181		109900181	
109900182		109900182	
109900183		109900183	
109900184		109900184	
109900185		109900185	
109900186		109900186	
109900187		109900187	
109900188		109900188	
109900189		109900189	
109900190		109900190	
109900191		109900191	
109900192		109900192	
109900193		109900193	
109900194		109900194	
109900195		109900195	
109900196		109900196	
109900197		109900197	
109900198		109900198	
109900199		109900199	
109900200		109900200	

- A82293 - Deletion of CAPS Tritium Oxidizer
- A82311 - System 96 Interface Revision
- A82330 - RAPS and CAPS Radioactivity Inventory Calculations
- A82339 - Thermocouple Deletion on System 82 P&IDs
- A82354 - Hold Removal from E-Spec N099NV812009
- A82363 - Vendor Clarification and Revision of Valve Data Sheets
- A82368 - Update E-Spec N10038
- A41515 - HEPA Filter Testing Provisions
- A81383 - System 81 to 82 Interface
- W1765 - Revision of System 90 Plant Annunciator System ICDs
- G1208 - (reopened) System 51B--IHX Intermediate Side Vent
- G1437 - IHTS Required Sampling Frequency

These approved ECPs should be included in the system design to provide the latest approved design.

12.4.3 Outstanding Activities to Complete

12.4.3.1 SDD 82

In order to update SDD 82, Sections 1 and 2 and the baselined appendices, the following approved ECPs should be incorporated:

- A41515 - HEPA Filter Testing Provisions
- A82293 - Deletion of CAPS Tritium Oxidizer
- A82311 - System 96 Interface Revision
- A82339 - Thermocouple Deletion on System 82 P&IDs
- W1765 - Revision of System 90 Plant Annunciator System
- G1437 - IHTS Required Sampling Frequency

Outstanding open items against SDD 82 are changes proposed in unapproved ECPs to downgrade CAPS to non-safety-class, Seismic II; to revise SDD 82, Sections 1

and 2 and appendices, in response to the two design reviews discussed in Section 12.2.4; to add bridge marks to valves on all the P&IDs; and to revise the argon purge lines to provide safer impurity control.

Sections 3 through 6 and Appendix H have not been baselined and should include approved ECPs A81383, A82330, and G1437 before they are baselined. The open items outstanding against SDD 82 Sections 1 and 2 would also impact Sections 3 through 6 and Appendix H and should be considered open items against these documents.

12.4.3.2 Other Documents

Other system documentation such as Interface Control Documents, Specifications, and I&C drawings still have approved ECPs that have not been included. These ECPs are:

- A41515 - HEPA Filter Testing Provisions
- A82231 - CAPS/FFMS Containment Isolation
- A82293 - Deletion of CAPS Tritium Oxidizer
- A82310 - System 82 Piping Design Spec Update
- A82311 - System 96 Interface Revision
- A82319 - In-Line Instrumentation Code Requirements
- A82330 - RAPS and CAPS Radioactivity Inventory Calculations
- A82334 - Field Panel Corrections
- A82339 - Thermocouple Deletion on System 82 P&IDs
- A82354 - Hold Removal from E-Spec N099NV812009
- A82363 - Vendor Clarification and Revision of Valve Data Sheets
- A82368 - Update E-Spec N10038
- G1208 - System 51B--IHX Intermediate Side Vent
- G1437 - IHTS Required Sampling Frequency

and should be included in the pertinent documents to update these documents to the terminal baselined revision. As stated in Section 12.4.1, ICD CA53764 has

not been released even though it has been submitted for approval. Specification control drawings have not been prepared for non-safety-class valves of all sorts. Instrument data sheets and many other I&C documents had not been prepared at the termination.

Construction and Preoperational Test Requirement documents are available in draft form.

Installation Requirements Documents have not been written. Since no System 82 components have been fabricated, these documents will not be part of the IGRP system documentation.

There are no Operations and Maintenance Manuals available, since they were to be furnished by component vendors and no components have been fabricated.

Much of the piping effort on System 82 by Burns and Roe has been released, but there were still many drawings that were in preparation at termination.

12.4.3.3 Miscellaneous Unapproved Design Changes

During the consolidation meetings reported in Ref. 30, a number of potential system changes to upgrade System 82 were discussed and agreed to as being necessary. A brief discussion of the changes follows.

To assist the A-E, it was proposed to add line numbers on P&IDs for the outlet piping from pressure relief devices and to add dimension tolerances for component interfaces on Interface Control Drawings. The SWRPRS piping was to be revised to provide necessary gas services. A number of I&C changes were proposed. Specifications and Interface Control Drawings would be revised to reflect the current baselined design. Design pressure differences between E-Specifications and the System 82 Pipeline List were to be rectified and reissued. The HOLD on the location of attachment points for supports on pneumatic valves was to be resolved after information was provided from the valve vendor.

In total, there were about 40 pending changes that affected SDD 82 and other System 82 documents.

12.4.4 Component Status

The final design of the small vessels in RAPS and CAPS, the controlled temperature profile condenser, and the filter vapor traps had been completed but no components had been fabricated. Other components were at various stages of design.

12.4.5 Outstanding Problems

At the time of termination, there were a number of proposed changes being coordinated with the several design organizations. Of these, the following changes would have had major impact on the system design and/or system component design.

ECP W-1733 would have added continuous gas monitoring instruments for nitrogen and hydrogen to the reactor cover gas system outlet piping to provide rapid detection of impurities that could affect material properties.

ECP W-2143 would have added emergency vents to the PHTS cover gas system to prevent excess loss of sodium from the reactor under certain accident conditions.

ECP G-1436 would upgrade the ASME Code Class of the PHTS cover gas supply loops and would add active valves to provide boundary isolation on the initiation of containment isolation. Additional changes were proposed to reduce the impact of IHTS/PHTS sodium leakage.

ECP G-1208 was reopened to require shrouded freeze vents on the IHX IHTS sodium inlet line vent. The need for such a device had not been resolved at the time of termination.

ECP A81398, which was to implement ECP L10-146, would require sodium spill analysis to determine whether pipe supports, component embedments, and/or components would be affected by the presence of sodium pools in inerted cells.

If the current CRBRP plant design is used to provide input to future breeder reactor designs, the logic for implementing these proposed changes should be considered.

12.4.6 Recommendations for Future Designs

The use of argon as the cover gas in the reactor presented problems in the design of the RAPS cryostill since the low operating temperature causes the condensation of argon and impurities such as krypton and xenon (which the cryostill was designed to remove) and oxygen (which is a potential hazard in the cryostill). Additionally, the cryostill does not remove argon radioisotopes, which are produced by nuclear reactions in the reactor from the cover gas and are therefore present in the recycle argon. This cover gas activity placed a limitation on the use of the recycle argon and an increase in the requirement for fresh argon. The use of helium as a reactor cover gas could allow the design of purification systems operating below the condensation temperature of oxygen, thus eliminating this problem. Since helium does not produce any radioisotopes, the recycle helium would be suitable wherever cover gas is needed.

The present concept of the IGRP system includes over 2000 valves in its design. The maintenance and surveillance of these valves require many man-hours, which has resulted in radiation doses to plant operating personnel that should be reduced. The piping layout and design should be reviewed with the intent of simplifying the design and reducing the number of valves or of locating valves in radiation shielded valve galleries.

From the inception of design, the IGRP system should be designed with redundant components for critical subsystems, with shielded components such as

vapor traps for ease in replacement, and with the location of redundant components in separate cells for ease in repair and/or replacement.

If the PHTS loops and reactor are designed with pressure equalization, the piping connecting these components should be supported by trace heated, insulated pipe clamps, and wall penetrations should be trace heated and insulated to eliminate plugging of these lines by condensed sodium.

Any cover gas piping containing radioactive reactor cover gas, the purification components for decontaminating the cover gas, and failed fuel monitoring systems using the reactor cover gas should be located in containment to reduce the presence of potential radioactive sources that could leak to the environment.

12.5 References

1. SDD 82, Rev. 35, "System Design Description - Inert Gas Receiving and Processing System," October 7, 1983
2. WARD-D-0244, Rev. 3, "CRBRP Preservice and Inservice Inspection Requirements for Sodium and Cover Gas Systems," January 7, 1983
3. WARD-D-0177, Rev. 14, "CRBRP ALARA Criteria and Guidelines," July 14, 1983
4. OPDD-10, Rev. 146, "Overall Plant Design Description," August 1, 1983
5. RDT Standard A 1-5T, "Purity Requirements for Operating Sodium Reactor Systems," March 1976
6. RDT Standard M 14-1T, "Sodium Cover Gas Purchase Specification," January 1975

7. N099ER820001, "Process Analysis and Equipment Sizing for the Argon Storage and Distribution Subsystem of System 82 - Final Design Report," March 1, 1983
8. N099ER820006, "Process Instrumentation and Control Requirements for Argon Storage and Distribution Subsystem of System 82," February 11, 1983
9. N099TI821083, "Instrumentation and Controls for the Inert Gas Receiving and Processing System - Design Report," February 7, 1983
10. N099TI821080, "Process Analysis and Equipment Sizing for Nitrogen Distribution Subsystem of System 82 - Final Design Report," March 1, 1983
11. N099ER820008, "Process Instrumentation and Control Requirements for the Nitrogen Storage and Distribution Subsystem of System 82," March 28, 1983
12. WARD-D-0193, "Radioactive Source Terms," March 30, 1983
13. N099ER820003, "Radioactive Argon Processing Subsystem (RAPS) - Final Design Report," March 1, 1983
14. N099ER820007, "Process Instrumentation and Control Requirements for the Radioactive Argon Processing Subsystem (RAPS) of System 82," March 8, 1983
15. N099ER820002, "Cell Atmosphere Processing Subsystem (CAPS) - Final Design Report," March 1, 1983
16. N099ER820005, "Process Instrumentation and Controls for the Cell Atmosphere Processing Subsystem," February 28, 1983
17. N099DRR821001, "Conceptual Design Review (Formal), System 82, Inert Gas Receiving and Processing," April 4, 1976
18. N099TI820023, "System 82 Responses to FFTF Experiences," March 18, 1983

19. N099DRR820005, "Preliminary Design Review, Argon and RAPS, System 82," March 29, 1979
20. N099DRR821002, "Preliminary Design Review, Nitrogen and CAPS, System 82/61," July 5, 1978
21. N099TI820022, "System 82 Responses to Key Systems Task Force Recommendations," March 14, 1983
22. N099DRR820010, "System 82 Final System Design Review," April 6, 1983
23. Letter 83ESG-1068, "CRBRP; System 82 FWSDR Review Package," March 2, 1983
24. N099TI820020, "System 82 Requirements and Verification List - SDD 82, Section 1," March 2, 1983
25. Letter LA830308, "CRBRP; Project Office Comments on System 82 FWSDR Package," September 9, 1983
26. Letter LA830301, "CRBRP; Additional Westinghouse Comments on System 82 FWSDR Package," September 7, 1983
27. Letter LA830108, "CRBRP; Westinghouse Comments on System 82 FWSDR Review Package," March 31, 1983
28. Letter 83ESG-5860, "AI-ESG Response to System 82's FWSDR Comments," September 28, 1983
29. Letter 83ESG-6150, "AI-ESG Responses to Comments on System 82 FWSDR Package," October 13, 1983
30. Letter LA830318, "CRBRP; Transfer of System Engineering Responsibility for Systems 81/63 and 82/61," September 19, 1983

13.0 IMPURITY MONITORING AND ANALYSIS SYSTEM - SYSTEM 85

13.1 Functions and Description

13.1.1 Functions

The functions of the impurity monitoring and analysis system (IMAS), also designated as System 85 in the Clinch River Breeder Reactor Plant (CRBRP), are defined as follows:

- Provide for monitoring of sodium impurity levels and determination of plugging ("saturation") temperature of all sodium subsystems.
- Obtain and identify sodium samples from the primary heat transport system (PHTS), ex-vessel storage tank (EVST), and intermediate heat transport system (IHTS) for chemical and radiochemical analysis, or for storage for archive purposes.
- Expose wires, foils, or tabs of appropriate materials to flowing sodium to obtain equilibration specimens of the following sodium impurities: oxygen, hydrogen, and carbon.
- Obtain and identify samples of primary cover gas, EVST cover gas, fuel handling cell (FHC) cover gas, and IHTS cover gas for chemical analysis.
- Obtain, identify, and analyze samples of incoming nitrogen and argon.

These functions are provided by the following subsystems: (1) PHTS Sodium Characterization Subsystem, (2) EVST Sodium Characterization Subsystem, (3) IHTS Sodium Characterization, (4) Primary Cover Gas Sampling and Monitoring Subsystem, (5) EVST, FHC, and IHTS Cover Gas Sampling and Monitoring Subsystem, (6) Incoming Argon and Nitrogen Acceptance Sampling Subsystem, and (7) Plant Service Building (PSB) Combined Laboratory.

13.1.2 Descriptions

13.1.2.1 System Features

Sodium impurities to be monitored and analyzed by System 85 in support of System 31 (reactor system), 41 (reactor refueling system), 51A (primary heat transport system), 51B (intermediate heat transport system), and 81 (auxiliary liquid metal system) are those required by Reactor Development and Technology (RDT) Standard A 1-5T (Ref. 1) and include concentrations of the interstitials (oxygen, hydrogen, and carbon), trace metals, fission products, and fuel materials. Plugging temperature indicators (PTI) and multipurpose samplers (MPS) are provided for on-line sodium monitoring and sampling.

Argon impurities to be monitored and analyzed by System 85 in support of System 31 (reactor system), 41 (reactor refueling system), 51A (primary heat transport system), 51B (intermediate heat transport system), 53 (steam generator system), and 82 (inert gas receiving and processing system) are those required by RDT Standard A 1-5T and include helium, hydrogen, oxygen, nitrogen, methane, and carbon monoxide. Incoming cover gas (argon and nitrogen) impurities to be monitored and analyzed by System 85 are oxygen, hydrogen, nitrogen, carbon monoxide, carbon dioxide, hydrocarbon gases (methane), and water.

A complete description of System 85 is contained in Reference 2. The overall system is designed and arranged to meet the inservice inspection (ISI) requirements of Westinghouse Advance Reactor Division (WARD)-D-0244, CRBRP Preservice and Inservice Inspection Requirements for Sodium and Cover Gas Systems (Ref. 3) and the ALARA* criteria of WARD-D-0177, CRBRP ALARA Criteria and Guidelines (Ref. 4).

*As Low As Reasonably Achievable

13.1.2.2 Safety and ASME Code Classification

The final safety and American Society of Mechanical Engineers (ASME) code classifications of System 85 components are listed in Table 13-1. The safety classes are defined in Section 1.3.1.6 of Overall Plant Design Description (OPDD-10) (Ref. 5).

TABLE 13-1
COMPONENT SAFETY AND ASME CLASSIFICATION, SYSTEM 85

Component	Safety Class	Code Class
PHTS Sodium Characterization Subsystem:		
Piping to System 81	1	III-1
Isolation valve	1	III-3
Other piping, valves, PTI, and sodium sampling packages (SSP)	3	III-3
EVST Sodium Characterization Subsystem:		
Sodium sampling loop piping, valves, PTI, and SSP packages	3	III-3
IHTS Sodium Characterization Subsystem:		
Sodium sampling loops, piping, valves, and intermediate sodium characterization packages	3	III-3
All other components	None	-

13.1.2.3 PHTS Sodium Characterization Subsystem

The PHTS Sodium Characterization Subsystem consists of a sodium sampling loop which provides a sodium sample stream to the primary plugging temperature indicator package (primary PTI) (Figures 13-1 and 13-2) and the primary sodium

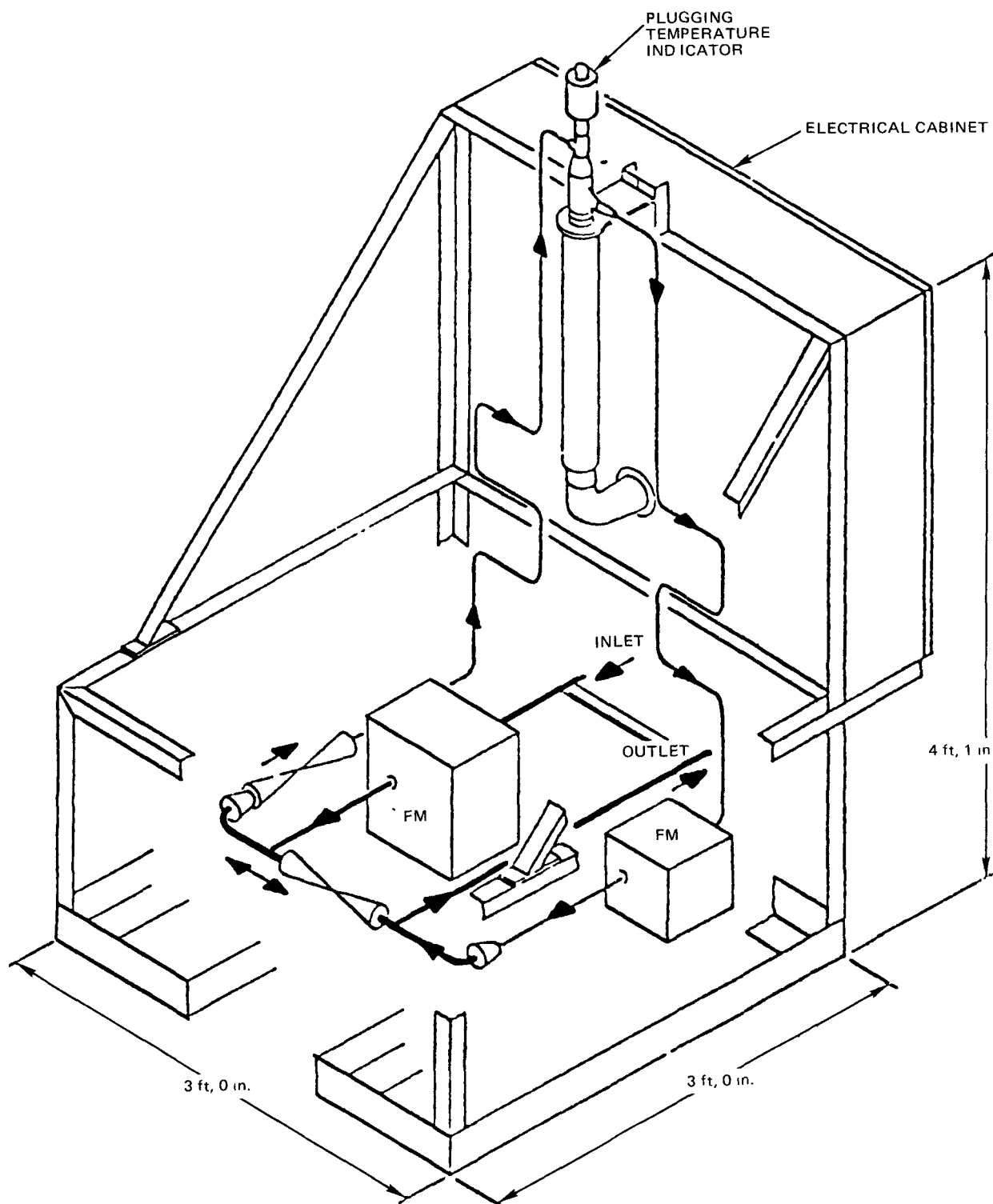


Figure 13-1. Primary Plugging Temperature Indicator Package

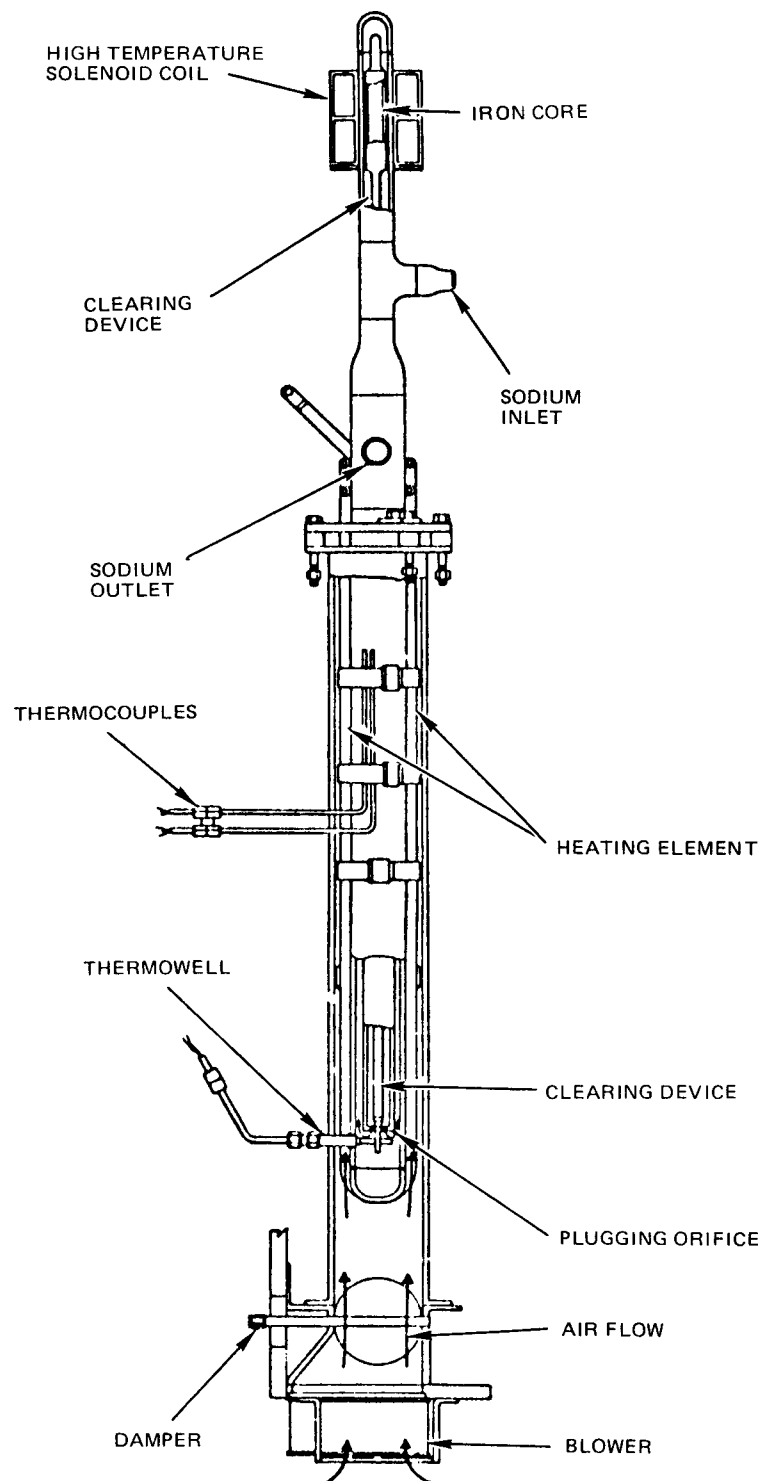


Figure 13-2. Plugging Temperature Indicator

sampling package (primary SSP) (Figures 13-3 and 13-4) for simultaneous monitoring and sampling, if needed. The primary PTI contains an internal distribution loop (which delivers a continuous sodium sample to a PTI assembly), a flowmeter, a flow control valve, and a freeze vent. The primary SSP contains an internal distribution loop which delivers a continuous sodium sample to two multipurpose sampler (MPS) devices, connected in parallel across the supply and return legs of the loop. Flowmeters and flow control valves are provided for the distribution loop and each MPS.

A detailed description of the PHTS Sodium Characterization Subsystem and its performance characteristics and the associated instrumentation can be found in Sections 2.2.1.1 and 2.5.2.1 of System Design Description (SDD) 85. Detailed descriptions of the components in this subsystem can be found in Section 2.4 of the SDD. The final design reports which include this subsystem are Refs. 6 and 7.

13.1.2.4 EVST Sodium Characterization Subsystem

The EVST Sodium Characterization Subsystem consists of a sodium sampling loop which provides a sodium sample stream to the ex-vessel plugging temperature indicator package (ex-vessel PTI) and the ex-vessel sodium sampling package (ex-vessel SSP) in parallel. The ex-vessel PTI contains an internal distribution loop (which delivers a continuous sodium sample to a PTI assembly), a flowmeter, a flow control valve, and a freeze vent similar to the primary PTI. The ex-vessel SSP (Figure 13-5) contains an internal distribution loop which delivers a continuous sodium sample to an MPS device, connected across the supply and return legs of the loop. Flowmeters and flow control valves are provided for the loop and the MPS.

13.1.2.5 IHTS Sodium Characterization Subsystem

The IHTS Sodium Characterization Subsystem consists of three sodium sampling loops, one from each intermediate heat transfer system (IHTS) loop, which provide sodium sample streams to three intermediate sodium characterization

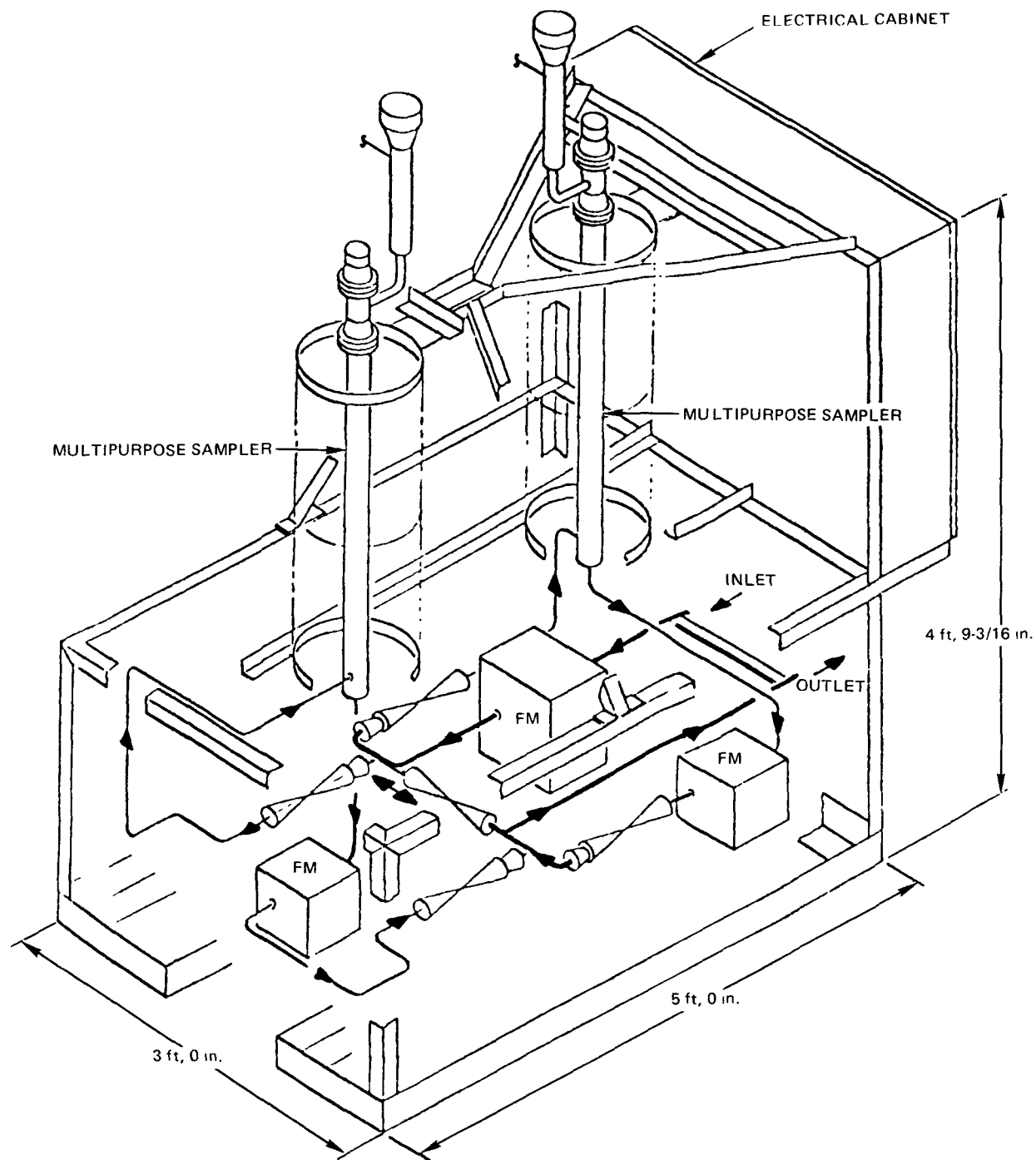


Figure 13-3. Primary Sodium Sampling Package

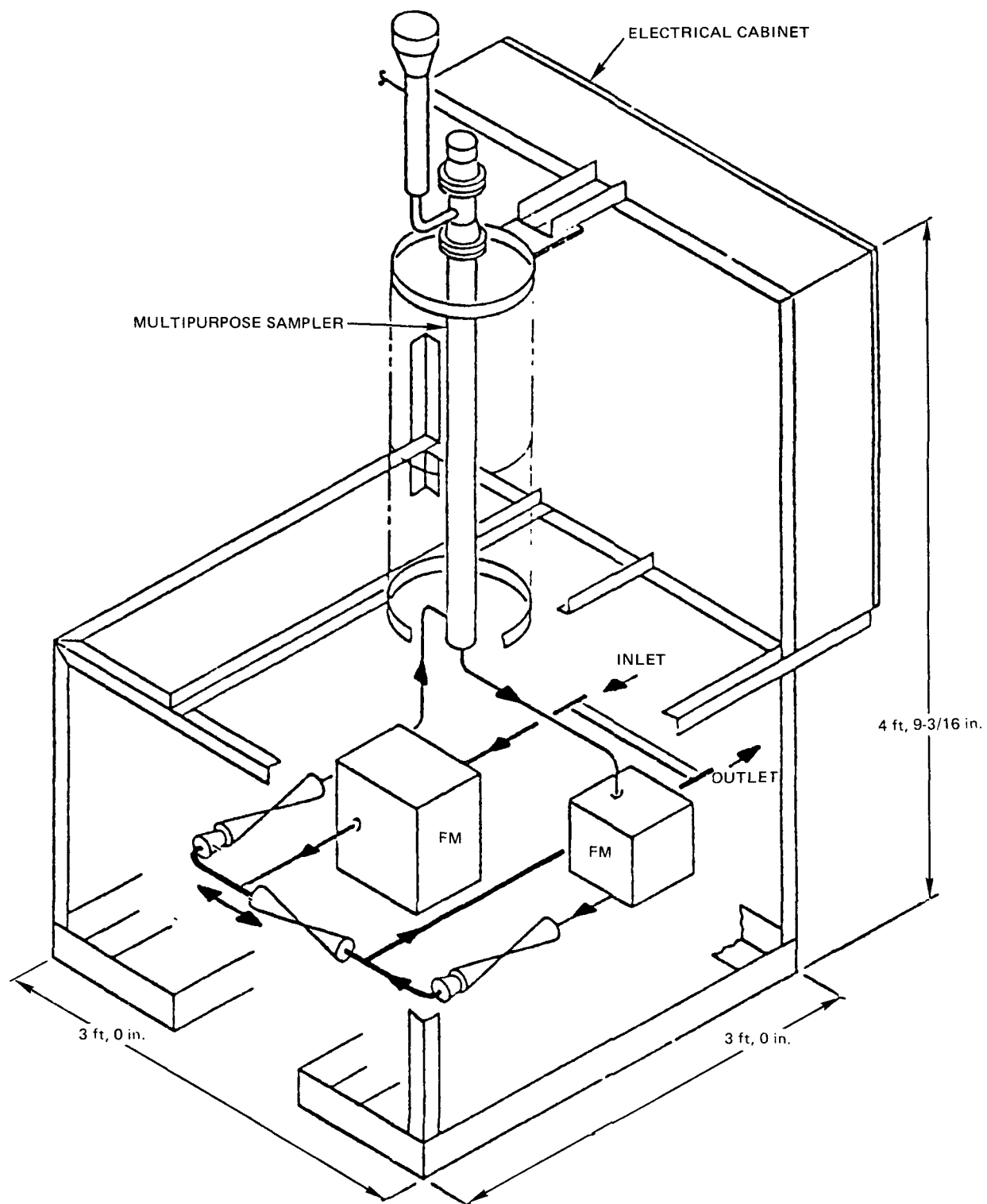


Figure 13-4. Multipurpose Sampler

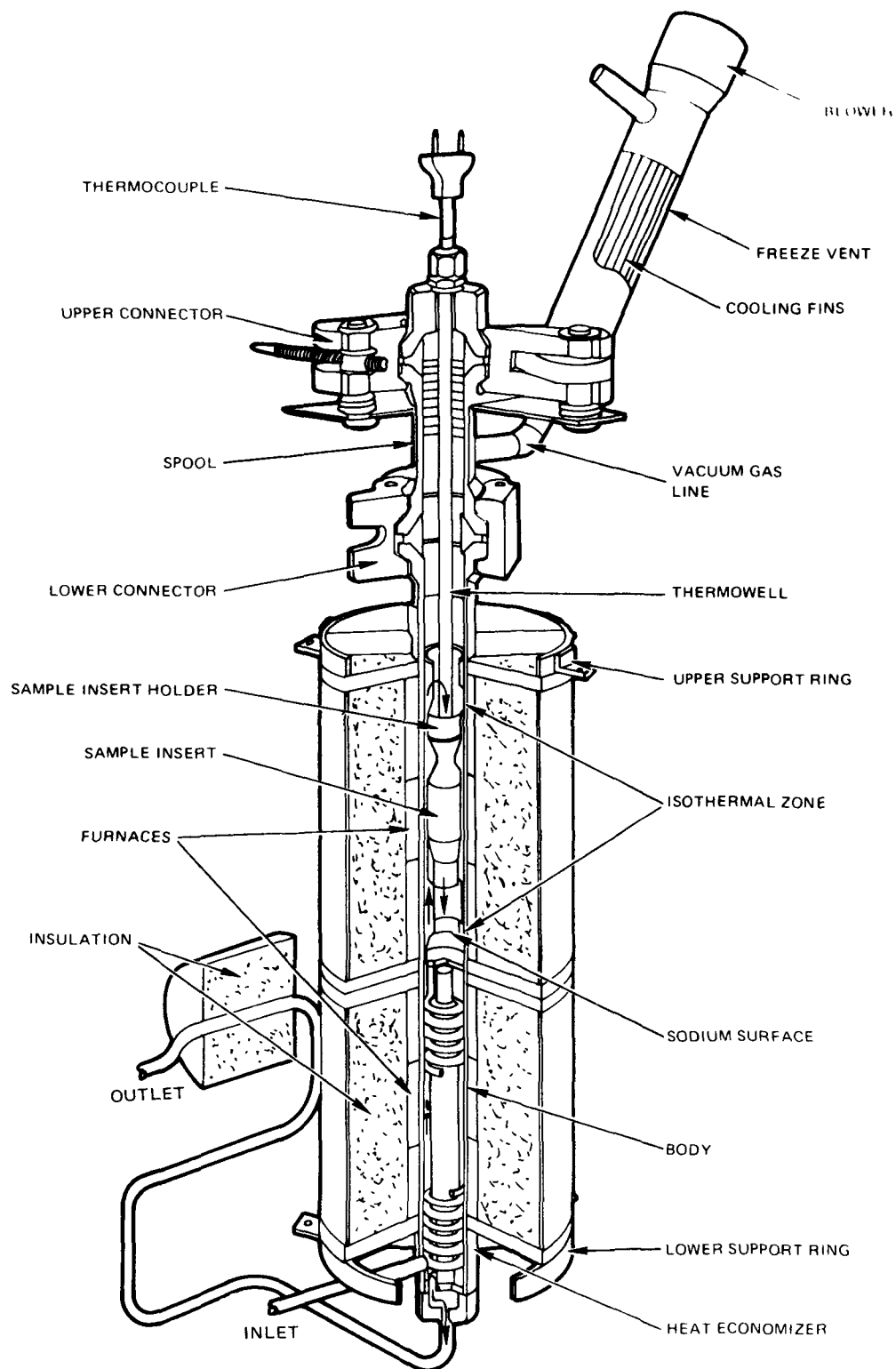


Figure 13-5. Ex-Vessel Sodium Sampling Package

packages. Each intermediate sodium characterization package (Figure 13-6) contains an internal distribution loop which delivers a continuous sodium sample to a PTI assembly and an MPS device, connected in parallel across the supply and return legs. The loops contains a flowmeter and a flow control valve. Isolation valves and a flowmeter are provided for the MPS.

13.1.2.6 Primary Cover Gas Sampling and Monitoring Subsystem

The fuel failure monitoring system (System 94) provides a grab sample of primary cover gas in a shielded bottle for analysis (gas chromatograph) in the PSB combined laboratory. Other samples are taken by System 94 and analyzed by the maintenance system (System 45) for impurities, such as tritium, not detectable by the gas chromatograph.

A description of the component for this subsystem is provided in System 94 Equipment Specification N10036 (Ref. 8).

13.1.2.7 EVST, FHC, and IHTS Cover Gas Sampling and Monitoring Subsystem

Sample connections are provided on the inert gas receiving and processing system (System 82) piping for sampling cover gas in the EVST, FHC, and IHTS. Provisions for taking these samples are provided via gas sample bottles which are then transferred to the PSB combined laboratory for gas chromatograph analysis.

13.1.2.8 Incoming Argon and Nitrogen Acceptance Sampling Subsystems

A sample from each lot of incoming liquid argon and each lot of incoming liquid nitrogen will be taken using a cryogenic sampler provided by System 45. The samples will be analyzed in the PSB combined laboratory for oxygen, water, carbonaceous gases, and other impurities to establish conformance to specification RDT Standard M 14-1T, (Ref. 9) before using the lot in the plant. Isolation valves and hose will be provided with the sampler.

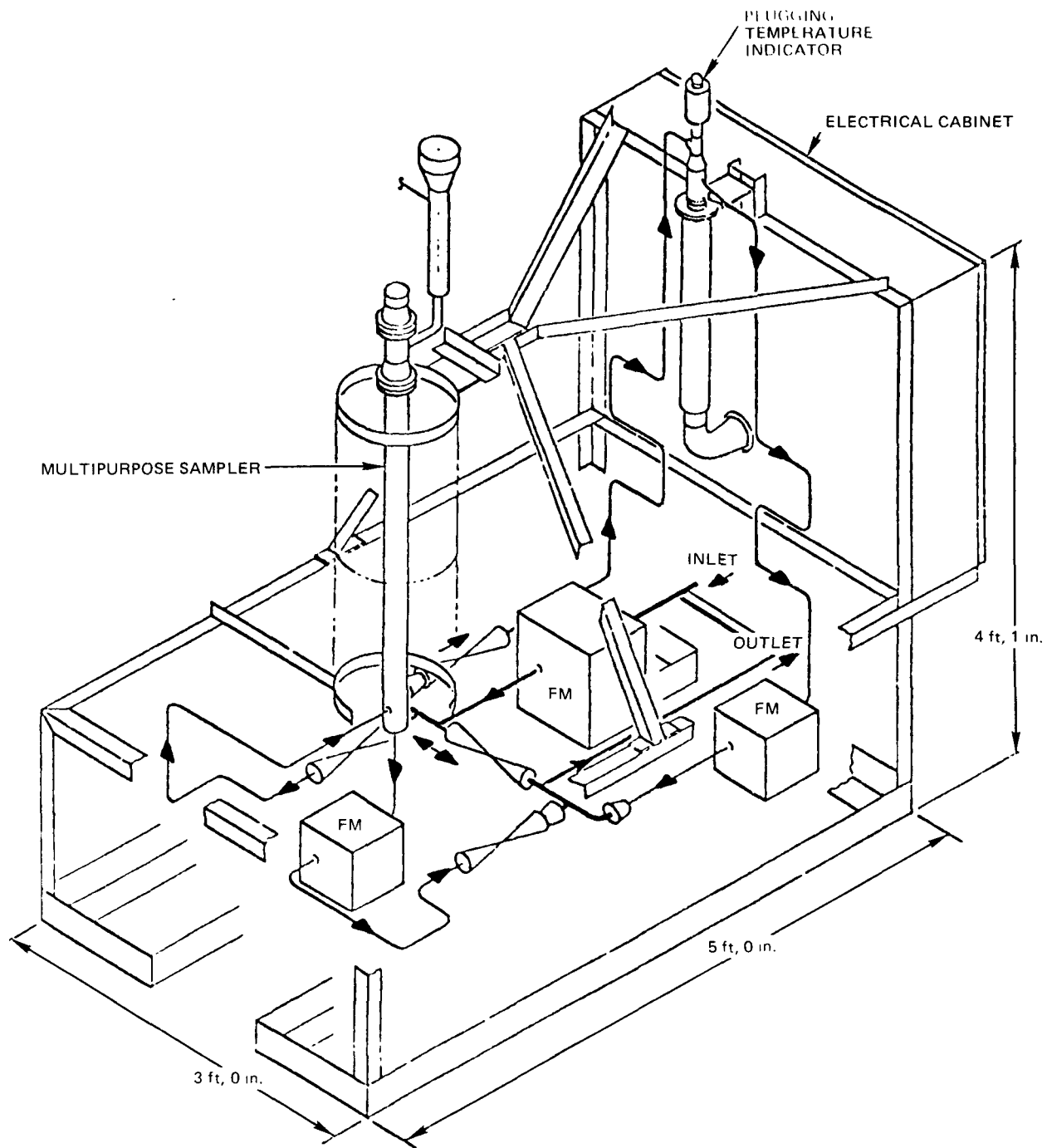


Figure 13-6. Intermediate Sodium Characterization Package

13.1.2.9 PSB Combined Laboratory

Systems 85 and 45 will provide to the PSB combined laboratory the equipment required for performing the RDT Standard F 3-40T (Ref. 10) analytical procedures identified. These analyses determine the sodium and cover gas impurities as required by RDT Standard A 1-5T. Counting equipment required for these procedures, such as alpha counter, alpha spectrometer, multichannel analyzer, and liquid scintillation counter, will be provided by System 45.

13.2 Evolution of System 85 Design

The design of the impurity monitoring and sampling system was changed throughout the design phase because of design evolution, design improvement, FFTF experiences, cost reduction, improved safety, changing requirements from interfacing systems, and changing requirements in OPDD-10. Only major design changes will be described in the following sections.

13.2.1 Conceptual Design

At the start of the conceptual design in 1973, the basis for the CRBRP design was the then current design for Fast Flux Test Facility (FFTF). The reference design was modified to correspond to the CRBRP requirements, but the CRBRP design initially consisted of (1) the PHTS sodium characterization subsystem, (2) the EVST sodium characterization subsystem, (3) a single IHTS sodium characterization subsystem arranged to sample selectively each of the three PHTS loops, (4) the incoming NaK sampling package, (5) the incoming sodium drum sampling package, (5) the continuous on-line primary cover gas sampling and analysis subsystem, and (6) the EVST and IHTS cover gas sampling subsystem.

The Conceptual Design Review for System 85 was held on July 17, 1974, for the process portion of the impurity monitoring and analysis system. The review was based on SDD 85, Sections 1 and 2, Draft 1, Revision 0, which had been transmitted for project approval. The comments from the Design Review (Ref. 11) were incorporated in a revised draft of SDD 85, Revision 0.

During the conceptual design phase, the instrumentation for the IMAS was defined in SDD 62, Impurity Monitoring and Analysis Instrumentation System; therefore, a separate Conceptual Design Review was held for System 62 on September 18, 1974, based on Draft 1 of SDD 62, Revision 0, Sections 1 and 2. The action items from the Design Review (Ref. 12) were incorporated in the next drafts of SDD 85 and SDD 62.

13.2.2 Preliminary Design

Following the Conceptual Design Reviews for Systems 85 and 62, the design descriptions, Sections 1 and 2, of SDD 62 were folded into SDD 85 so that a single document was available for the IMAS. During the preliminary design phase, SDD 85, Revision 0 was issued in modified draft versions for approval by the Project Office. These draft versions were in response to interface requirement changes, to changing OPDD-10 requirements, and to the comments and action items from the Conceptual Design Reviews.

During the preliminary design, draft versions of subsystem flow diagrams and P&I diagrams were prepared in response to system changes.

13.2.2.1 Primary Sodium Characterization Subsystem

The Primary Sodium Sampling System at the end of preliminary design consisted of the sodium sampling loop from the outlet of the System 81 primary sodium makeup pumps and return to the overflow vessel. The loop provided sodium to the primary plugging temperature indicator and to the primary sodium sampling package which contains two multipurpose samplers (MPS) to provide redundant sampling capability. The sodium sampling loop also provided sodium to and from capped connections in a different cell which was to be used for an auxiliary PTI or an on-line oxygen/hydrogen meter, the selection to be made at a later time. The use of a forced-convection cooled freeze vent on the PTI was on HOLD at this point in the preliminary design. This subsystem was designed for remote operation due to the radioactive sodium in the piping.

The primary sodium sampling package was installed in a cell with manipulators and a shielding window to reduce the exposure of the operator.

13.2.2.2 EVST Sodium Characterization Subsystem

The EVST Sodium Sampling System consisted of a sample loop to and from either the EVST or the fuel handling cell. This loop provided sodium to the ex-vessel PTI without a forced-convection cooled freeze vent and to the ex-vessel sodium sampling package with one MPS. At the present condition of preliminary design, the ex-vessel sodium sampling package was on HOLD. This system was designed for reach-rod operation of valves and remote readout due to the presence of radioactive sodium. A similar cell, as for the primary sodium sampling package, was designed for the ex-vessel sodium sampling package.

13.2.2.3 IHTS Sodium Characterization Subsystem

The IHTS Sodium Sampling System consisted of a single sodium sampling loop which connects to each of the System 81 purification loops for each IHTS sodium loop. Selection of the loop to be sampled was to be performed under administrative control of locked-open or locked-close isolation valves on the System 85 piping connected to the purification loops. The sodium sampling loop provided sodium to the intermediate sodium characterization package which contains a PTI and an MPS. Since the sodium in this system is not radioactive, the subsystem was designed for hands-on operation.

13.2.2.4 Incoming Sodium and NaK Sampling and Monitoring Subsystem

While a requirement for this service was identified, the design of the sampling system was on HOLD.

13.2.2.5 Primary Cover Gas Sampling Subsystem

A reactor cover gas flow of 1 scfm was provided to System 85 by System 82. This stream was compressed by two redundant compressors; a portion was transferred to System 94 components and the balance to the System 85 on-line gas chromatograph. The chromatograph was controlled by a sequencer to perform its analysis function. Any reactor cover gas not used in the chromatograph was returned to the Radioactive Argon Process System (RAPS) inlet. All gas vented from the chromatograph was fed to a CAPS header for processing before being vented to the atmosphere.

13.2.2.6 EVST, FHC, and IHTS Cover Gas Sampling and Monitoring Subsystem

Sampling connections were provided by System 82 on the bleed system from the EVST, FHC, and IHTS cover gas spaces. System 85 would provide a gas sample bottle for the sample.

13.2.2.7 Component Design

Following the Conceptual Design Review, Hanford Engineering Development Laboratory (HEDL) began to design the CRBRP plugging temperature indicator packages and the sodium sampling packages based on their earlier designs of the comparable FFTF components.

13.2.2.8 Preliminary Design Review

In accordance with the requirements of the project, a preliminary design review was held on August 12, 1975. The review was based on SDD 85, Revision 0, which had been submitted for approval to the Project Office and Engineering Change Proposal (ECP) A8501001.

At the same meeting, the Conceptual Design Review of the HEDL-designed plugging temperature indicators, sodium sampling package, and instrument panels was held.

In response to the comments in the Design Review Report (Ref. 13), the system design was revised, and the preliminary design was approved. The comments from the Conceptual Design Review were incorporated in the design of the hardware, and the design was allowed to continue.

13.2.2.9 Component Preliminary Design Review

Following approval of the conceptual design of the sodium sampling packages and plugging temperature indicators, HEDL continued the design effort using FFTF designs which had been fabricated and tested. During this effort, HEDL redesigned the sodium distribution loops to reduce fabrication costs and to relieve the crowding of components in the FFTF design which simplified the access for maintenance.

A Preliminary Design Review was held on October 19, 1976, at which it was determined that too many questions were raised for the design to be approved as reported in Ref. 14. A second design review was held on January 28, 1977. Following this review, the comments and action items in Ref. 15 were incorporated in the pertinent documents and the preliminary design was approved.

13.2.3 Final Design

There were many changes to the SDD 85 which are minor in nature, such as revision of piping, valves, and instruments and their part numbers, and identification of all wall penetration locations and part numbers.

A series of changes were made which applied to all the sodium sampling systems. The argon/vacuum valve cabinets for each sodium sampling package were redesigned as panels to be mounted on the frame of the sodium sampling package. Details of the sodium transfer tunnels for the sodium sampling package cells and the required gas connections were provided for the primary and EVST sodium sampling packages. The design pressure for the sodium sampling packages was increased. Freeze seal lengths were specified for the inlet and outlet piping to and from the PTI and SSP modules to provide additional isolation

of these components for maintenance. Additionally, a number of changes were made to incorporate FFTF experiences and recommendations from their usage of similar devices.

To improve plant safety, an analysis was performed to determine the safest position for system pneumatic valves to be after a failure of these valves. This information was added to the P&IDs.

13.2.3.1 Primary Sodium Characterization Subsystem

Changes specific to the Primary Sodium Characterization Subsystem include revisions to the piping; automatic isolation valves were used between System 81 and System 85 to isolate System 85 in case of sodium leakage in System 85 components; details of the sodium transfer cask transporter were provided; and the transfer cask hoist and support at the transfer tunnel were detailed in pertinent documentation. The cell proposed for the auxiliary PTI was transferred to System 82, and the proposed PTI or oxygen/hydrogen on-line meter was deleted from the system design.

13.2.3.2 EVST Sodium Characterization Subsystem

The major changes to the design of this subsystem were the decision to use a sodium sampling package for EVST sodium impurity monitoring, thereby removing the HOLD, and to replace the sampling loop isolation valves with System 81 valves.

13.2.3.3 IHTS Sodium Characterization Subsystem

The existing design was modified to provide three sodium characterization packages for independent and/or simultaneous sampling, one for each System 81 purification loop, to eliminate cross connections between IHTS loops in the steam generator building.

13.2.3.4 Incoming Sodium and NaK Sampling Subsystem

This subsystem was eliminated from the IMAS design since System 81 deleted this requirement from SDD 81.

13.2.3.5 Primary NaK and EVST NaK Characterization Subsystem

These subsystems were eliminated from the design to satisfy new System 81 requirements. However, a concept is provided in SDD 85 for NaK sampling from the primary and EVS cold trap NaK storage vessels.

13.2.3.6 Primary Cover Gas Sampling and Monitoring Subsystem

A study was made of the use of an on-line gas chromatograph to monitor impurities in the reactor cover gas. The FFTF experience had been that the instrument lost sensitivity due to electrical noise, distance between sample inlet and chromatograph, and had a proclivity for false alarms. Such a unit for attachment to the reactor cover gas piping (meeting ASME code requirements) and for exposure to radiation, was very expensive. Cover gas samples could be obtained by use of the existing System 94 cover gas sample station and the System 94 shielded gas sample bottle. In addition, System 82 was installing on-line monitors in the reactor cover gas system for N_2 and O_2 to provide continuous monitoring of these impurities. Based on these considerations, the on-line chromatograph and other components in the subsystem were eliminated. The interfaces with System 94 were deleted and were included in System 82.

13.2.3.7 EVST, FHC, and IHTS Cover Gas Sampling and Monitoring Subsystem

System 85 added a gas sample bottle to its equipment which could be connected to sample taps provided by System 82 on its piping. This would allow samples to be taken for analysis in the combined laboratory in the PSB.

13.2.3.8 Incoming Argon and Nitrogen Acceptance Sampling Subsystems

In response to a System 82 requirement, this subsystem was added to System 85. By agreement between Systems 85, 82, and 45, System 82 would provide connections on the fill lines to the System 82 liquid argon and nitrogen Dewars from which System 85 could draw samples of the incoming cryogenic liquids using a cryogenic sampler provided by System 45.

13.2.3.9 PSB Combined Laboratory

During the final design phase, System 85 identified the laboratory equipment needed in the PSB combined laboratory for analyses on the basis of RDT F3-40T. This equipment included a sodium analysis cell with shielding window, master-slave manipulators, and transfer tunnel. Other major equipment items included glove boxes with a Dri-train unit, gas chromatographs, oxygen, hydrogen, and carbon determinators, and other laboratory instruments to be used specifically for the analyses of System 85 samples. The equipment to be furnished by System 85 was added to the equipment list in SDD 85. The other needed equipment and analysis cell would be provided by System 45. The laboratory was shared with facilities for the analyses of balance-of-plant samples.

13.2.4 Design Changes from the Final Design Review

13.2.4.1 Final System Design Review

A Final System Design Review of System 85 was held on January 12 and 13, 1983, based on Revision 28 of SDD 85, Sections 1 and 2 and on equipment specifications and specification control drawings listed in Ref. 16, which reference also provides a list of comments and action items. In the reference, System 85 had provided responses to the design review action items with the intent that applicable action items would be reflected in changes in the SDD and other documentation. These action items had been incorporated in the

system documentation, to the extent shown in the responses, at the time of termination.

13.2.4.2 Component Final Design Review

On July 15, 1977, a Final Design Review was held for the sodium sampling packages and the plugging temperature indicators utilizing the documents listed in Ref. 17. All the action items have been incorporated in the applicable documentation, and the final design was approved.

On October 20, 1982, a Final Design Review was held for the System 85 sodium sampling cell shielding windows based on the documents listed in Ref. 18. All action items have been incorporated in the shielding window design, and the final design has been approved.

On February 15, 1983, a Final Design Review was held on the analysis cell shielding window based on the documents listed in Ref. 19. The final design was approved.

13.2.4.3 Final Working System Design Review

A Final Working System Design Review (FWSDR) was scheduled for the Impurity Monitoring and Analysis System on January 18 and 19, 1983, with Westinghouse-LRM and the CRBRP Project Office. A design review package was transmitted to the participants by Ref. 20.

As part of the FWSDR, the System 85 Final Design Report (Ref. 6) was used to demonstrate the implementation of requirements from OPDD-10, interfacing systems, and System 85 design evolution. This document also was used to demonstrate the System 85 response to FFTF experiences and recommendations. All such recommendations had either been included in the system design or were added after they were identified by FFTF. The design report also showed the System 85 responses to Key Systems Task Force recommendation. All have been considered and applied to the system design.

Comments to the FWSDR design package were received prior to the design review (Ref. 21) and after the discussions on the system design and documentation (Ref. 22). System 85 responded to these comments in Ref. 23 which was reviewed and modified by Westinghouse-LRM before transmittal to the Project Office by Ref. 24. The responses in Ref. 24 provided three types of implementation: the comments were rejected as being unnecessary or unneeded; the comments were rejected as already being in the design; or the comments would be incorporated by a procedure described in the response. Those comments scheduled to be incorporated have all been implemented except that not all the operation and maintenance manuals for the System 85 components have been included in the Document Status Report.

Most of the comments were corrective in nature with a few that required minor component design changes. There were none which required system design changes.

After the FWSDR, a consolidation meeting was held between Westinghouse-LRM and System 85 on March 16, 1983. During this meeting, the results of which were transmitted by Ref. 25, a number of engineering change proposals (ECPs) to SDD 85 and baselined documents were identified to be incorporated to produce the final design. Other documents were identified which needed revision for the final design. Many of these proposed changes were the result of the FWSDR and had been incorporated in the latest revisions of the applicable documents. None of the planned changes from the consolidation meeting involved system design changes.

13.3 Key Problems and Resolution

The only key problem that arose in System 85 was due to the late development of the System 85 laboratory requirements which impacted the existing design of the combined laboratory in the plant services building.

The general requirements and instrumentation needs for the combined laboratory had been developed by System 45, as designer of the laboratory facilities, and balance of plant buildings system (System 20), as designer of the physical structure of the laboratory. In 1979, System 85 identified the need for a shielded analysis cell in the combined laboratory. In 1981, ECP A85040 was approved which defined the System 85 laboratory equipment requirements including the analysis cell. Integrating these new requirements into the design of the combined laboratory required extensive coordination and interfacing efforts between System 45, which handled the balance-of-plant requirements and the gas analyses for System 85, and System 85, which provided the analyses of radioactive and nonradioactive sodium.

Ultimately, the combined laboratory was designed to meet the requirements of System 45 and System 85. In this design, the area for sodium analyses and the analysis cell were separated from the balance of the laboratory. System 85 would provide the special equipment for handling and processing sodium samples in preparation for analyses. All other equipment and laboratory items would be provided by System 45 as identified in Interface Control Document (ICD) CA 53706 (Ref. 26). During the coordination meetings, it was agreed that System 85 would place its unique service requirements directly on the supplying system while normal requirements would go by way of System 45 to the supplying system.

At the termination of the project, there was still one problem to be resolved. Since the shielded analysis cell was added to the existing laboratory, its size impacted the available floor space. ICD CA 53706 specified that the cell wall was to be thick enough to reduce the radiation from 100 g of primary sodium at 1 ft from the inner wall to meet Zone I criteria (Ref. 5) at the outer surface of the cell wall. Initial calculations of the source strength indicated that 3 ft of concrete would be sufficient for this purpose. Subsequent calculations of source strength by System 85 and of the shielding requirements by B&R indicated that as much as 4-1/2 ft of concrete would be required. Cell walls of this thickness would severely hamper laboratory operations due to the reduction of usable floor space. Since there was

so much congestion in the laboratory, the analysis cell was placed on HOLD in both SDD 20 and SDD 45 until the interface between the cell and the laboratory could be better defined in a pending ECP by System 85 and a decision could be made by the project to increase the wall thickness, if needed for radiation protection.

13.4 Final System Status

13.4.1 Status at Project Termination

Revision of the system design continued after the transfer of design responsibility to Westinghouse (Ref. 27). At termination, the baselined SDD 85 with Sections 1 and 2 and Appendices A, B, D, E, F, P, and Q, at termination was Revision 33. Included in this revision were P&IDs for the various subsystems as follows:

N099851021: Rev. 14, P&I Diagram, System 85, Primary Sodium Sampling System

N099851024: Rev. 12, P&I Diagram, System 85, EVST Sodium Sampling System

N099851022: Rev. 10, P&I Diagram, System 85, IHTS Sodium Sampling System

The following unbaselined sections of the SDD had been transmitted to Westinghouse (Refs. 28 and 29) for approval:

Appendix X: Maintenance Outline Procedures, D5

Section 3: System limitations, Set Points, and Precautions, D4

Section 4: Operations, D4

Section 5: Casualty Events and Recovery Procedures, D5

Section 6: Maintenance, D6

A complete listing of the applicable documents for System 85 are shown on the System 85 documentation diagrams, Figures 13-7 and 13-8, except for the documents noted on Figure 13-8, which were in work. The documents in the diagrams totally describe the system design at termination.

There are no open items from the two design reviews remaining to be incorporated in SDD 85.

13.4.2 Outstanding Approved Changes

At termination, there was one approved ECP G1437 (Ref. 30), that was not incorporated in SDD 85. This ECP proposed adding a requirement that System 85 provide sampling for the GE sodium-containing components during argon inerting of these components to verify acceptable levels of impurities in the cover gas spaces before sodium fill.

13.4.3 Outstanding Activities to Complete

13.4.3.1 SDD 85

In order to update SDD 85, Sections 1 and 2, and the baselined appendices, ECP G1437 should be incorporated in these documents. The system P&IDs should be updated to include the freeze seal positions on the inlet/outlet piping of the PTI/SSP packages. This detail had been erroneously deleted during P&ID revisions.

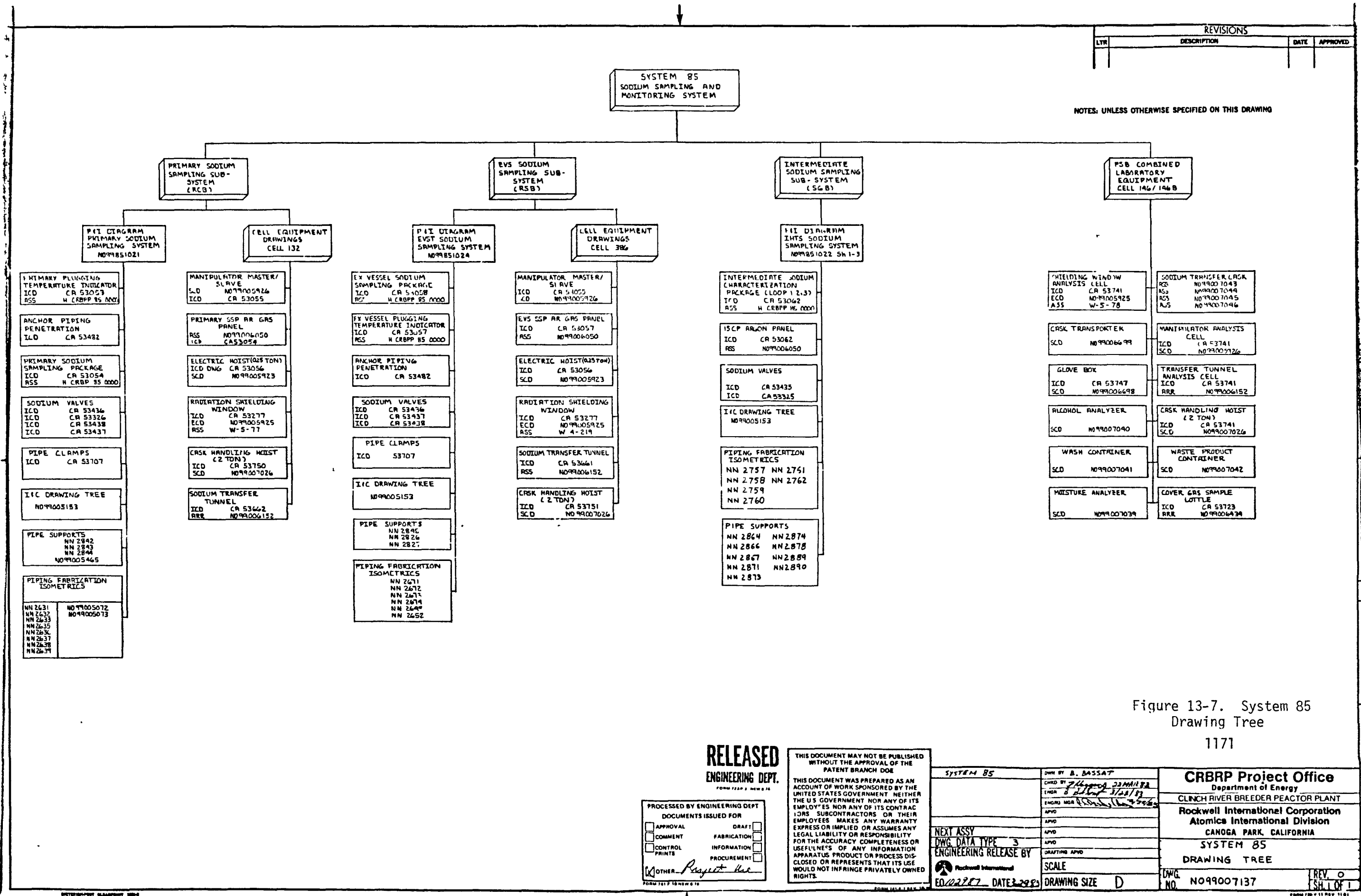


Figure 13-7. System 85
Drawing Tree

1171

RELEASED
ENGINEERING DEPT.
FORM 1220-3 NEW 6-74

PROCESSED BY ENGINEERING DEPT
DOCUMENTS ISSUED FOR

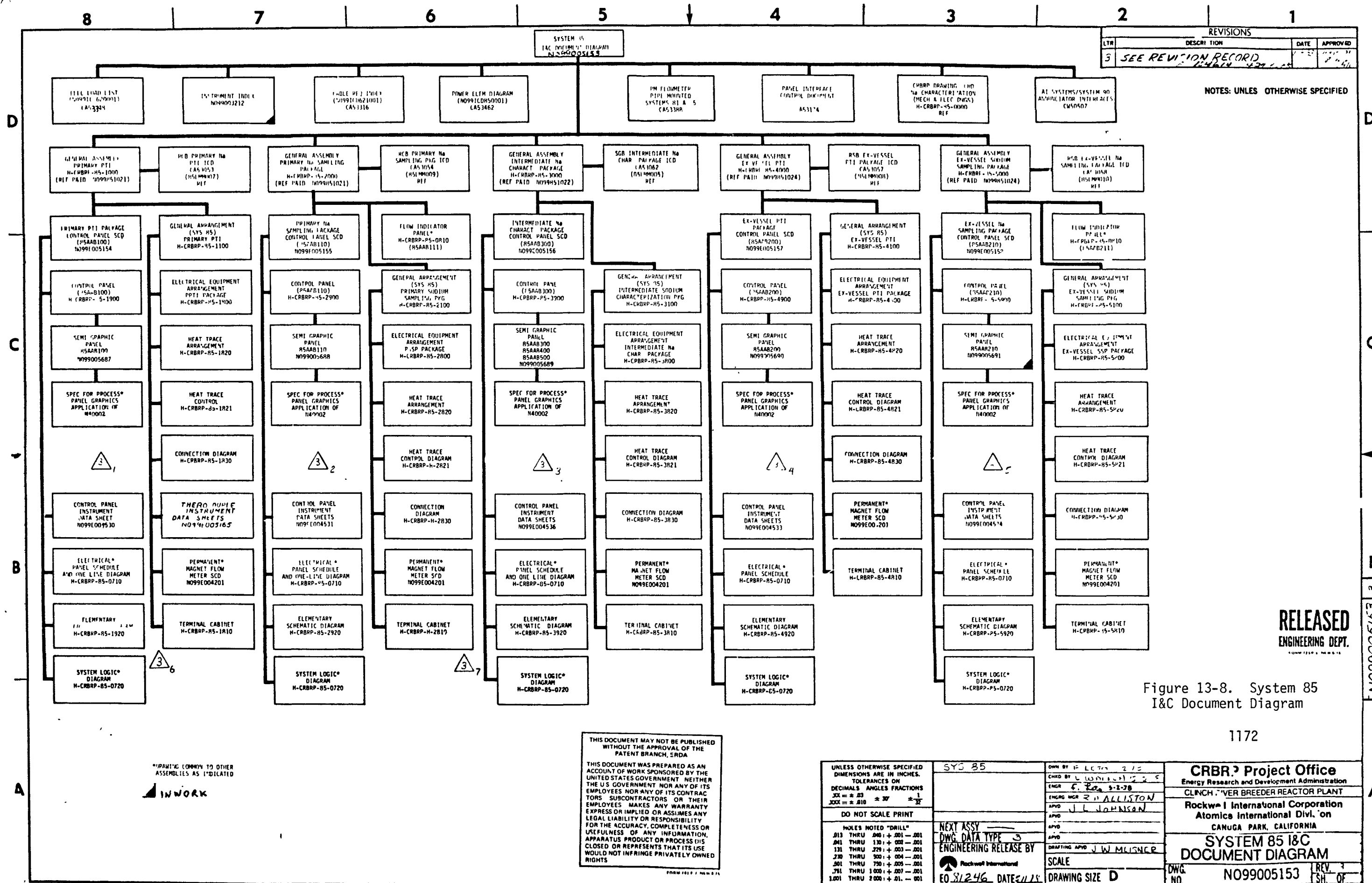
☐ APPROVAL	☐ DRAFT
☐ COMMENT	☐ FABRICATION
☐ CONTROL	☐ INFORMATION
☐ PRINTS	☐ PROCUREMENT

OTHER: *Project Use*

THIS DOCUMENT MAY NOT BE PUBLISHED WITHOUT THE APPROVAL OF THE PATENT BRANCH DOE

THIS DOCUMENT WAS PREPARED AS AN ACCOUNT OF WORK SPONSORED BY THE UNITED STATES GOVERNMENT. NEITHER THE U.S. GOVERNMENT NOR ANY OF ITS EMPLOYEES NOR ANY OF ITS CONTRACTORS, SUBCONTRACTORS OR THEIR EMPLOYEES MAKES ANY WARRANTY EXPRESS OR IMPLIED OR ASSUMES ANY LEGAL LIABILITY OR RESPONSIBILITY FOR THE ACCURACY, COMPLETENESS OR USEFULNESS OF ANY INFORMATION APPARATUS, PRODUCT OR PROCESS DISCLOSED OR REPRESENTS THAT ITS USE WOULD NOT INFRINGE PRIVATELY OWNED RIGHTS.

SYSTEM 85		OWN BY: B. BASSAT		CRBRP Project Office Department of Energy CLINCH RIVER BREEDER REACTOR PLANT Rockwell International Corporation Atomica International Division CANOGA PARK, CALIFORNIA SYSTEM 85 DRAWING TREE	
DRAWN BY: <i>7/24/80 23M/182</i>		CHECKED BY: <i>8/2/80 3/28/81</i>			
ENGR. MOD. <i>8/2/80 3/28/81</i>		APPROVED BY: <i>8/2/80 3/28/81</i>			
NEXT ASSY		DWG. DATA TYPE 3			
ENGINEERING RELEASE BY		DRAWING APPROV		SCALE	
ED: <i>12/2/82</i> DATE: <i>2/2/83</i>		DRAWING SIZE D		DWG. NO. NO99007137	
				REV. 0 SH. 1 OF 1	



RELEASED
ENGINEERING DEPT.

Figure 13-8. System 85
I&C Document Diagram

1172

THIS DOCUMENT MAY NOT BE PUBLISHED
WITHOUT THE APPROVAL OF THE
PATENT BRANCH, SDA

THIS DOCUMENT WAS PREPARED AS AN
ACCOUNT OF WORK SPONSORED BY THE
UNITED STATES GOVERNMENT. NEITHER
THE U.S. GOVERNMENT NOR ANY OF ITS
EMPLOYEES NOR ANY OF ITS CONTRAC-
TORS SUBCONTRACTORS OR THEIR
EMPLOYEES MAKES ANY WARRANTY
EXPRESS OR IMPLIED OR ASSUMES ANY
LEGAL LIABILITY OR RESPONSIBILITY
FOR THE ACCURACY, COMPLETENESS OR
USEFULNESS OF ANY INFORMATION,
APPARATUS PRODUCT OR PROCESS DIS-
CLOSED OR REPRESENTS THAT ITS USE
WOULD NOT INFRINGE PRIVATELY OWNED
RIGHTS

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES. TOLERANCES ON DECIMALS ANGLES FRACTIONS .XX = .03 .30 = .04 .XXX = .010 .30 = .04	SYS 85	OWN BY F. L. C. 215 CHKD BY L. W. J. 215 ENGR F. R. 215 ENGR MGR R. J. ALLISTON APVD J. L. JOHNSON APVD APVD APVD DRAFTING APVD J. W. M. SNCR	CRBR. Project Office Energy Research and Development Administration CLINCH - EVER BREEDER REACTOR PLANT Rockwell International Corporation Atomica International Div. on CANUCA PARK, CALIFORNIA
DO NOT SCALE PRINT	NEXT ASSY DWG. DATA TYPE 3 ENGINEERING RELEASE BY Rockwell International EO 81246 DATE 4/18	SCALE DRAWING SIZE D	SYSTEM 85 I&C DOCUMENT DIAGRAM
NOTES NOTED "DRILL" .011 THRU .001 + .001 - .001 .041 THRU .130 + .002 - .001 .131 THRU .229 + .003 - .001 .230 THRU .300 + .004 - .001 .301 THRU .750 + .005 - .001 .751 THRU 1.000 + .007 - .001 1.001 THRU 2.000 + .01 - .001			DWG. NO. N099005153 REV. 3 SH. OF

Outstanding open items against SDD 85 are pending ECPs to add cover gas analysis at the reactor outlet, to provide structural requirements for components in cells following sodium or NaK spills in these cells, and to resolve the analysis cell interface requirements.

The latest drafts of Sections 3 through 6 and Appendix H have been transmitted to Westinghouse for approval but should include ECP GI437 before baselining. The pending ECPs would also impact these sections and should be considered as open items against Sections 3 through 6 and Appendix H.

13.4.3.2 Other Documents

Equipment specifications for sodium sampling packages and plugging temperature indicators, piping design, sodium transfer tunnels, shielding windows for sodium sampling package and analysis cells, and shielded transfer cask have been baselined and released. Pending ECPs against the equipment specifications would define the weld preparation procedure for piping welds, would provide structural requirements for components exposed to sodium spills in inerted cells, and would impose transients on components connected to the IHTS.

Specification control drawings, assembly drawings, and arrangement drawings have all been released as approved documents. Interface control documents and drawings have been baselined and released except for the System 85 tubing and instrument indices, which are still to be written. A pending ECP to revise trace heating requirements is outstanding on the interface control document between System 85 and System 68.

Installation requirements documents are not available except that a draft of the installation requirement document for the sodium sampling and analysis cell shielding windows was prepared.

Operation and maintenance manuals are not available except for the Shielding Window Operation and Maintenance Manual, which has been released.

All the design drawings for the fabrication of the plugging temperature indicators and sodium sampling packages have been completed by HEDL and are released, ready for use.

Draft versions of the System 85 construction test requirements document and the pre-operational test specification document were transmitted to Westinghouse.

The electrical drawings are complete except for the wiring diagrams, schematics, and front panel arrangements for the PTI and SSP control panels.

Fabrication isometrics and detailed stress analyses for System 85 piping have been provided by Burns and Roe.

13.4.3.3 HOLD Status

There were one System 31, two System 51, and four System 85 HOLDS still outstanding against SDD 85 at termination which cannot be resolved. There is one HOLD to be resolved in the sodium transfer tunnel equipment specification. Interface control drawings contain three existing HOLDS relating to the embeddings and mounting details of the sodium transfer tunnels and manipulators. Three HOLDS remained to be resolved by Burns and Roe for the System 85 detailed stress analyses packages.

13.4.4 Component Status

Before termination, the sodium sampling package and analysis cell shielding windows were fabricated and shipped to storage. The small sodium valves procured through System 81 to be used in the PTI and SSP assemblies have been placed in storage. The fabrication drawings for the PTI and SSP have been approved for use and are available.

13.4.5 Outstanding Problems

The overpressure protection report for System 85 was not completed at the time of termination.

The pending ECP A81398, which was to implement the requirement that the failure of System 85 components following a liquid metal spill would not impact the integrity of Safety Class 1 components in the same cell, has not been completed and therefore cannot be implemented. The results of analyses to determine whether piping, pipe supports, component embedments, and/or components that would be affected by the sodium spill could impact piping isometric drawings and component design.

The pending ECP A85102, which proposed to provide updated analysis cell interfaces with the combined laboratory, could impact the design of the laboratory.

13.4.6 Recommendations for Future Designs

Most of the samples taken by System 85 are to verify the purity of sodium and of cover gases circulating in the various plant systems. System 81 uses the results of these analyses to control the operation of their sodium cold traps, and System 82 uses the results to control the use of fresh, inert gases for purge and dilution of the monitored cover gas spaces. Since System 85 obtains its samples through connections to Systems 81 and 82 piping and since the analyses results eventually are used by Systems 81 and 82 to control operations in their systems, it is recommended that the functions of the impurity monitoring and analysis system be divided between the auxiliary liquid metal system (System 81) and the inert gas receiving and processing system (System 82). Those subsystems for sodium sampling should be included in the design of the auxiliary liquid metal system, and those for cover gas sampling should be included in the design of the inert gas receiving and processing system.

The analysis of sodium from the primary coolant systems will require the handling and processing of sodium that is radioactive to such a degree that the plant personnel handling this sodium must be protected. Therefore, it is recommended that the design of the plant laboratory complex include a shielded analysis cell designed to protect laboratory personnel from the radiations of two 50-g sodium samples in the cell at the same time. Since the handling and analysis of radioactive samples have more stringent safety and contamination criteria than nonradioactive samples, it is recommended that this analysis cell and facilities for the analysis of sodium be included in a radioactive laboratory module that is physically separated from the laboratory facility for balance-of-plant (BOP) sample analysis. The radioactive laboratory, the BOP laboratory, the counting room, and any additional laboratory ancillary space should be designed for safety and controlled sample transfer.

13.5 References

1. RDT Standard AI-5T, "Purity Requirements for Operating Sodium Reactor Systems," March 1976
2. SDD 85, Revision 33, "System Design Description - Impurity Monitoring and Analysis System," September 28, 1983
3. WARD-D-0244, Revision 3, "CRBRP Preservice and Inservice Inspection Requirements for Sodium and Cover Gas Systems," January 7, 1983
4. WARD-D-0177, Revision 14, "CRBRP ALARA Criteria and Guidelines," July 14, 1983
5. OPDD-10, Revision 146, "Overall Plant Design Description," August 1, 1983
6. N099TI850002, "System 85 Impurity Monitoring and Analysis Final Design Report," July 26, 1983

7. N099TI851006, "Updating System 85 Primary, EVST, and IHTS Sodium Sampling Systems Flow Diagrams," March 14, 1983
8. N10036, "Failed Fuel Location Subsystem (FFLS - System 94)," September 24, 1982
9. RDT Standard M 14-1T, "Sodium Cover Gas - Purchase Specification," January 1975
10. RDT Standard F 3-40T, "Methods for the Analysis of Sodium and Cover Gas," November 1978
11. DRR-099-850-001, "Design Review Minutes for the Impurity Monitoring and Analysis System SDD 85," October 7, 1974
12. DRR-099-620-001, "Design Review Minutes for the Impurity Monitoring and Analysis Instrumentation System SDD 62," September 26, 1974
13. N099-DRR-850-002, "Impurity Monitoring and Analysis (System 85) Preliminary Design Review Minutes," August 26, 1975
14. N099DRR850003, "Preliminary Design Review Report, System 85, Sodium System Components, HEDL," November 2, 1976
15. N099DRR850004, "Sodium Monitoring Components Preliminary Design Review," February 15, 1977
16. N099DRR850007, "Final Design Review of System 85," March 11, 1983
17. N099DRR850005, "Final Design Review, CRBRP, System 85, Na Sampling Packages and Plugging Temperature Indicator (PTI)," July 28, 1977
18. N099DRR850006, "Design Review Report for the CRBRP, System 85 Shielding Windows," February 28, 1983

19. NO99DRR850008, "Design Review Report for CRBRP, System 85 Analysis Cell Shielding Window (Cell 146B)," April 27, 1983
20. Letter 82ESG-7909, "System 85 FWSDR Review Package," November 9, 1982
21. Letter LR830029, "CRBRP; Westinghouse Comments on System 85 FWSDR Review Package," January 14, 1983
22. Letter CR-747:DH:83-042, "System 85 Final Working System Design Review," March 4, 1983
23. Letter 83ESG-1969, "AI Disposition of System 85 FWSDR Comments," April 4, 1983
24. Letter LR830291, "CRBRP; System 85 FWSDR Comment Disposition," May 24, 1983
25. Letter LA830104, "CRBRP; System 85 Consolidation," March 31, 1983
26. ICD NO99ICD850007 (CA53706), "System 85 Analysis, Laboratory Equipment, and Laboratory Interface Requirements on System 45," August 12, 1983
27. Letter 83ESG-3667, "Transfer of System 85 Design Responsibility to W-OR," June 15, 1983
28. Letter 83ESG-3563, "CRBRP; Transmittal of Draft 4 of Section 3, Draft 5 of Section 5 and Appendix H, and Draft 6 of Section 6 of SDD 85," July 1, 1983
29. Letter LR830567, "CRBRP; Request for Approval to Baseline SDD 85, Sections 3, 4, 5, and 6," October 19, 1983
30. Letter XL-595-830618, ECP G1437, "IHTS Required Sampling Frequency," October 5, 1983

14.0 PLANT CONTROL SYSTEM DESIGN

14.1 Summary of Functions and Description

The function of the plant control system is to provide overall control and coordination of the reactor, heat transport system, steam generator, turbine and balance of plant for all normal operating modes. These modes include automatic and manual plant operating modes for plant startup and shutdown (0+ to 40% rated power) and the power range (40% to 100% rated power). In addition, the following requirements are imposed by Reference 1:

- A. Automatic control range between 40% and 100% rated power conditions.
- B. Automatic control capability (load follow) for:
 - (1) 3% of full power/minute ramp
 - (2) 10% of full power step
- C. Regulation of specific plant variables over a desired part-load profile.

The plant control system designed for the Clinch River Breeder Reactor Plant (CRBRP) meets these functional and performance requirements by providing a two level configuration as shown in Figure 14-1. The top level is the supervisory controller which, based on operator power commands and selected process signals, provides demand signals to the level two controllers in order to maneuver the plant on the desired part-load profile. The part-load profile is given in Figure 14-2. This figure shows that the primary and intermediate flow rates vary with power, thus enabling steam to be delivered to the turbine at a constant pressure of 1465 psig and a slightly drooping temperature profile over the power range.

The level two controllers are process servos that regulate specific plant variables in response to the supervisory control commands. These control systems include the following:

Figure 14-1 CRBRP Plant Control Schematic

EXPECTED SYSTEM PARAMETERS VERSUS POWER LEVEL FOR CONSTANT STEAM THROTTLE PRESSURE (14 65 PSI)

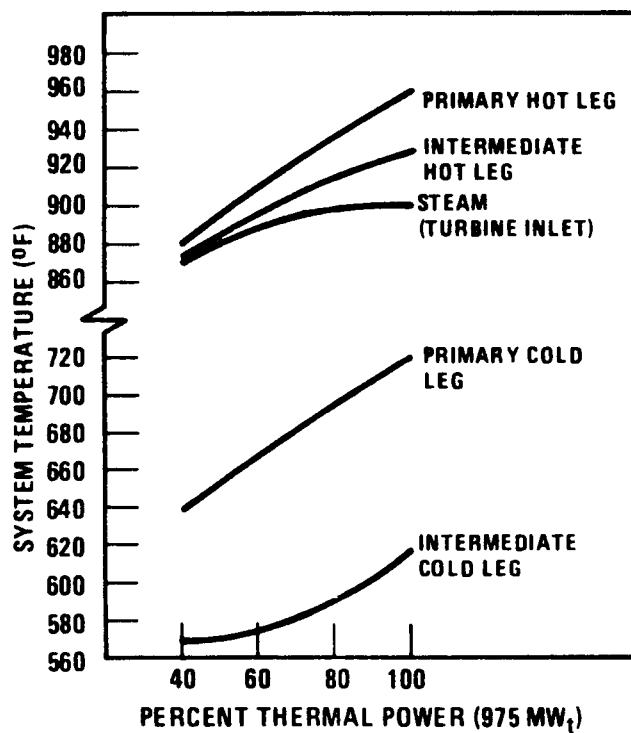
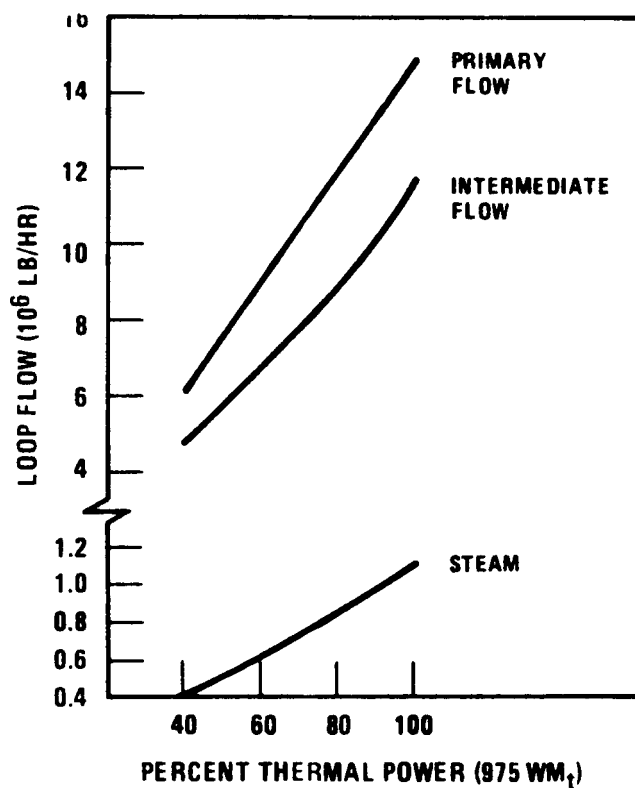


Figure 14-2 CRBRP PLANT LOAD PROFILE

- o Reactor Control
- o Primary Heat Transport System (HTS) Flow Control
- o Intermediate HTS Flow Control
- o Drum Level Control
- o Turbine Control
- o Turbine Bypass Control

The reactor control system is comprised of two controllers, an inner loop flux controller and an outer loop core exit temperature controller. The flux controller accepts a flux demand signal from the core exit temperature controller and provides rate and direction signals to the control rod drive mechanism (CRDM). Feedback is provided by a median select function of the primary flux monitoring instrumentation. The setpoint for the core exit temperature controller is provided by the supervisory controller with the feedback signal as the average of 30 core exit thermocouples. In addition, rod block circuitry is included which is activated based on the secondary flux instrumentation exceeding a limit.

The primary and intermediate flow control systems are both comprised of three identical controllers, one for each HTS. Each controller has an inner loop pump speed controller and an outer sodium flow controller. Speed and flow controller designs are System 56 and 90 responsibilities, respectively. The speed controller receives a command from the flow controller and provides a position command to the scoop tube of the fluid coupled motor generator (MG) set pump drive. The flow controller is commanded by the supervisory controller with the feedback signal as the median select of three permanent magnet flowmeter signals.

The drum level control is an independent system comprised of three identical controllers, one for each steam generator. Each controller regulates water in the drum by throttling a feedwater valve in response to a comparison of measured water level and setpoint and a comparison of steam leaving the superheater and feedwater flow entering the drum. The design of this system is the responsibility of System 56.

The turbine control is an electro-hydraulic control system provided by System 72. The supervisory control system regulates the turbine load by controlling the position of a load set motor by means of contact closures.

The turbine bypass control is provided by System 69. This system allows opening of valves to bypass steam to the condenser based on commands from the supervisory control system.

The supervisory controller coordinates the control actions of these process controllers to obtain stable operation as well as transient performance that meets overall plant requirements. Operator inputs for power level and rate via the main control panel are processed by a unit load controller to provide a ramp signal to the turbine control system. Commands to the process servos to change the Nuclear Steam Supply System (NSSS) power level are determined from a programmed part-load profile and the plant load determined from steam generator exit steam flows.

The supervisory controller also provides trim controllers. The turbine inlet steam pressure controller trims intermediate sodium flow in all three HTS to regulate turbine inlet pressure to a constant 1465 psig. The turbine inlet steam temperature controller trims reactor core exit temperature to regulate turbine inlet temperature in accordance with the part-load profile. Also provided are manual trim controls on the main control panel that allow the operator to adjust setpoints of the HTS sodium flow controllers. This feature allows trimming flows to balance HTS temperatures.

A more detailed description of the plant control system is given in Reference 2.

14.2 Evolution of Design

14.2.1 Conceptual Design

The emphasis during the conceptual design phase was on establishing a plant control system configuration, producing a plant simulation upon which transients could be run and conducting initial stability and transient studies. These items are addressed below:

A. Plant Control System Configuration

A plant control system design which utilizes a combination of feedforward and feedback controllers in a two-level configuration was selected. This configuration allowed for tight regulation of a set of plant variables by the process servos and overall control and coordination by the supervisory controller. This type of control was judged necessary for operation in accordance with the part-load profile at steady state and for load follow. Since load follow was a design requirement, it was recognized that some type of control function on turbine inlet conditions would be needed to meet requirements on swings of turbine inlet temperature and pressure. Load follow, where the turbine load is changed first followed by a change in the NSSS power, is an operational mode where these swings could be quite large. Therefore, two trim controllers were included in the design; a turbine inlet pressure controller that trims intermediate flow and a turbine inlet temperature controller that trims reactor core exit temperature.

For the reactor control system an outer core exit temperature control and inner loop flux control was selected. This configuration had the following advantages:

- (1) The highest loop temperature (core exit) is controlled and therefore the temperature seen by other components is limited.
- (2) The loop transport time delay and outlet plenum lag is eliminated in the feedback path resulting in a faster responding system.
- (3) The effect of primary sodium flow or inlet temperature disturbances on the hot leg temperature can be mitigated.
- (4) The flux loop can be operated alone for startup and shutdown modes.

- (5) The inner flux loop tends to linearize the nonlinear characteristics.

B. Plant Simulation

A model of the plant from the reactor to the turbine was developed and programmed on a hybrid computer. It was decided that the thermal/hydraulic code DEMO was too detailed for this application so a simplified version of this code was used. The size limitation of the computer was also a constraint on the model detail. A single and lumped loop representation in the HTS was chosen to reduce the program size while still allowing loop-to-loop interactive studies.

C. Stability and Transient Analysis

The plant control system configuration and servo designs were evaluated by conducting stability and transient analysis. The stability (phase and gain margins) of the system were determined by running frequency responses at selected power levels throughout the power range. Also, step and ramp responses were run to evaluate the system performance relative to time domain requirements. Emphasis was placed on the reactor control since stability and transient performance (speed of response) was judged to be very important. References 3, 4 and 5 provide results of this analysis.

14.2.2 Preliminary Design

Work in the preliminary design phase was concentrated in the following analysis and hardware areas:

A. Feedback Scheme

A median select feedback scheme was selected for the control loops. With this scheme three redundant sensors are used with the middle signal in magnitude always selected as the feedback. Two instrument channel failures are necessary before a plant disturbance and

possible plant scram occurs. This method was selected over auctioneer high and low schemes since these only require one instrument channel failure to cause a plant scram. Extra sensors are not required since PPS had chosen a two out of three logic system. The median select method also had the advantages of reduced accuracy error in the feedback path and that of allowing PPS channel testing without effecting control where sensors were common to both control and protection.

For the core exit temperature controller an average and reject scheme was selected. In this scheme thirty thermocouple signals are compared to high and low limits and rejected from the average if exceeded. This scheme was selected to provide adequate coverage of the core and the temperature variation from location to location as well as eliminate the effect of failed sensors on the output.

The rationale employed in the selection of thermocouple locations is summarized as follows:

- (1) Approach the core mixed mean temperature and follow it during the cycle as closely as possible.
- (2) Monitor the limiting assemblies.
- (3) Scatter the control positions evenly between inner and outer regions.
- (4) Provide redundancy.

More detail on this subject is provided in Reference 6.

B. Rod Block

A rod block feature was included in the reactor control design to meet System 31's requirement to regulate flux below 103%. The scheme selected compares the median select secondary flux instrumentation to

high flux and flux-to-flow setpoints. With the flux controller using the primary flux instrumentation and the rod block the secondary flux system, this configuration ensures that a failure of either flux monitoring system cannot fail both the flux controller and rod block.

C. Stability, Sensitivity and Set Point Studies

Frequency responses and transients (step and ramp responses) were run to establish setpoints for controller deadbands and compensation functions and to determine stability margins. Setpoints are provided in Section 6 of Reference 2. Emphasis in this stability analysis was placed on determining the existence of limit cycles (periodic, non-sinusoidal oscillations). Analytical methods such as describing function and popov criteria as well as computer simulation were used. A major effort was the evaluation of the flux controller under the effects of a positive power coefficient due to core assembly bowing in the startup range. Results are reported in Reference 7.

Following the stability studies, sensitivity studies were conducted for the reactor, primary sodium flow and supervisory control systems. In these studies parameters of the plant, sensors and actuator dynamics were varied in the model. The effect of this variation on stability margins was then determined by repeating frequency responses. These studies were performed because there was some degree of uncertainty associated with plant data used in the simulation and it was desirable to show that stability and transient performance was not sensitive to this uncertainty.

Following the sensitivity studies, setpoint studies were conducted where, after the parameter variation, compensation settings were changed to bring stability margins back to their original values. The range of these changes were then incorporated into the equipment specification. Results of the sensitivity and setpoint studies are reported in References 8, 9 and 10.

D. Hardware Concept Trade-Off Study

A hardware study was performed prior to writing an equipment specification. This study was initiated to determine the feasibility of utilizing Direct Digital Control (DDC) for the plant control system. Results given in Reference 11 show that DDC at the time of the study was competitive with analog control hardware on performance, cost, and system availability. However, a DDC process computer could not be qualified as a seismic category II component. Therefore, based on this limitation, an analog hardware concept was chosen for the plant control system design.

E. Turbine Trip Without Reactor Trip

In addition to plant control system analysis for normal operating modes, analysis was conducted for off-normal events as well. A scheme was developed for keeping the reactor and NSSS on-line in the event the turbine tripped. With this feature a significant amount of time could be saved by not having to perform a startup from hot standby. Following the turbine trip, steam was bypassed to the condenser and the NSSS was reduced in power to 40 percent at 3 percent of full power/minute. Since the supervisory control system determined load based on steam generator steam flow, no switching logic was required.

F. Flow Trim

In the preliminary design phase the method by which HTS sodium flow would be trimmed was determined. A manual control scheme was selected since fast response was not needed and it offered better system reliability and plant availability than an automatic scheme.

14.2.3 Final Design

During this design phase the equipment order was placed while controller setpoints were finalized. Plant parameters in the simulation were updated

based on final component designs and prototype testing and transients were repeated as necessary.

The primary flow controller was included in the sodium pump tests at Energy Technology Engineering Center (ETEC) as a test article. Step, ramp and frequency response tests were conducted on this unit from 30 to 100 percent rated flow along with the pump speed controller and fluid coupled MG set pump drive. Test results showed that the controller performance met all design requirements. One change made to the controller to improve steady state performance was the elimination of the deadband function. This function was originally considered necessary to stop undesirable controller action in response to loop flow fluctuations.

The plant operating philosophy and procedures were also developed during this design phase and documented in Reference 2. The plant modes of control were defined for the fully automatic mode, semi-automatic and manual modes. Pre-operational test specifications were also developed.

Other work initiated during this phase included the development of an interface between the plant control system and the plant protection system, development of part-load profile allowable limits of operation and the design of failure detection circuits for loop controllers.

14.3 Key Problems and Resolution

In the course of designing the plant control system for CRBRP several problems were encountered that required resolution. These items are addressed below.

14.3.1 Delayed Turbine Trip Following Reactor Scram

In some plant control system designs the turbine is tripped immediately after a reactor scram. This type of logic was found to be present in the bypass valve control system in the conceptual design phase. Evaluation of this transient showed a problem at the superheaters. As a result of the drum pressure being greater than the relief valve settings at the superheater exit, these valves would pop open with a loss of water inventory upon the turbine

tripping. In addition the thermal transient on the steam generator was more severe than expected.

The solution to this problem that was incorporated into the plant control design is described below. Rather than tripping the turbine immediately following the scram, the turbine trip was delayed. A setpoint of 1200 psi was selected for initiating the trip after coordinating this value with normal loading events, bypass control, outlet steam isolation system and the protected air cooled condenser system. In other words, the energy stored in the HTS was allowed to decay or dissipate through the turbine until the drum pressure dropped below the relief valve setting. Tripping of the turbine at this point did not result in the valves opening. In addition, the bypass valves were switched into a pressure control mode following the turbine trip to regulate main steam pressure and thus control the energy dissipation. In this way, the transient on the steam generator was improved.

14.3.2 Turbine Trip Without Reactor Trip

Although not a requirement, it was a goal of plant control designers to develop a system that would allow the reactor to stay operational for a turbine trip event. The thinking was that features could be incorporated into the design that would open the bypass valves and decrease reactor power down to 40% in a controlled fashion without activating Plant Protection System (PPS) in preparation for a restart of the turbine.

Such a system was developed as described below. An early plant control design used turbine impulse chamber pressure as an indication of plant load. Commands to the plant process servos were based on this signal. This signal, however, disappeared (zero value) when the turbine tripped and logic was required to switch to another calculation for determining the setpoints. The solution that was chosen was to use main steam system flow as an indication of load. The steam flow signal would always be present regardless of the event and additional logic would not be required. A measurement of main steam flow was not available so the sum of the individual loop steam flows measured at the superheat exits was selected. To smooth out the transient at the beginning a rate limiter of 20%/second was included. The normal load demand

signal was compared to the generator MW meter and used to control opening at the bypass valves. This method was called load error.

Unfortunately, this scheme could not be implemented on a plant-wide basis. Due to the sudden decrease in steam flow exiting the superheaters upon the turbine trip, the PPS steam-to-feedwater flow trip was activated causing a plant scram. Several proposed solutions were evaluated such as adding a delay to the PPS function to smooth out the steam flow signal or faster acting bypass valves. These did not turn out to be viable solutions, however. It is suggested that on future plants the designs of plant control, PPS and turbine bypass system be coordinated from the initial design phase to provide this capability.

14.3.3 HTS Sodium Flow Trim

It was recognized that HTS loop temperatures would not necessarily be the same and there may be a need for some type of control. This issue was studied first to determine the magnitude of the imbalance and second to select a control method to be included into the overall plant control system design.

The magnitude of loop-to-loop temperature imbalances was determined using the plant simulation as documented in Reference 11. Differences in loop flows and differences in loop fouling and plugging of heat exchangers were identified as the major causes of this condition. Maximum error values were determined and then programmed as disturbances in the simulation. The error in loop flow was determined by assessing the various control, instrumentation and sensor errors. The most important of these was flowmeter inaccuracy which was comprised of the following error terms:

- o Signal conditioning ($\pm 2\%$)
- o Thermal calibration ($\pm 1-1.5\%$)
- o Repeatability ($\pm 0.5\%$)
- o Pipe movement ($\pm 1\%$)

The accuracy values correspond to a 5% accurate flowmeter as required by PPS. Simulation results showed loop-to-loop differences of 34°F at 100% power and

74°F at 40% power. Since these values were quite large, some type of control had to be considered for balanced plant operation. A particular concern was the effect on reactor core inlet temperatures and whether some inlet assembly temperatures could be higher than values assumed in design calculations.

Consideration was given as to whether automatic or manual control should be used for this function. Some aspects of the problem are addressed below:

A. Operator Response Time

Automatic control offers the advantage of continuous regulation and relieves the operator of the responsibility of making setpoint adjustments. This capability is important in applications where quick response is required. In this application, however, control is required for temperature variations due to deterministic type errors (e.g., instrumentation accuracy and calibration) and slowly changing disturbances (e.g., circuitry drift and heat exchanger fouling). For this case automatic control is not considered a necessity. Under manual control, the operator is capable of making the necessary adjustments and since these operations are required infrequently, they are not a burden to the operator.

B. System Reliability/Plant Availability

Automatic control to balance HTS loop temperatures adds hardware to the system in the form of sensors and transmitters as well as analog circuitry. Since three HTS loops are involved, and considering that multiple sensors per loop are needed, the increase in hardware to the plant control system is not trivial. This additional hardware makes the plant control system less reliable and this, of course, has an unfavorable impact on plant availability. With manual control, the impact on system reliability and plant availability is much less since a minimal amount of additional hardware is required.

C. Control Stability

Plant stability could be a problem with automatic HTS temperature control. Sodium flows and temperatures in each HTS loop and between HTS loops (depending on the design) are more tightly coupled. Due to this increased interaction of plant variables, oscillatory system behavior is a possibility.

D. Operator Response to Failures

When plant variables deviate from the part load profile due to degraded hardware, it is desirable for the operator to detect the off-normal condition and take appropriate action to keep the plant on-line. This operation requires recognition of the condition and then switching of the plant mode of operation to bypass the component malfunction. With automatic HTS temperature control, this task becomes harder to accomplish. Depending on the controller configuration, several plant variables may be affected as well as several HTS loops. With a simpler manual control scheme, this process is more straight forward and easier to perform. The source of the problem is also easier to detect resulting in shorter repair times.

E. PPS Impact

HTS temperature control as part of the plant control design may introduce failure modes for which the PPS does not provide adequate protection. The PPS can be a constraint, therefore, on the control system design and impact performance. Whether automatic or manual control is used, this impact needs to be considered.

F. Hardware

The choice of hardware can affect both design and performance of the control system. Microcomputers, for example, have less drift than analog circuitry thereby decreasing the frequency of required

setpoint adjustments in manual control. Initial and periodic turning of setpoint function generators is also easier and performed more accurately by changing input computer data than by making adjustments on analog printed circuit boards. The design can also be influenced since more complicated control schemes can be implemented using microcomputers than analog hardware.

It was decided that manual control would be a better method of control considering all aspects of the problem. This method was viewed as simpler and more reliable and therefore result in a higher plant availability. Stability of automatic control was a major concern. It is noted, however, that for a future plant design automatic control should be considered. Advances in instrumentation may resolve any such concerns expressed above.

14.3.4 Control/Protection Interface

A difficult problem to resolve was the interface between the control system and plant protection system. In order for successful operation in the power range, the PPS system trip points, taking into account all sensor and instrumentation errors, should be outside a region of the operating profile for normal transients. Otherwise, there is some probability of tripping for normal operation. For the flux-to-flow and primary-to-intermediate flow mismatch subsystems this situation was the case. A discussion of this problem in more detail is provided in Section 15.3.1 of this report.

14.3.5 Rod Blocks and Failure Detection

As a result of plant licensing the following requirement was added to the plant control system:

Means shall be provided to assure that the likelihood of continuous, uninterrupted withdrawal of multiple rods shall be sufficiently low to justify exclusion of such an event as an hypothetical core disruptive accident (HCDA) initiator when combined with the assumption of a failure to scram.

This requirement was added to remove rod bank withdrawal without scram from the list of potential HCDA initiators. Since it was added in the final design phase and a complete redesign was not acceptable, modifications were considered. A different approach may have been taken at an earlier point in the design. It is recommended on future plant designs that this issue be resolved earlier so that this requirement can be integrated into the design process.

The following changes to the plant control system which are documented in Reference 12 were made to meet this requirement:

- A. Modify the rod misalignment rod block system in the primary rod controller so that a block of all rods also occurs when the average bank position exceeds a limit set point. This feature protects against malfunctions by using rod position.
- B. Modify the flux controller and rod controller to add a second redundant rod withdrawal stop circuit. This feature protects against malfunctions by using flux and flow.
- C. Modify the flux controller to add a failure detector that disables CRDM commands (rate and direction) upon detection of a failure. Failure detectors for other plant controllers were also added. This feature protects against controller malfunctions as well as malfunctions of upstream controllers and feedback signals.

For A above, the limit set point was made adjustable so that it could be set at some specified time interval as a function of burnup. Future work was to determine how the bank position set point was to be calculated, the organization responsible for developing the set point calculation, the frequency of adjustment and the type and frequency of tests/calibrations to be performed on the rod blocks.

It was estimated that the modifications described above would have a probability of failure of preventing a bank withdrawal of 10^{-6} which is

sufficiently low to justify exclusion of a continuous uninterrupted bank withdrawal with assumed failure to scram as an HCDA initiator.

14.4 Final System Status

The System 90 Final Working System Design Review was held the last week of June, 1983. The design for the most part was well received by the review team. There were, however, some areas of the design where the team thought additional work was needed. The major items are listed below.

- A. An interface document should be baselined between plant control and plant protection. In particular, interface problems involving the flux-to-flow and primary-to-intermediate flow mismatch subsystems need to be resolved.
- B. Adjustment intervals and method of calculating bank position set point for the misalignment rod block system needs to be determined.
- C. The method of bypass control from the supervisory control system should be reviewed. Pressure control should be investigated.
- D. A method for balancing HTS loops should be developed and included as an operating procedure.

14.5 References

1. OPDD-10, Overall Plant Design Description.
2. System Design Description, SDD-90, Plant Control System, Rev. 96.
3. "Initial Reactor Control System Design for CRBRP," WARD-D-0088, January 1975.
4. Letter WL70197, W. W. Dewald to W. M. Jacobi, "CRBRP; Plant Control System - Primary HTS Flow Control Design Report," February 9, 1977.
5. "Initial Supervisory Control System Design for CRBRP," WARD-D-0096, April 1975.
6. Letter WL60348, C. A. Anderson to W. M. Jacobi, "CRBRP; CRBRP Core Exit Instrumentation," dated March 12, 1976.
7. Calculation LIC-CP-79-3, Flux Control Stability in the Startup Range with Core Assembly Bowing Reactivity.
8. "Reactor Control Sensitivity Analysis for CRBRP," WARD-D-0153, dated May 26, 1976.
9. Letter WL61701, W. W. Dewald to W. M. Jacobi, "CRBRP; Plant Control System - Reactor Control System Setpoint Analysis," dated December 9, 1976.
10. Letter WL70914, W. W. Dewald to W. M. Jacobi, "CRBRP; Plant Control System - Primary HTS Flow Control Sensitivity and Setpoint Analysis," dated June 7, 1977.
11. Calculation LIC-CP-77-02, Primary Cold Leg Loop-to-Loop Temperature Variation, dated February 28, 1977.
12. Letter CR-742:SEH:82-351, C. H. Fox to W. J. Purcell, Approval of Urgent ECP W2027, Rev. 0, "Modification to Control and Protection System Requirements."

15.0 PLANT PROTECTION SYSTEM (PPS)

15.1 Summary of Functions and Description

The Reactor Shutdown System (RSS) is a portion of the PPS which is described and documented in the System Design Description (SDD) 99 [1]. It protects the public and plant against abnormal plant conditions which, if left unmitigated, would result in unacceptable consequences. The RSS performs this function by shutting down the reactor and heat transport systems upon sensing an abnormal condition.

The RSS includes the Primary Shutdown System (PSS) and the Secondary Shutdown System (SSS). The two systems are independent, redundant, and diverse such that the failure of one system will not prevent the other from shutting down the plant. Due to the intentional use of diversity, however, the performance of the two systems are not equivalent. Greater damage occurs as a result of the generally slower response associated with the SSS instrumentation. This is reasonable in light of the much lower probability of PSS failure concurrent with an abnormal plant condition.

The important safety function performed by the RSS demands design features to protect against hardware failures. Protection against single random failures is provided through redundancy. A shutdown system contains sufficient redundancy to ensure that a single internal random failure will not fail that system. Each shutdown system includes the following redundancy:

- A. Three redundant instrument channels for each plant parameter.
- B. Three redundant protection logic trains arranged in two of three coincidence.
- C. Multiple control rods which will shutdown the reactor to a safe level even if the highest worth rod is not inserted.

Protection against RSS failure resulting from the propagation of failures within the redundant systems is provided through independent redundant

functional units. Independence exists within each shutdown system, between the two shutdown systems, and between the RSS and non-PPS systems. This independence is assured by physical separation (space or barriers) and electrical isolation.

Protection against RSS failure due to common mode effects which impact both Primary and Secondary Systems is provided through functional and equipment diversity. There are some situations, for example the nuclear instrumentation, for which functional diversity is not achievable. In cases where functionally diverse parameters which meet the necessary performance requirements do not exist, equipment diversity is provided between the Primary and Secondary Shutdown System hardware. The Primary Shutdown System is implemented using local coincidence logic; the Secondary, with general coincidence logic [1].

The PSS and the SSS each contain sixteen protective subsystems. Table 15-1 lists these subsystems. Each subsystem uses measurements of one or more plant parameters to sense the occurrence of abnormal plant conditions. Table 15-2 summarizes the types of events, instrumentation, and permissive features associated with the protective subsystems.

The RSS must limit the consequences of the design basis events of Table 15-3 to the specific design requirements of Table 15-4. Note that there is a difference in the level of protection required of the Primary and Secondary Shutdown Systems. The Secondary Shutdown System provides protective action if, and only if, there is a failure of the Primary Shutdown System. Such a failure is considered extremely unlikely. The simultaneous occurrence of the failure of the Primary Shutdown System and an anticipated event is therefore also extremely unlikely and should logically be placed within the major incident damage category. However, prudent design conservatism justifies the more restrictive damage limit of minor incident. Similarly, the simultaneous occurrence of an unlikely fault event and the failure of the Primary Shutdown System is conservatively restricted to the major incident category. The failure of the Primary Shutdown System and the simultaneous occurrence of an extremely unlikely event is so highly improbable that it is not considered part of the design basis of the Secondary Shutdown System for all but the Safe

TABLE 15-1
REACTOR SHUTDOWN SYSTEM PROTECTIVE SUBSYSTEMS

Primary Shutdown System

1. High Flux
2. Flux to Pressure
3. Negative Flux Rate
4. Positive Flux Rate
5. H1S Pump Bus Frequency
6. Speed Mismatch: Loop 1
7. Speed Mismatch: Loop 2
8. Speed Mismatch: Loop 3
9. Reactor Vessel Level
10. Steam to Feedwater Mismatch: Loop 1
11. Steam to Feedwater Mismatch: Loop 2
12. Steam to Feedwater Mismatch: Loop 3
13. IHX Primary Outlet Temperature: Loop 1
14. IHX Primary Outlet Temperature: Loop 2
15. IHX Primary Outlet Temperature: Loop 3
16. Primary Overpower-Low Setpoint

Secondary Shutdown System

1. Flux to Flow
2. Modified Negative Flux Rate
3. Modified Positive Flux Rate
4. High Flow Mismatch
5. Low Flow Mismatch
6. Steam Drum Level: Loop 1
7. Steam Drum Level: Loop 2
8. Steam Drum Level: Loop 3
9. Evaporator Outlet Sodium Temperature: Loop 1
10. Evaporator Outlet Sodium Temperature: Loop 2
11. Evaporator Outlet Sodium Temperature: Loop 3
12. Sodium Water Reaction: Loop 1
13. Sodium Water Reaction: Loop 2
14. Sodium Water Reaction: Loop 3
15. Startup Flux
16. HTS Pump Bus Voltage

TABLE 15-2
PROTECTIVE SUBSYSTEM SUMMARY

<u>Subsystem</u>	<u>Design Basis</u>	<u>Instrumentation</u> ⁽¹⁾	<u>Permissive</u>	<u>Remarks</u>
Primary Shutdown System				
High Flux	increasing reactor power at or near full power operations	o compensated ion chamber o comparator	No	
Flux to Pressure	increasing reactor power or decreasing reactor coolant flow over the normal load range	o compensated ion chamber o 2 primary cold leg pressure detectors o square root module o comparator	Yes	o primary cold leg pressure is a diverse measurement of total reactor coolant flow o two pressure detectors justified by maintenance considerations o permissive provided due to small operating margin at low power levels
Negative Flux Rate	rapid decrease in reactor power leading to undesired thermal (undercooling) transients	o compensated ion chamber o comparator	Yes	o protective feature not required at lower power levels
Positive Flux Rate	rapid increases in reactor power during startup leading to undesired thermal transients	o compensated ion chamber o primary summation module o comparator	No	o While most effective in the 0 to 40% power range, limited protection exists in the normal load range
HTS Pump Bus Frequency	a loss of off-site power	o underfrequency relay o comparator	No	o one underfrequency relay on each of the HTS pump buses
Speed Mismatch	imbalance in the primary and intermediate speeds of the coolant pumps within a HTS loop	o primary pump tach o intermediate pump tach o primary absolute value module o comparator	Yes	o permissive provided for startup operation
Reactor Vessel Level	loss of primary sodium inventory	o reactor vessel sodium level detector o comparator	No	
Primary Overpower-Low Setpoint	increases in reactor power during startup	o compensated ion chamber o comparator	Yes	o permission permits ascent to power operations
Steam to Feedwater Flow Mismatch	imbalance in the superheater steam and feedwater flow within a HTS loop	o superheater steam mass flowmeter o feedwater mass flowmeter o primary absolute value module o comparator	Yes	o permissive provided for startup operation

(1) per subsystem per channel

TABLE 15-2 (Continued)

<u>Subsystem</u>	<u>Design Basis</u>	<u>Instrumentation</u> ⁽¹⁾	<u>Permissive</u>	<u>Remarks</u>
IHX Primary Outlet Temperature	unacceptable increases in the primary cold leg coolant temperature	o primary cold leg thermocouples o comparator	No	
Secondary Shutdown System				
Flux to Flow	increasing reactor power or decreasing reactor coolant flow over the normal load range	o fission counter o secondary summation module o primary loop flowmeter o comparator	No	
Modified Negative Flux Rate	rapid decreases in reactor power leading to undesired thermal transients	o fission counter o secondary summation module o comparator	No	o bypass provided at low power levels through choice of setpoint
Modified Positive Flux Rate	rapid increases in reactor power during startup leading to undesired thermal transients	o fission counter o secondary summation module o comparator	No	
High and Low Flow Mismatch	imbalance in the primary and intermediate coolant flows	o primary loop flowmeter o intermediate loop flowmeter o secondary summation module o Hi/Low Selector o comparator	Yes	o permissive provided for startup operations
Steam Drum Level	decreasing steam drum water level	o steam drum water level detector o comparator	Yes	o permissive provided for two loop operations
Evaporator Outlet Sodium Temperature	increasing intermediate sodium temperature	o intermediate cold leg thermocouple o comparator	No	
Sodium Water Reaction	sodium water reaction	o Flow switches o Hi/Low signal selector o comparator	No	
Startup Flux	increases in reactor power during startup	o fission counter (msv channel) o comparator	Yes	o must be bypassed during power operations
HTS Pump Bus Voltage	loss of off-site power	o undervoltage relay o comparator	No	

(1) per subsystem per channel

TABLE 15-3
REACTOR SHUTDOWN SYSTEM DESIGN BASIS EVENTS

<u>Event/Description</u>	<u>Enveloping Value</u>	<u>Classification</u>
A. Reactivity Insertions		
(1) Positive reactivity ramp insertions ⁽¹⁾		
(a) Control rod withdrawal at less than maximum design speed (9 ipm)	5¢/sec	Anticipated
(b) Postulated RSS design event to envelop reactivity ramp insertions	35¢/sec	Unlikely
(c) Postulated RSS design event to demonstrate adequate safety margin	2\$/sec	Extremely Unlikely
(2) Positive Reactivity Step Insertions ⁽²⁾		
(a) Step insertions to envelop small reactivity events	10¢	Anticipated
(b) Reactivity step insertions due to hydraulic imbalance or core radial movement	60¢	Unlikely
(3) Seismic events		
(a) OBE	30¢	Anticipated
(b) SSE	60¢	Extremely Unlikely
(4) Negative Reactivity Insertions		
(a) Control rod drop	0.5¢/sec	Anticipated
B. Primary and Intermediate Sodium Flow Disturbances		
(1) Electrical Fault		
(a) Coastdown of one primary pump		Anticipated

(1) Positive reactivity ramp insertions of less than 0.05¢/sec are not part of the RSS design basis (Manual Scram will terminate these events prior to automatic RSS action).

(2) This does not include steps due to seismic events.

TABLE 15-3 (Continued)

<u>Event/Description</u>	<u>Enveloping Value</u>	<u>Classification</u>
(b) Cooldown of one intermediate pump		Anticipated
(c) Pump cooldown in one HTS loop		Anticipated
(d) Pump cooldown in two HTS loops		Unlikely
(e) Pump cooldown in three HTS loops		Anticipated
(2) Mechanical Fault		
(a) Instantaneous primary pump seizure		Unlikely
(b) Instantaneous intermediate pump seizure		Unlikely
(3) Controller Fault - Pump Runout or Rundown		
(a) One primary pump		Anticipated
(b) One intermediate pump		Anticipated
(c) Three primary pumps		Anticipated
(d) Three intermediate pumps		Anticipated
(4) Loss of Sodium Inventory		
(a) Inadvertent dump of intermediate loop sodium		Unlikely
(b) Primary HTS leak		Extremely Unlikely
(c) Intermediate HTS leak		Extremely Unlikely
C. Steam Side Faults		
(1) Reduction or Loss of Feedwater Flow		
(a) One steam generator		Anticipated
(b) All steam generators		Anticipated

TABLE 15-3 (Continued)

<u>Event/Description</u>	<u>Enveloping Value</u>	<u>Classification</u>
(2) Inadvertent Closure of Steam Generator Module Isolation Valve		
(a) Evaporator module inlet		Anticipated
(b) Superheater module inlet		Anticipated
(c) Superheater module outlet		Anticipated
(3) Inadvertent Isolation and Blowdown of Steam Generator Module		
(a) Both evaporators and superheater modules		Anticipated
(b) Evaporator module		Anticipated
(c) Superheater module		Anticipated
(d) Rupture disk failure		Unlikely
(4) Loss of One Recirculation Pump		Anticipated
(5) Loss of Main Condenser		Anticipated
(6) Inadvertent Valve Opening		
(a) Evaporator outlet safety/power relief valve		Anticipated
(b) Superheater outlet safety/power relief valve		Anticipated
(c) SGAHRS steam drum vent valve		Anticipated
(d) Evaporator inlet dump valve		Anticipated
(e) Turbine bypass valve		Anticipated
(7) Design Basis Sodium Water Reaction		Unlikely
(8) Feedwater Line Rupture		
(a) Between steam drum and isolation valve		Extremely Unlikely

TABLE 15-3 (Continued)

<u>Event/Description</u>	<u>Enveloping Value</u>	<u>Classification</u>
(b) Main incoming header		Extremely Unlikely
(c) Rupture of steam drum blowdown line		Extremely Unlikely
(9) Steam Line Rupture		
(a) Saturated steam line		Extremely Unlikely
(b) Main steam line		Extremely Unlikely
(c) Between superheater outlet and isolation valve		Extremely Unlikely
(d) Between superheater outlet isolation valve and main steam line		Extremely Unlikely
(10) Recirculation Line Break		
(a) Between steam drum and recirculation pump inlet		Extremely Unlikely
(b) Between evaporator outlet and drum inlet		Extremely Unlikely

TABLE 15-4
SPECIFIC DESIGN REQUIREMENTS FOR THE CRBRP REACTOR SHUTDOWN SYSTEMS⁽⁵⁾

Initial Plant Condition ⁽⁴⁾	Damage Severity Limits ^(4,6)	
	Primary System Only Functioning	Secondary System Only Functioning
Normal Operation ⁽²⁾	Not Applicable	Not Applicable
Anticipated Faults	Operational Incident	Minor Incident ⁽¹⁾
Unlikely Faults	Minor Incident	Major Incident
Extremely Unlikely Faults	Major Incident	Not a Design Basis ⁽³⁾

- (1) Failure of the Primary System to scram when required for an anticipated fault is defined as an extremely unlikely event (faulted condition). However, the damage severity limit for the Secondary Shutdown System is conservatively specified to assure fuel pin integrity even for the concurrent anticipated fault and failure of the Primary Shutdown System.
- (2) No action required by Plant Protection System during normal operation.
- (3) Combined probability of two independent failures (extremely unlikely fault and failure of Primary Control Rod System) is exceedingly low and not appropriate as a design basis. However, as an exception, upon concurrent loss of off-site power, a Safe Shutdown Earthquake with a consequent step in reactivity insertion (60¢) and failure of the Primary Control Rod System, the Secondary Shutdown System shall be capable of shutting down the reactor without exceeding Major Incident limits.
- (4) See Table 15.1-5 for definition of terms.
- (5) Reference - Table 2.2-3, Overall Plant Design Description (OPDD-10).
- (6) Limits apply to Shutdown System action with or without the highest worth rod stuck.

Shutdown Earthquake. The design requirements of Table 15-4 specify the allowable consequences of design basis events in terms of Damage Severity Limits. Definitions of the event classifications and Damage Severity Limits are provided in Table 15-5. Each Damage Severity Limit specifies in general terms 1) allowable fuel damage, 2) allowable plant damage, and 3) allowable radioactivity releases.

15.2 Evolution of Design

15.2.1 Conceptual Design

Work on the Plant Protection System began in 1973. Key decisions made included:

- A. Coincidence logic: Local and general coincidence were chosen for the primary and secondary shutdown systems, respectively. Local coincidence provided the best defense against false scrams and was thus used in the system with the largest number of signals. General coincidence is less effective against false scrams, but was judged to be adequate for the secondary shutdown system where the number of subsystems was limited to sixteen.
- B. Channel redundancy: Three redundant instrument channels were chosen for the primary and secondary shutdown systems. Three channels were shown to meet the reliability and availability guidelines without resulting in excessive costs or system complexity.
- C. Protective Subsystems: Functional configurations were chosen for the Primary and Secondary Shutdown Systems. Many of the subsystems were based on the configurations used in the Fast Flux Test Facility (FFTF). These protective subsystems are listed and described in Section 2 of SDD 99 [1]. Subsystems were chosen so as to respond to a preliminary list of design basis events. Generally, the fast responding variables were used in the Primary Shutdown System to provide the faster response. Scoping transient analysis was

TABLE 15-5
EVENT CLASSIFICATION AND CLADDING DAMAGE SEVERITY LIMITS

<u>ASME Code Section III (Article NR-3113)</u>	<u>Event Classification</u>	<u>Severity Level</u>
	<u>RDТ Standard C16-1</u>	<u>RDТ Standard C16-1</u>
Normal:	Normal Operation:	No Damage:
Any condition of system startup design range operations, hot standby, or shutdown other than an upset, emergency, faulted or testing conditions.	Normal operation includes steady power operations and those departures from steady operation which are expected frequently or regularly in the course of power operations, refueling, maintenance, or maneuvering of the plant.	No damage is defined as 1) no significant loss of effective fuel lifetime; 2) accommodations within the fuel and plant operating margins without requiring automatic or manual protective action; and 3) no planned release of radioactivity.
Upset:	Anticipated Faulted:	Operational Incident:
Any abnormal incident not causing a forced outage or causing a forced outage for which the corrective action does not include any repair of mechanical damage	An off-normal condition which individually may be expected to occur once or more during the plant lifetime.	An operational incident is defined as an occurrence which results in 1) no reduction of effective fuel lifetime below the design values; 2) accommodation with, at most, a reactor trip that assures the plant will be capable of returning to operation after corrective action to clear the trip cause; and/or 3) plant radioactivity releases that may approach the 10CFR20 guidelines.
Emergency:	Unlikely Faulted:	Minor Incident:
Infrequent incident requiring shutdown for correction of the condition for repair of damage in the system. No loss of structural integrity	An off-normal condition which individually is not expected to occur during the plant lifetime; however, when integrated over all plant components, event in this category may be expected to occur a number of times.	A minor incident is defined as an occurrence which results in 1) a general reduction in the fuel burnup capability and, at most, a small fraction of fuel rod cladding failures; 2) sufficient plant or fuel rod damage that could preclude resumption of operation for a considerable time and/or 3) plant radioactivity releases that may exceed 10CFR20 guidelines, but does not result in interruption or restriction of public use of areas beyond the exclusion boundary.
Faulted:	Extremely Unlikely Faulted:	Major Incident:
Postulated event and consequences where integrity and operability may be impaired to the extent that considerations of public health and safety are involved.	An off-normal condition of such extremely low probability that no events in this category are expected to occur during the plant lifetime, but which nevertheless represents extreme or limiting cases of failures which are identified as design bases.	A major incident is defined as an occurrence which results in 1) substantial fuel and/or cladding melting or distortion in individual fuel rods, but the configuration remains coolable; 2) plant damage that may preclude resumption of plant operations, but no loss of safety functions necessary to cope with the occurrence; and/or 3) radioactivity release that may exceed the 10CFR20 guidelines but are well within the 10CFR100 guidelines.

(1) Reference: Table 1.1.2-1, PSAR.

performed to determine if subsystem functional arrangements would respond to design basis events. The results of these evaluations are documented in Reference 2.

15.2.2 Preliminary Design

Analysis continued to confirm that the subsystems function as required. Results of these analysis are documented in References 3 and 4.

The decision was made to use analog hardware instead of the digital systems which were beginning to appear at the time. This decision was based on licensing considerations. Less problems were envisioned for a system similar in design to existing systems (i.e., FFTF) than a system based on a new unproven design approach.

A great degree of emphasis was placed on Plant Protection System reliability. A program was initiated to obtain quantitative reliability data on hardware performance. The plan called for testing of equipment under simulated operating conditions. Originally, five years of testing was envisioned. To support this plan and the original site construction date, an early equipment procurement date was required. This early procurement date was readily attained since the equipment was quite similar to that built for the FFTF; a contract was negotiated with the Weston Controls Division of Schlumberger, Ltd. who had designed and built the FFTF equipment. Initial procurement was started June, 1976.

The design of the PPS was based on the FFTF meaning that there were no significant changes early in the design. The evolution of the design was quite smooth and it is documented in Reference 2 and its progression is documented in References 3 and 4.

15.2.3 Final Design

Following manufacture and delivery of the equipment to the Westinghouse Waltz Mill Site, an extensive reliability test was undertaken. Results from this test [5] were used to confirm reliability objectives. More importantly, where

test results showed that deficiencies existed, design changes were made. Examples are power supply upgraded to correct diode failures and primary scram breakers modified.

Analysis continued to verify the adequacy of the design. Emphasis was placed on establishing interfaces with components in order to document the adequacy of PPS trip settings.

15.3 Key Problems and Resolution

During the design of the Plant Protection System, a number of issues were identified that required a significant engineering effort to address. A number of these issues are discussed in the following paragraphs.

15.3.1 Control/Protection Interface

In order for the plant to operate with a minimum of spurious scrams, the PPS settings must be chosen so as not to interfere with the normal operating range of the plant. Verification that a margin exists between the parameter values during normal operation and those that could result in PPS action involves several steps.

- A. Transient analysis of the control system response to normal events is performed.
- B. The normal transient zone is increased to account for control system hardware errors and time delays.
- C. The PPS operating margin is defined. The operating margin is the difference between the PPS nominal performance and the parameter values that may result in a trip.

An example of this approach can be found in the High and Low Flow Mismatch Subsystems. These subsystems calculate the difference between the primary and intermediate flows and initiate a trip if this difference exceeds predefined limits.

Figure 15-1 plots four curves of interest. The low control (lo con) curve shows the relationship between primary and intermediate flows during an unloading transient from 100% to 40% reactor power. The high control (hi con) curve shows the relationship of the flows during a loading transient from 40% to 100% reactor power. These curves were determined from a computer transient analysis program used for control system studies. The trajectories represent the nominal control response (i.e., no uncertainties due to hardware implementation).

The errors in the hardware must also be considered. In the case of the Flow Mismatch Subsystems, the hardware that can introduce performance uncertainties include the flowmeters, buffers, median-select modules, and controller errors. Margin is provided between the nominal control system response and the low and high control/protection interface lines in order to account for these errors. The minimum margin provided is slightly over 1000 gpm or approximately 4% intermediate flow.

Figure 15-2 plots the worst case, nominal, and operations performance predictions for the Flow Mismatch Subsystems together with the control/protection interface values. Several points should be noted:

- A. First, the part load profile does not give exactly 40% primary and 40% intermediate flow at 40% reactor power. Instead, the values used in developing PPS settings give a primary flow of $14375/35890 = 40.05\%$ and an intermediate flow of $10765/25493 = 42.23\%$. The control system performance of Figure 15-1 is based on the assumption that primary and intermediate flow are 40% at 40% reactor power. Therefore, the interface shown in Figure 15-1 must be adjusted by 2.23% to account for the different assumptions.
- B. Second, the control and protection systems share the permanent magnet flowmeter signals. It is important to ensure that the flowmeter errors are not counted twice. Therefore, the flowmeters uncertainties are included in the margin between the nominal control

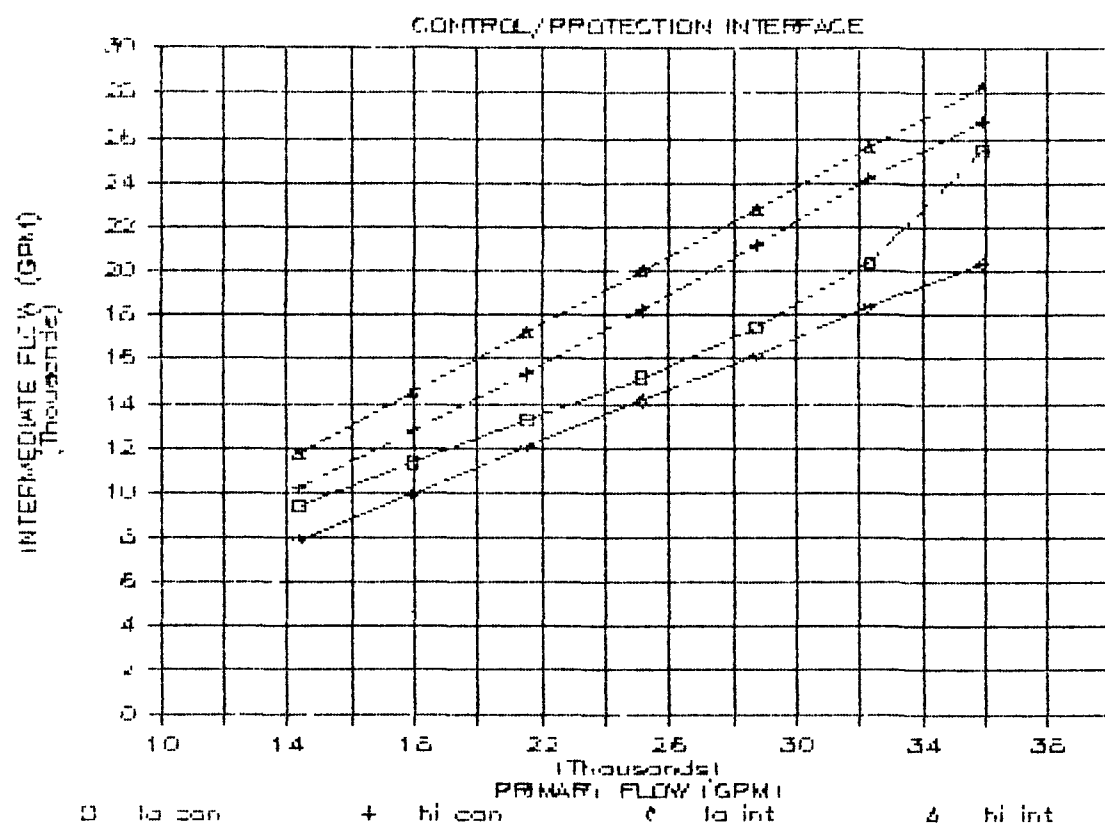


Figure 15-1 Control / Protection Interface

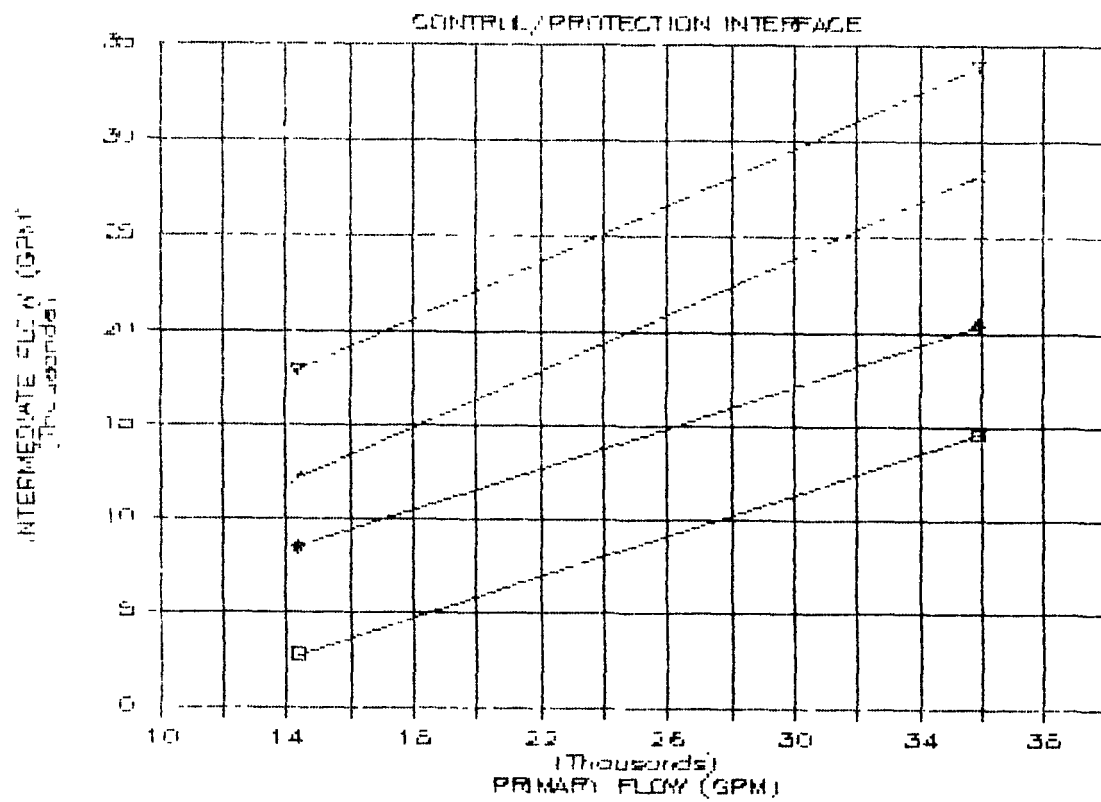


Figure 15-2 Control / Protection Interface

system response and the control/protection interface, and not in the margin between nominal PPS response and the control/protection interface.

The margin between the worst case and nominal PPS performance is significantly greater than the margin between the nominal and operations performance. One reason for this is that the safety margin includes the effects of flowmeter errors, whereas, as explained above, the operations margin does not. In addition, the safety margin is based on an algebraic combination of errors, while the operations margin is based on a statistical combination. This provides the conservatism necessary to ensure that the safety limits are never exceeded. It does mean, however, that there is a greater chance of initiating a spurious trip. It has not been possible to assign a probability to the chance of exceeding the operations limits due to the lack of statistical hardware performance data. However, the probability of getting a spurious trip if variables just exceed the operations interface is judged to be acceptable. The nominal trip setting, by definition, is the center of the trip distribution so that, as the variables approach the nominal trip setting, the probability of a trip approaches 50%. All of the protective subsystems are analyzed and documented in Reference 1.

The control/protection interfaces developed for CRBRP are documented in Reference 6. At the end of the Project, a number of these interfaces were still on HOLD. Work remained to verify that the control and the protection systems could successfully satisfy the defined interfaces.

15.3.2 Control/Protection Interaction

Studies of the interaction between the control and protection systems were performed to ensure that Criterion 24 of 10CFR50 [7], Appendix A, "General Design Criteria," was satisfied.

10CFR50, Appendix A, Criterion 24 states:

"The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or

removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnections of the protection and control systems shall be limited so as to assure that safety is not significantly impaired."

This requirement is included in the CRBRP Design Basis in a slightly modified form as Criterion 22. IEEE Standard 279, 1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," provides an interpretation of this requirement in Criterion 4.7.3:

"Where a single random failure can cause a control system action that results in a generating station condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining redundant protection channels shall be capable of providing the protective action even when degraded by a second random failure."

CRBRP Control and Reactor Shutdown Systems use common instrumentation as listed in Table 15-6. Sharing of instrumentation both minimizes the complexity of instrumentation and simultaneously enhances the quality of the control systems by utilizing safety grade instruments to perform important plant functions. The Reactor Shutdown System contains two sets of three redundant instrument channels in the Primary and Secondary Shutdown Systems. Complete separation between control and protection functions would require a third set of instruments dedicated to the control function. Although the third set of control instruments would not necessarily have to include redundancy, redundancy does increase plant safety by reducing the number of control system induced excursions caused by single failures of instrument channels. The use of common sensors in the Control and Reactor Shutdown Systems permits the inclusion of redundant design features in the Control System which also provide compliance with the requirements of IEEE 279 Criterion 4.7.3.

Table 15-7 summarizes the consequences of a single failure of instrumentation common to Control and Protection Systems while one of the redundant channels

TABLE 15-6
INSTRUMENTATION COMMON TO THE CONTROL AND PROTECTION SYSTEMS

<u>Sensor</u>	<u>Shutdown System Function</u>	<u>Control System Function</u>
1. Primary Flux	Primary Shutdown System o High Flux o Flux to Pressure o Flux Rates	Control of Nuclear Power (Reactor Controller)
2. Secondary Flux	Secondary Shutdown System o Flux to Flow o Modified Flux Rates	Overpower and P/F Rod Blocks (Reactor Controller)
3. Primary Sodium Flow	Secondary Shutdown System o Primary to Intermediate Flow Mismatch o Flux to Flow	Control of Primary Sodium Pumps (Primary Flow Controller)
4. Intermediate Sodium Flow	Secondary Shutdown System o Primary to Intermediate Flow Mismatch	Control of Intermediate Sodium Pumps (Intermediate Flow Controller)
5. Primary Pump Speed	Primary Shutdown System o Primary to Intermediate Speed Mismatch	Provide Bumpless Transfer When Switching From Speed to Flow Control (Primary Flow Controller)
6. Superheater Steam Flow	Primary Shutdown System o Steam to Feedwater Flow Mismatch	Control of Feedwater Flow (Steam Drum Level Controller) Indicate Plant Load (Supervisory Control System)
7. Feedwater Flow	Primary Shutdown System o Steam to Feedwater Flow Mismatch	Control of Feedwater Flow (Steam Drum Level Controller)
8. Steam Drum Level	Secondary Shutdown System o Steam Drum Level	Control of Feedwater Flow (Steam Drum Level Controller)

TABLE 15-7
PROTECTION/CONTROL INTERACTION

<u>Sensor</u>	<u>Channels</u>				<u>Control System Response</u>	<u>Protection System Response</u>
	<u>A⁽¹⁾</u>	<u>B⁽²⁾</u>	<u>C</u>	<u>Median</u>		
1. Primary Flux	H	L	N	N	Normal	Not required
	L	H	N	N	Normal	Not required
	H	H	N	H	Flux Control: Decrease in reactor power	Not required
					Temp. Control: Initial decrease in reactor power followed by partial/total recovery	Not required
	L	L	N	L	Flux Control: Increase in reactor power limited by rod block circuits (secondary flux)	Not required
					Temp. Control: Increase in reactor power limited by rod block circuits or temperature feedback loop	Not required
2. Secondary Flux	H	L	N	N	Normal	Not required
	L	H	N	N	Normal	Not required
	H	H	N	H	Spurious actuation of rod block circuits	Not required
	L	L	N	L	Failure of rod block circuits	Not required
3. Primary Sodium Flow	H	L	N	N	Normal	Not required
	L	H	N	N	Normal	Not required
	L	L	N	L	Speed/Manual Flow Control: Normal	Not required
					Auto Flow Control: Increase flow in one primary loop	Not required
	H	H	N	H	Speed/Manual Flow Control: Normal	Not required
					Auto Flow Control: Decrease flow in one primary loop	PSS responds upon demand
4. Intermediate Sodium Flow	H	L	N	N	Normal	Not required
	L	H	N	N	Normal	Not required
	L	L	N	L	Speed/Manual Flow Control: Normal	Not required
					Auto Flow Control: Increase flow in one intermediate loop	Not required
	H	H	N	H	Speed/Manual Flow Control: Normal	Not required
					Auto Flow Control: Decrease flow in one intermediate loop	PSS responds upon demand

(1) Channel A is assumed to be undergoing test. Channel is tripped during test.

(2) Channel B is assumed to experience first failure. Signal deviation is assumed not to cause a channel trip.

TABLE 15-7 (Continued)

<u>Sensor</u>	<u>Channels</u>				<u>Control System Response</u>	<u>Protection System Response</u>
	<u>A</u> ⁽¹⁾	<u>B</u> ⁽²⁾	<u>C</u>	<u>Median</u>		
5. Primary Pump Speed	H	L	N	N	Normal	Not required
	L	H	N	N	Normal	Not required
	H	H	N	H	Speed/Flow Control: Normal	Not required
					Switching from Speed to Flow Control: Increase in primary sodium flow in one loop	(3)
	L	L	N	L	Speed/Flow Control: Normal	Not required
					Switching from Speed to Flow Control: Decrease in primary sodium flow in one loop	(3)
6. Superheater Steam Flow	H	L	N	N	Normal	Not required
	L	H	N	N	Normal	Not required
	H	H	N	H	Supervisory Control: Total steam flow indicates load greater than actual. Reactor power and sodium flows increase to meet apparent steam demand. Temperatures and pressures increase. Balance of control system responds to keep plant within acceptable limits.	Not required
	L	L	N	L	Supervisory Control: Total steam flow indicates load greater than actual. Reactor power and sodium flows decrease to meet apparent steam demand. Temperatures and pressures initially decrease. Balance of control system responds to keep plant within acceptable limits.	Not required

(3) Switching from speed to flow control is normally accomplished during pump startup before primary or secondary control rods are latched. Consequently, no protective action is required.

is undergoing testing. One of the two Shutdown Systems could be disabled by this single failure and application of IEEE 279 Criterion 4.7.3. However, the second Shutdown System, using independent and diverse instrumentation, is always available to detect the accident. Consequently, together, the Primary and Secondary Shutdown Systems meet Criterion 4.7.3 of IEEE 279 for all single failures. Since a single failure of common instrumentation can only initiate an accident during testing of a redundant channel and since testing of common instruments occurs only during a small period of plant operations, individually, the Primary and Secondary Shutdown Systems meet Criterion 4.7.3 for most, but not all single failures. Failure of the Control System median select circuits does not affect this conclusion.

The control/protection interaction studies are documented in Reference 8.

15.3.3 PPS Worst Case Performance

A large amount of engineering analysis was used to estimate the worst case performance analysis of the PPS. This analysis centered on identifying trip setpoints that, when the hardware uncertainties were accounted for, would protect the plant and public without impacting the economic operations of the plant. A conservative approach was used to estimate worst case performance. Hardware uncertainties were identified and combined algebraically. This resulted in performance estimates that could be defended during licensing activities despite the lack of firm performance data on the PPS instrumentation errors.

Once the worst case trip settings were identified, it was necessary to verify that they met the PPS design requirements. The PPS design requirements are summarized in Table 15-4. These limits are qualitative and provide no specific practical guidance for PPS designers.

Ideally, the Plant Duty Cycle should provide quantitative guidelines describing the required PPS performance. Unfortunately, the number of Duty Cycle events was minimized in order to reduce the number of thermal analyses. Therefore, the Duty Cycle events contain little information concerning the required PPS settings.

In order to verify adequate PPS performance, it was necessary to establish interfaces with individual component systems. These essential interfaces are documented in References 9 and 10. The establishment of these interfaces proved to be an inefficient, time consuming activity. The reason for this is that the Duty Cycle provided the basis for component design. Any event not obviously enveloped by the Duty Cycle required additional analysis by the component designers to confirm that the event fell within the envelopes.

The difficulties associated with verifying adequate PPS settings under these conditions can be shown by examining the following:

The U-2b transient as described in OPDD-10 [11] is the only overpower event from 100% initial conditions included in the CRBRP Duty Cycle. The U-2b is a single rod withdrawal at 1/2%/second, increasing to a maximum 115% power. The event was scrammed after five minutes.

Based on the U-2b, the Primary Reactor Shutdown System overpower trip setting was set at 115%. The Secondary Shutdown System, since it is permitted a one level shift in damage limits, initially had an overpower trip setting of 118%. However, in trying to confirm the adequacy of this setting, it was found that the Duty Cycle had no event in the emergency category that could envelop an overpower event with Secondary Shutdown System action. After extensive discussions, the Secondary Shutdown was modified by ECP W1832, dated February 1983, to trip at 115% power, rather than modifying the Duty Cycle to be consistent with the design requirements.

A second example involves the protection provided for overpower events at part load conditions. The U-2b has a flux to flow ratio at trip of 1.15. Applying this ratio at 40% initial conditions gives an overpower trip at 46%. This value is overly restrictive and impractical to implement physically. Rather, an overpower trip of 55% representing an increase in power of 15% is more reasonable. Although, the 55% trip setting was eventually shown to fall within the U-2b envelope, significant effort was spent resolving the issue.

An approach suggested for future Projects would require baselining PPS performance envelopes early in the design process. Although not possible for CRBRP, this should be possible for future projects if their instrumentation systems are similar to FFTF or CRBRP. Once the settings are baselined, the Design Duty Cycle can be developed with complete knowledge of the PPS response. To verify the adequacy of the PPS design would then only require the PPS designer to show that the baselined performance envelopes are met. Any changes to the baselined performance envelopes required due to design evolution could be handled via the normal engineering change procedure.

15.3.4 Permanent Magnet Flowmeter Signal Considerations

The initial flowmeter requirements were baselined by ECP W1080 R1 dated August 1981. Operating experience at FFTF indicated that the permanent magnet flowmeter requirement was not met due to a sizeable noise component riding on top of the signal. Studies were performed to try to determine the magnitude and frequency content of the noise signal and to determine ways to reduce the noise.

The first step was to obtain data on the expected noise content. This data was obtained from the Sodium Pump Tests performed at the Energy Technology Engineering Center (ETEC). Data from a typical flowmeter signal were linearized. Although this linearization somewhat distorted the signals' frequency content, it still provided a good approximation to the low frequency characteristics of the signal.

The linearized data and a first order lag time constant were modeled on an analog computer. Time constants over the range of 0.5 to 5 seconds were analyzed. Figure 15-3 plots the results. The following conclusions were reached for 100 percent flow:

- 1/4 sec filter: 3.1 percent
- 1/2 sec filter: 2.9 percent
- 1 sec filter: 2.6 percent
- 2 sec filter: 2.2 percent
- 5 sec filter: 1.5 percent

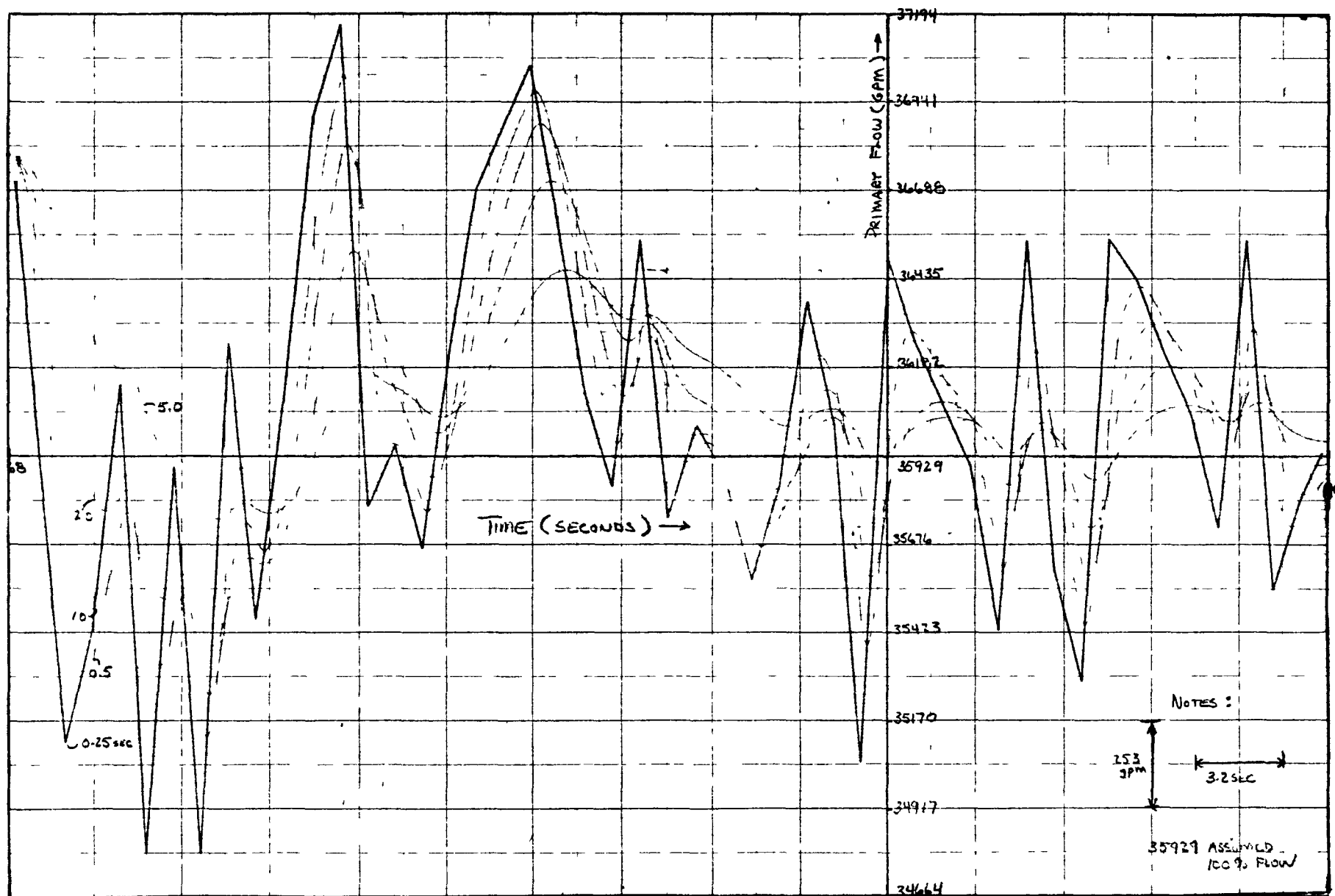


Figure 15-3 Effect Of PM Flowmeter Time Constant On Signal Noise

These results are for one specific time and do not necessarily represent the worst case points.

The CRBRP Control System Design permitted a maximum delay of 0.5 second. Additional filtering would be included in the PPS hardware.

Although filtering did reduce the noise content, it also slows down the response of the PPS to loss of flow accidents. The Secondary Shutdown System responding to anticipated loss of flow accidents is required to limit conditions to the minor incident category. It was concluded that a trip at or above 80 percent reactor flow within 10 seconds will meet the minor incident accident level. This is within the capability of the flux to flow subsystem with a 5 second filter time constant [12].

One concern was that despite this long time constant, significant low frequency flowmeter noise would still exist. If this were true, additional filtering would not be effective and the 5 percent flowmeter uncertainty, which contains 0.5 percent margin for noise, would need to be increased to account for the larger than expected noise content.

A request was made to FFTF to provide CRBRP with a frequency analysis of its 16 inch permanent magnet flowmeter during normal operations. At the time the Project was terminated, this information had not been received.

15.3.5 Reactor Power Measurements

In the design of the RSS, the assumption was made that, after calibration, the variable reactor flux is directly proportional to reactor power. Observations at FFTF, however, have shown that this linear relationship is not strictly true due to an influence by primary sodium temperatures. The cause of this influence was unclear, but thought to be due to the change in absorption characteristics with temperature of the sodium above the core.

The effect of this temperature influence on CRBRP was unknown. Based on FFTF data, the change in flux could result in an underestimation in reactor power of up to 4 percent. Since readings would be lower than actual power, this was in the unsafe direction.

The Project was terminated before this issue was resolved. The steps that were planned to resolve this issue were:

- A. Obtain a good understanding of the physical mechanism behind this phenomena based on experimentation at FFTF.
- B. Estimate the effect on CRBRP.
- C. Determine if the PPS trip margins could absorb the additional measurement uncertainty.
- D. Examine instrumentation that would provide correcting circuitry.

Concerning the last two activities, the following points should be noted. First, the PPS margins were not sufficient to include an additional 4 percent error term in flux measurement without adjusting worst case trip points. Second, the use of a core exit temperature signal as a correcting term was not necessarily a sure solution due to the different system dynamic delays.

15.3.6 Reactor Flux Tilt

Analysis of overpower events determined that there was a significant difference between the withdrawal of a single control rod and a bank of control rods. This difference was due to the local power peaking effect in the area surrounding a rod out of bank. Various detection means were examined. The use of the flux sensors to detect flux tilts was not possible because the detectors responded principally to an average flux signal. A PPS detection scheme based on rod position was then examined but dropped due to the expense of providing independent and redundant measurements of rod location. A suitable PPS grade detection scheme was not found. Instead, the rod control system was upgraded to include interlocks which reduced the

probability of a single rod withdrawal accident to a level that existing PPS trips could limit the consequences to appropriate levels. This change was made by ECP W1663 dated May 1981. This ECP was against the Plant Control System (SDD-90).

Since providing a PPS system to detect an accidental single rod withdrawal accident is difficult, it is suggested that future Projects give appropriate consideration to control rod drive systems that minimize both the probability of occurrence and the consequences of unsymmetric control rod alignments.

15.4 Final System Status

The System 99 Independent Design Review was held April 5 and 6, 1983 at Waltz Mill. The Design Review went well, nonetheless, there were a number of findings that had not yet been resolved at the time of Project termination.

One area where additional work was required was the verification of the adequacy of PPS trip settings. A draft report [12] was issued as part of the design review package. This report documented the basis for the PPS trip settings. However, all the interfaces needed to verify the adequacy of these settings had not been baselined. Key interfaces not yet resolved involved System 31 and System 90.

A second area where additional work was required was the analysis of the reliability test data. A draft Reliability Design Support Document [5] was issued as part of the design review package. This report contained both raw data and a portion of the analysis required by the reliability tests. A large amount of test data had been taken during the reliability test period. Although a number of findings had been incorporated into the hardware design, information concerning the performance was just beginning to be evaluated. This information would be of value in showing that the hardware uncertainties (e.g., setpoint drift) during normal systems operation fell significantly within the values assumed in the establishment of the PPS trip settings.

During the evolution of the PPS design from January 1974 through September 1983, 83 revisions resulted from over 85 ECPs that covered system modifications, interfaces, performance requirements and documentation. Further information on the ECPs is available in SDD 99 [1].

15.5 References

1. Plant Protection System Design Description, SDD-99, Rev. 83, October 1983.
2. Deleted
3. Deleted
4. Deleted
5. Reliability Design Support Document (RDSO) for the CRBRP Plant Protection System, System 99, 99 RDSO-1.
6. System 99/90 Operations Interface, ICD CW50679, Rev. 0, May, 1982.
7. Title 10, Code of Federal Regulations, Part 50, "Licensing of Production and Utilization Facilities," Superintendent of Documents U.S. Government Printing Office, Washington, D.C., 20402.
8. Control/Protection Interaction Evaluation, September 27, 1982 [99CPI-1].
9. Reactor Shutdown System Performance: Interface Between Systems 31 and 99, ICD CW50664.
10. Reactor Shutdown System Setpoints for Steam Side Subsystems: System 99 and 53 Interface, ICD CW50663, Rev. 0, June 26, 1980.
11. Clinch River Breeder Reactor Plant (OPDD-10), Overall Plant Design Description, Rev. 146, August 1, 1983.
12. Plant Protection System Performance Analysis Report, Draft 0, WARD-D-0329, September 1, 1982 [99PAR-1].

15.6 Major ECPs Affecting PPS Design

1. ECP W-1080, "Performance Requirements for PHTS Magnetic Flowmeters."
2. ECP W-1663, "Block Row 7 PCRDM for Rod Out of Alignment."
3. ECP W-1832, "Flux to Flow Modification."

APPENDIX A - DETAILED DESCRIPTION OF THE THERMAL STRIPING METHODOLOGY

To obtain fatigue damage estimates for the fuel blanket, RRS, and primary control assemblies, it was necessary to develop a thermal striping methodology. This methodology consisted of five steps:

- Step 1 Inputting the material properties into the "RAINFAT" computer code.
- Step 2 Performing a parametric sensitivity study, using the "RAINFAT" computer code to see how elastic slope, film coefficients, temperature and time effect damage.
- Step 3 Selecting temperature striping design cases for the fuel, blanket, removable radial shield, and primary control assemblies.
- Step 4 Deriving a design curve for the fuel and blanket assemblies to aid in core orificing.
- Step 5 Tabulating the final orificing results.

A.1.0 "RAINFAT" COMPUTER CODE (STEP 1)

A.1.1 Input Parameters

The following information is input to "RAINFAT";

- o The fatigue curve coefficients and exponents
- o The cycle stress-strain properties
- o The normalized temperature signal taken from the outlet nozzle striping tests in water (load history)
- o The temperature gradients between adjacent core assemblies
- o The residence time of the assembly in the reactor

The fatigue curve was generated from 316 SS data collected at different temperature ranges. It was decided that one curve at a middle temperature range of 538°C to 566°C could be used because outlet nozzle temperatures are at the average below this temperature. For this temperature range, four curves were generated, two for in-air tests (both nominal and -2σ limits) and two for in-sodium tests (both nominal and -2σ limits). These curves are shown in Figures A-1 and A-2. The curves were statistically derived from data. To input the thermal stress affect, the analogy of a flat plate restrained at the edges and subjected to uniform temperature was used. A thermal expansion of 11.23×10^{-6} in/in°F and a Poisson ratio of 0.3009 were used in the calculations. The values used were obtained from Reference 1. The cyclic stress-strain properties for 316 SS were obtained from References 1, 2, and 3. The properties used were 22,530,000 lb/in² for the modulus of elasticity, 58,900 lb/in² for the cyclic yield strength, and 0.127 for the strain hardening exponent. The temperature signals (or load history) were obtained from the computer codes TAPA and STRIPE (Reference 4 and Reference 5). These codes convert fluid striping to surface striping and normalize the temperature amplitudes around the structural mean temperature. Figures A-3, A-4, and A-5 illustrate the conversion of the fluid striping signal into a surface striping signal. The temperature gradients between assemblies were obtained from Reference 6. Reference 6 explains how the temperature gradients were found. Only plant expected operating conditions were used in the final striping analysis. The length of time that the assemblies were in the reactor varies from 328 days for the first core assemblies to 878 days for both first and second core assemblies combined. The RRS assemblies were considered to be in the reactor for 30 years. Calculations for each of the above inputs can be found in Reference 7.

A.2.0 RAINFLOW Procedure

The following steps are done by the RAINFLOW computer code.

- o Find the maximum amplitude in the normalized temperature history.
- o Rearrange the history to start at the maximum amplitude.

FATIGUE OF SA 316 SS IN AIR AT 538 TO 566 °C

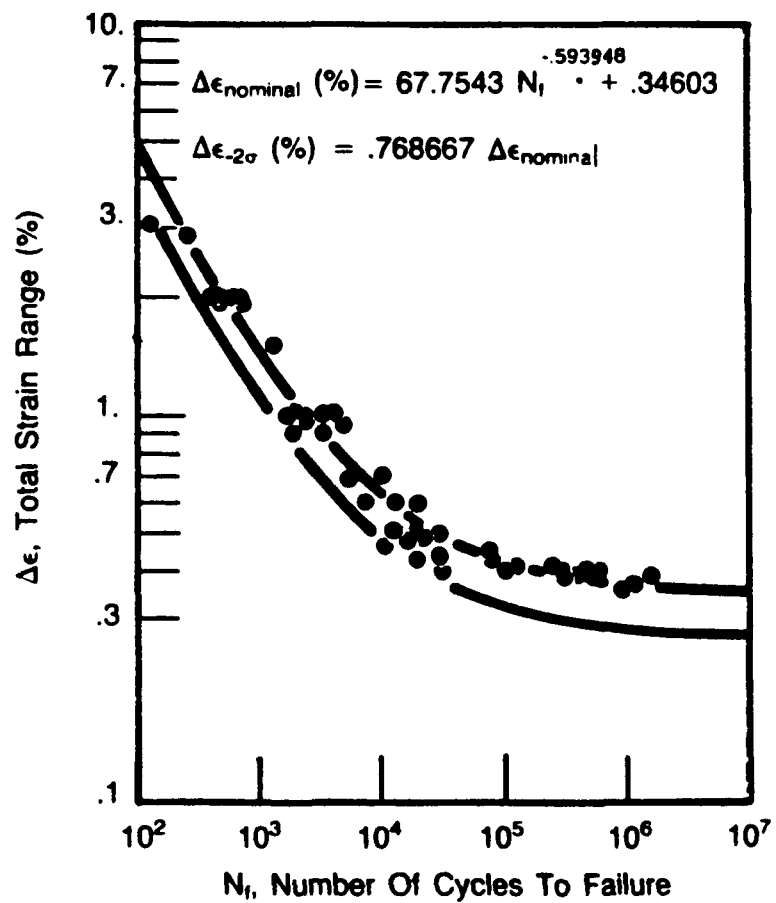


Figure A-1

**FATIGUE OF SA 316 SS AT 538 TO 566°C,
IN SODIUM DATA VERSUS IN SODIUM CORRELATION
WITH IN AIR CORRELATION ASYMPTOTE**

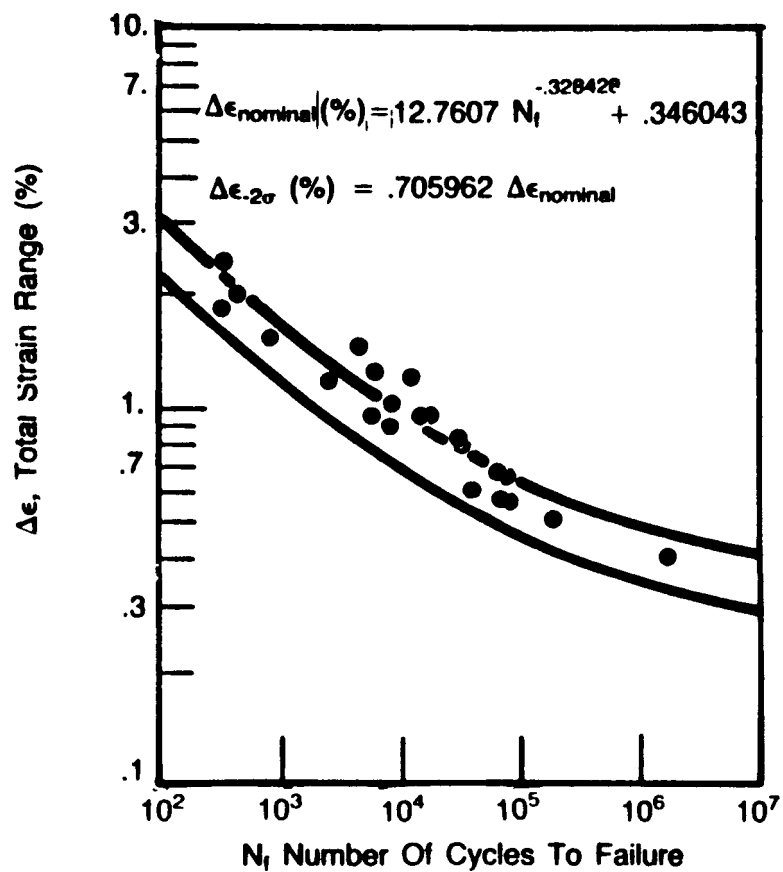
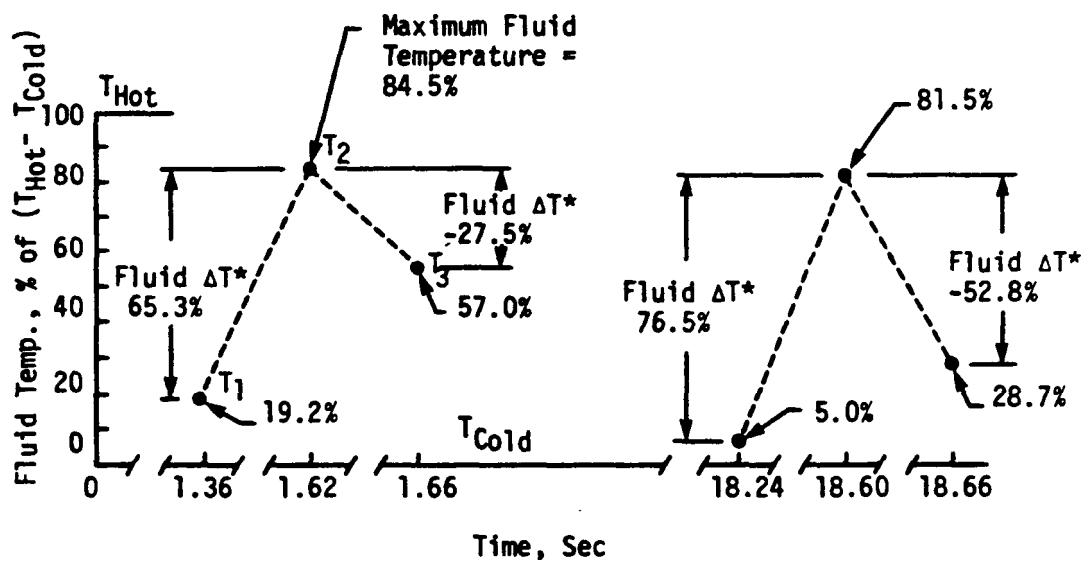


Figure A-2

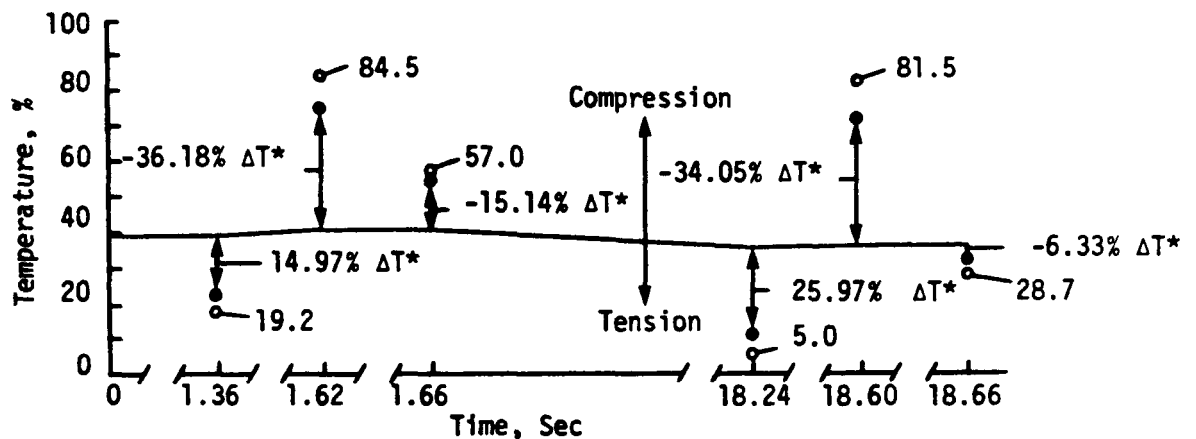


* Fluid $\Delta T = T_{I+1} - T_I$, indicating a positive or negative temperature slope

The fluid temperatures (19.2, 84.5, 57, 5.0, 81.5, 28.7) are input to the "TAPA" program which calculates the thermal effects of the fluid temperature fluctuations on a structural model (illustrated in Figure 2).

Figure A-3

ILLUSTRATION OF FLUID TEMPERATURE HISTORY AS INPUT TO "TAPA" (AS BOUNDARY TEMPERATURES)



Legend:

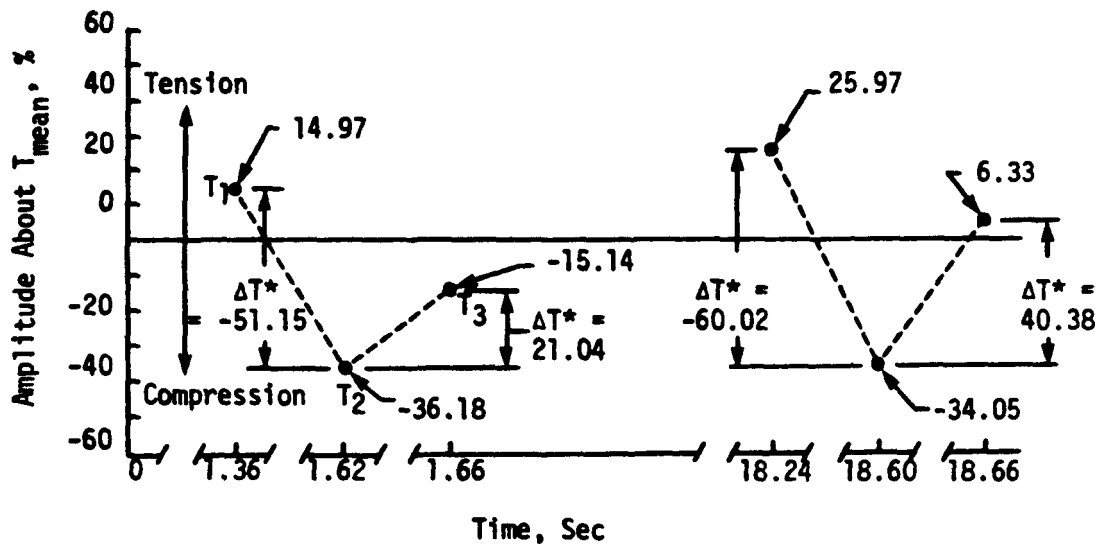
- Instantaneous thermal amplitude above or below T_{mean} , ΔT^*
- Mean structure temperature, T_{mean}
- Surface temperature of structure, $T_{surface}$
- Boundary temperature (fluid), T_{bound}

* $\Delta T = T_{mean} - T_{surface} =$ instantaneous thermal amplitude above or below T_{mean}

The instantaneous structural temperature ΔT 's (14.97, -36.18, -15.14, 25.97, -34.05, 6.33) are input to the "STRIPE" program which calculates total structural thermal striping amplitudes (see Figure 3) and performs statistical analyses on the amplitudes.

Figure A-4

ILLUSTRATION OF INSTANTANEOUS STRUCTURAL TEMPERATURE AMPLITUDE (ΔT^*) HISTORY AS INPUT TO "STRIPE"



Legend:

- Mean structure temperature, T_{mean}
- Instantaneous thermal amplitude about T_{mean}
- ↕ Total thermal striping amplitude for a given time span, ΔT^*

$$* \Delta T = T_{I+1} - T_I = \text{Total striping amplitude for a given time span}$$

The total thermal striping amplitudes (-51.15, 21.04, -60.02, 40.38) are input to the "RAINFAT" program which calculates stresses, strains and fatigue damage values for the structure.

A negative amplitude, e.g., -51.15, indicates that the surface temperature of the structure has increased during the time span (1.36 to 1.62) in question and vice versa for a positive amplitude (see Figure 2) causing "RAINFAT" to calculate compressive and tensile stresses and strains, respectively.

Figure A-5

ILLUSTRATION OF TOTAL THERMAL STRIPING AMPLITUDE (ΔT^*) HISTORY AS INPUT TO "RAINFAT"

- o Multiply each amplitude in the history by the temperature gradient between adjacent assemblies. This converts the amplitudes into strains (See Figure A-6).
- o Calculate the equivalent load due to a maximum strain.
- o Divide the maximum strain into 100 equal elements. The elements represent segments of the cyclic stress strain curve. Figure A-7 illustrates how the monotonic curve (loop shape curve) is changed to a cyclic curve and is divided into equivalent elements.
- o Calculate stresses and load range for each of the elements using the Newton Iteration Technique.
- o Calculate the starting stress and load range.
- o Simulate the load-stress-strain path. Figure A-7 illustrates how the code simulates the hysteresis loop for the given history.
- o Calculate the damage as each cycle is completed.
- o Sum the damage for each completed cycle.

The summation of the damage completes the procedure. Additional information on this procedure can be obtained from Reference 9.

A.3.0 DAMAGE SENSITIVITY (STEP 2)

Damage obtained through the RAINFAT code was found to be sensitive to elastic slope/no elastic slope, film coefficients, temperature and time.

A.3.1 Sensitivity of Damage to Elastic Slope/No Elastic Slope

The non-elastic slope fatigue equation (Eq. 2 in Figure A-8) was obtained from actual test data (Reference 1). The equation for a fatigue curve with elastic slope (Eq. 1 in Figure A-8) was derived in Reference A-7. Figure A-8 illustrates the differences for the minimum (-2σ) sodium fatigue equation. Equation 1 is the more general form of the fatigue equation, which consists of both a plastic and an elastic strain fatigue component. Equation 2 is the in-sodium plastic fatigue equation with an air asymptote. Equation 1 calculates plastic fatigue damage and variable elastic fatigue damage.

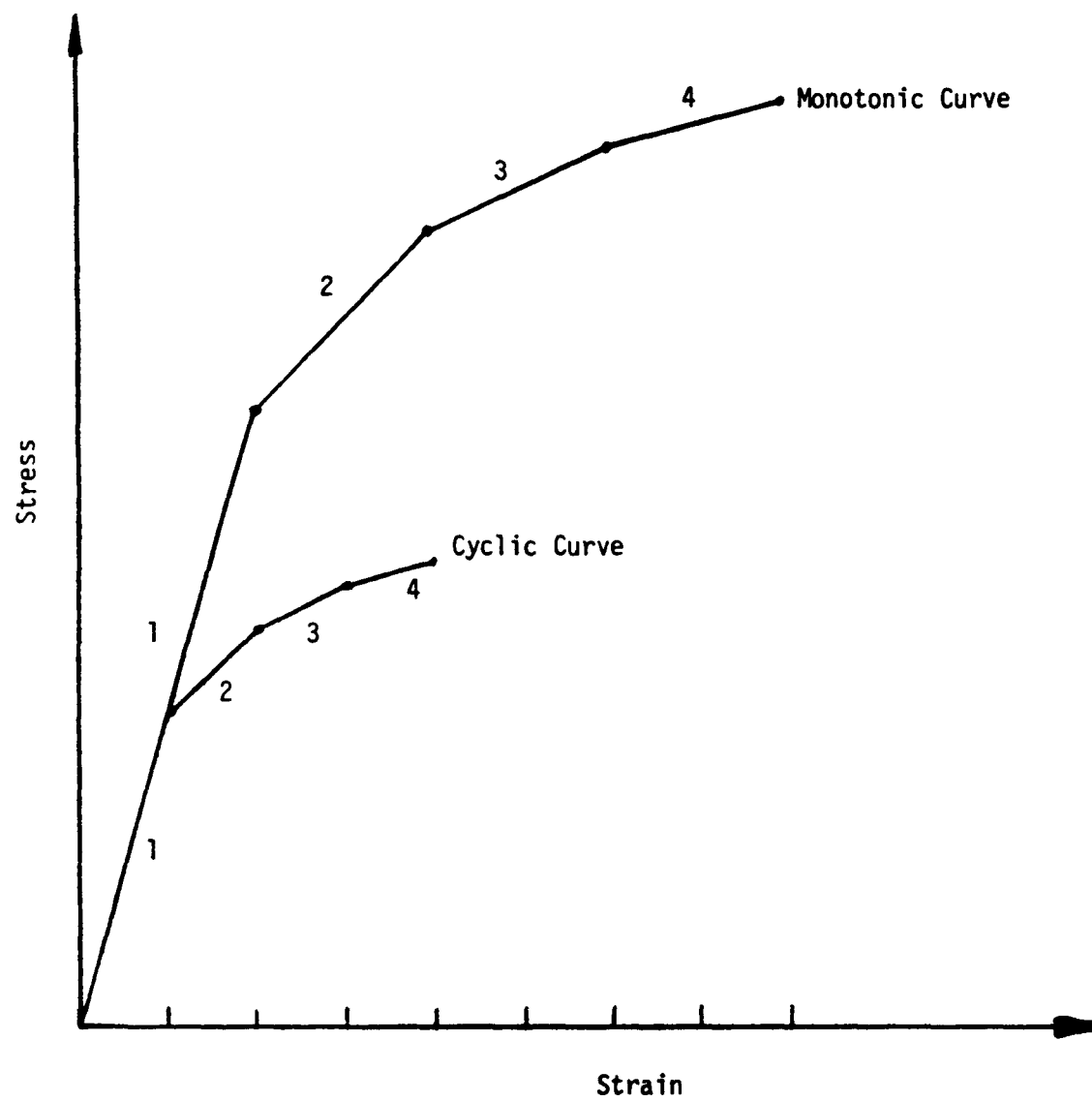
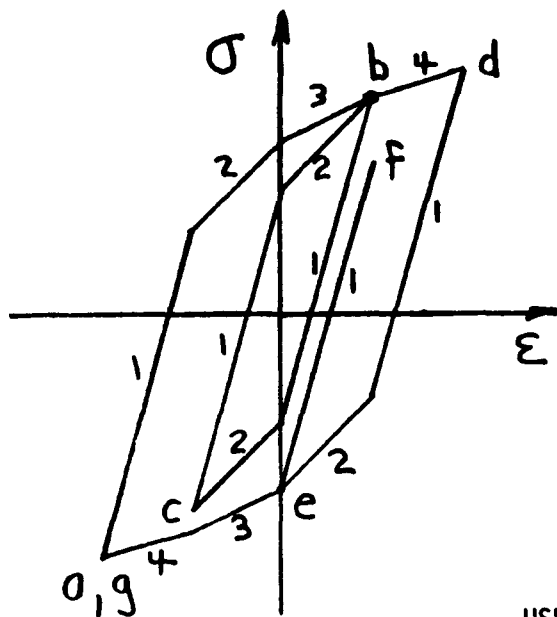
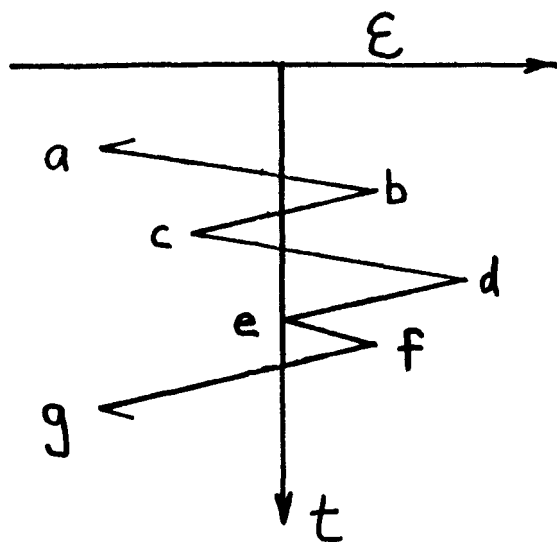


Figure A-6
ELEMENT REPRESENTATION OF STRESS-STRAIN CURVES



	a	→b	→c	→d	→e	→f	→g
1	-1	+1	-1	+1	-1	+1	-1*
2	-1	+1	-1	+1*	-1	-1	-1
3	-1	+1	+1	+1	+1	+1	-1
4	-1	-1	-1	+1	+1	+1	-1*

Figure A-7

USE OF ELEMENTS AND AVAILABILITIES TO COUNT
CYCLES AND SIMULATE STRESS-STRAIN BEHAVIOR

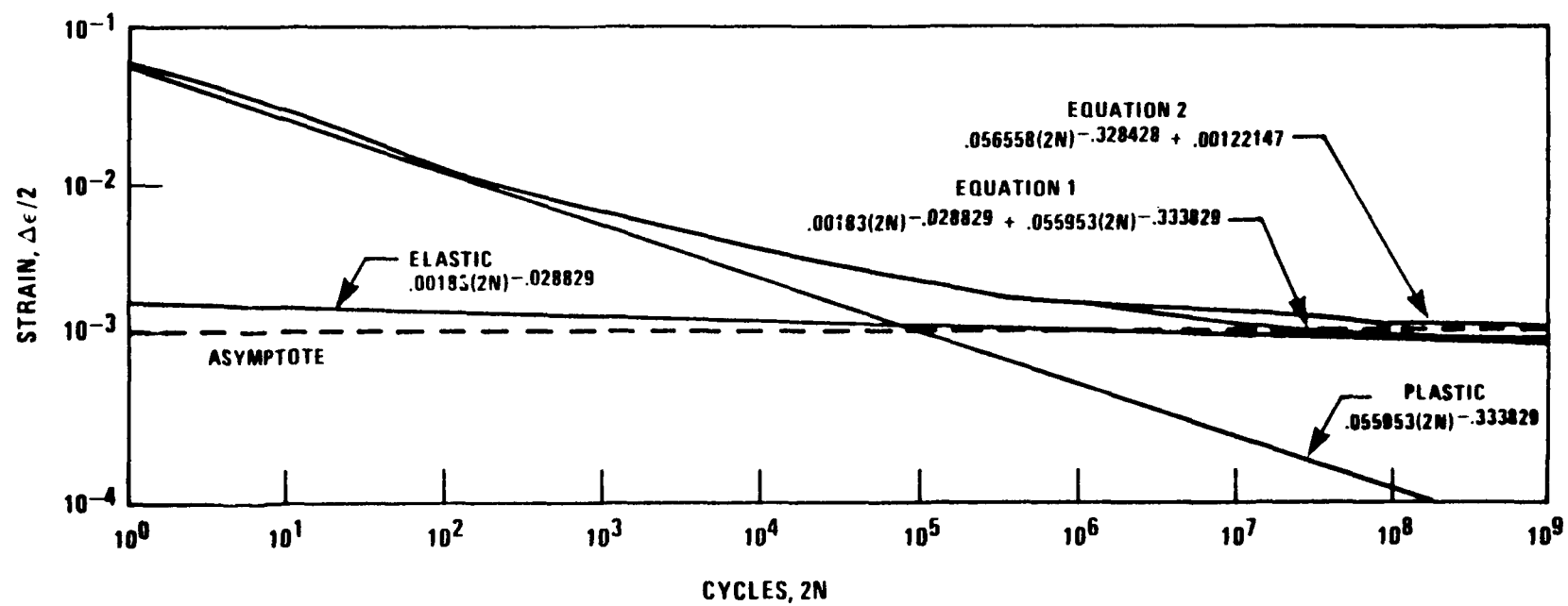


Figure A-8
Fatigue Design Curve Sodium, 20

Equation 2 calculates only plastic fatigue damage. The elastic fatigue damage is constant. Equation 1 can be divided into two separate curves; one for plastic strain amplitude and one for elastic strain amplitude. Note that the plastic strain amplitude is higher than the elastic strain amplitude up to 8×10^4 cycles, after which the plastic strain amplitude is less than the elastic strain amplitude. In general, equation 1 and equation 2 give similar results up to 1×10^6 cycles. However, at this point, the two curves diverge making equation 1 conservative at higher cycles. The elastic slope damage ranges from 1.6 to 15 times higher than the no elastic slope damage.[10]

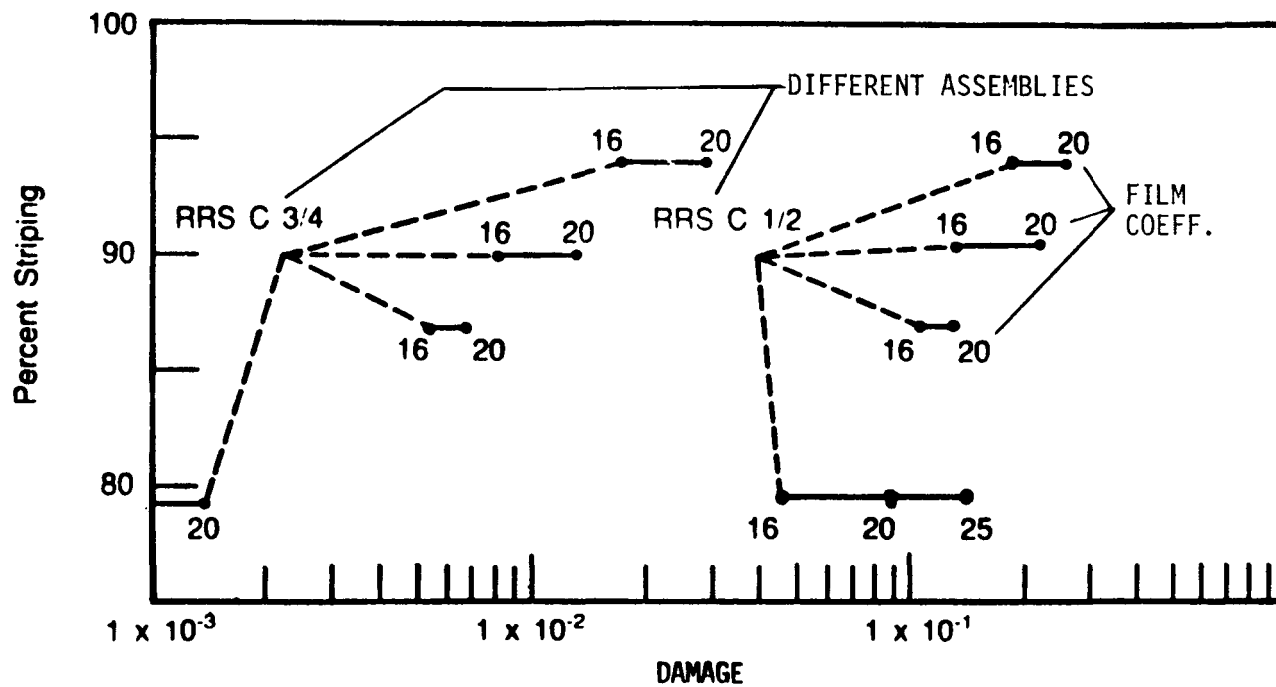
A.3.2 Sensitivity of Damage to Film Coefficient

Figure 9 shows that a change in film coefficient of just a few thousand BTU/hr-ft²°F significantly changes the amount of accumulated damage. At 25K BTU/hr-ft²°F, the damage for the 79.3% fluid striping case (RRS 3/4) was 1.4×10^{-1} . At 20K BTU/hr-ft²°F, the damage was 38.5% less than the damage at 25K BTU/hr-ft²°F. At 16K, the damage was 48% less than the damage at 20K. All other fluid striping cases in Figure A-9 show that there is a significant difference in damage for different film coefficients.

A.3.3 Sensitivity of Damage to Temperature

Figures A-10 and A-11 show the effect of typical fuel assembly outlet temperatures on striping. Figure A-10 represents a fluid striping case with a low striping percentage. Figure A-11 represents a fluid striping case with a high striping percentage. Both cases were run at the same outlet temperature. However, the fluid striping case with the lower striping percentage saturated out at a higher temperature than the higher striping case. This shows that increasing the percent striping decreases the temperature at which damage saturation occurs. Therefore, in order to get the same damage level at a high percent striping as for the lower percent striping the outlet temperature must be lower.

STUDY OF THE EFFECT OF FILM COEFFICIENT ON DAMAGE



RRS STRIPING POTENTIAL
FATIGUE CURVE: IN-SODIUM, -2σ , WITH AIR ASYMPTOTE
FILM COEFFICIENT: $1000 \times \text{BTU}/^\circ\text{F}\cdot\text{FT.}^2\cdot\text{HR.}$

Figure A-9

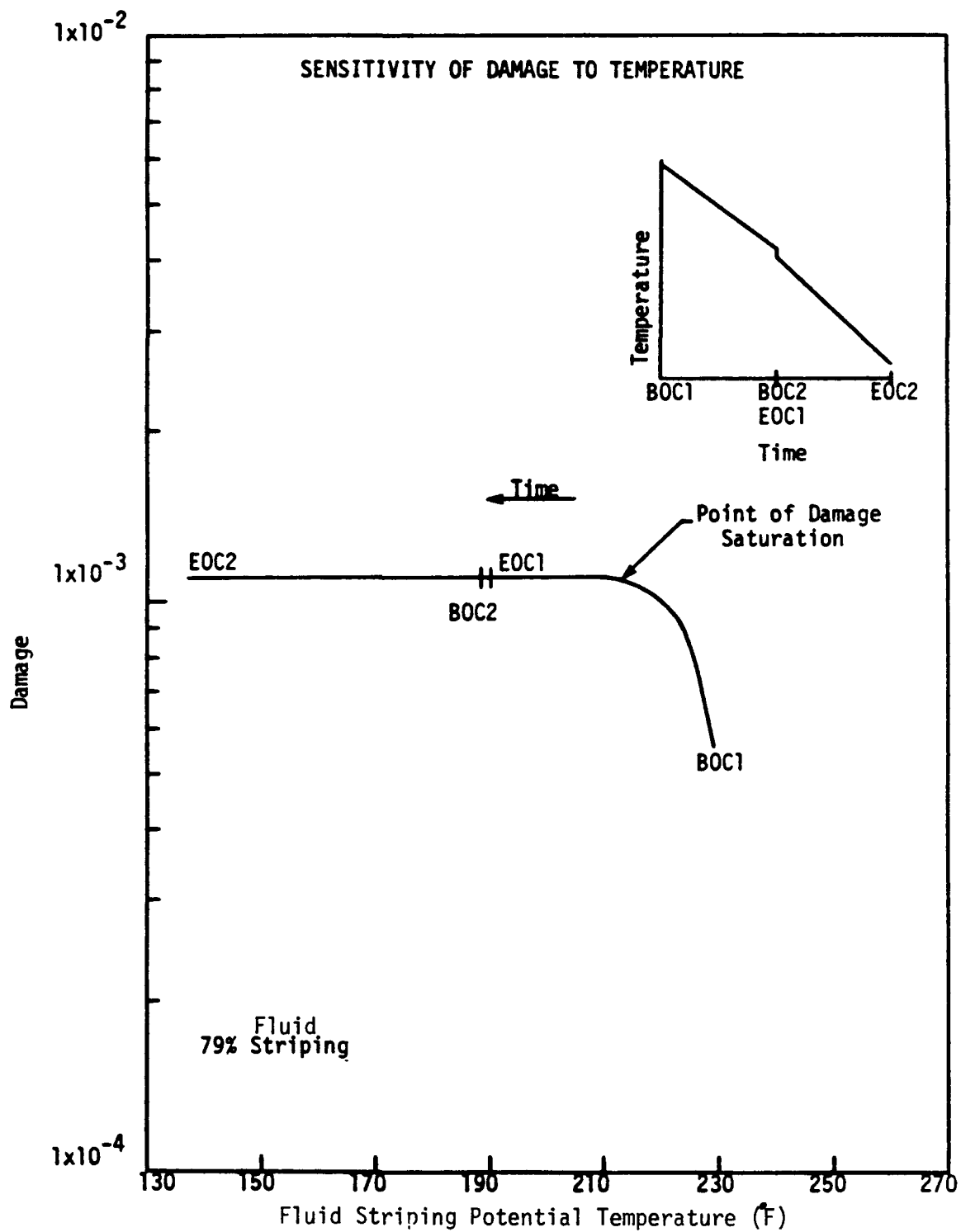


Figure A-10
TYPICAL FUEL ASSEMBLY OUTLET TEMPERATURE STRIPING POTENTIALS

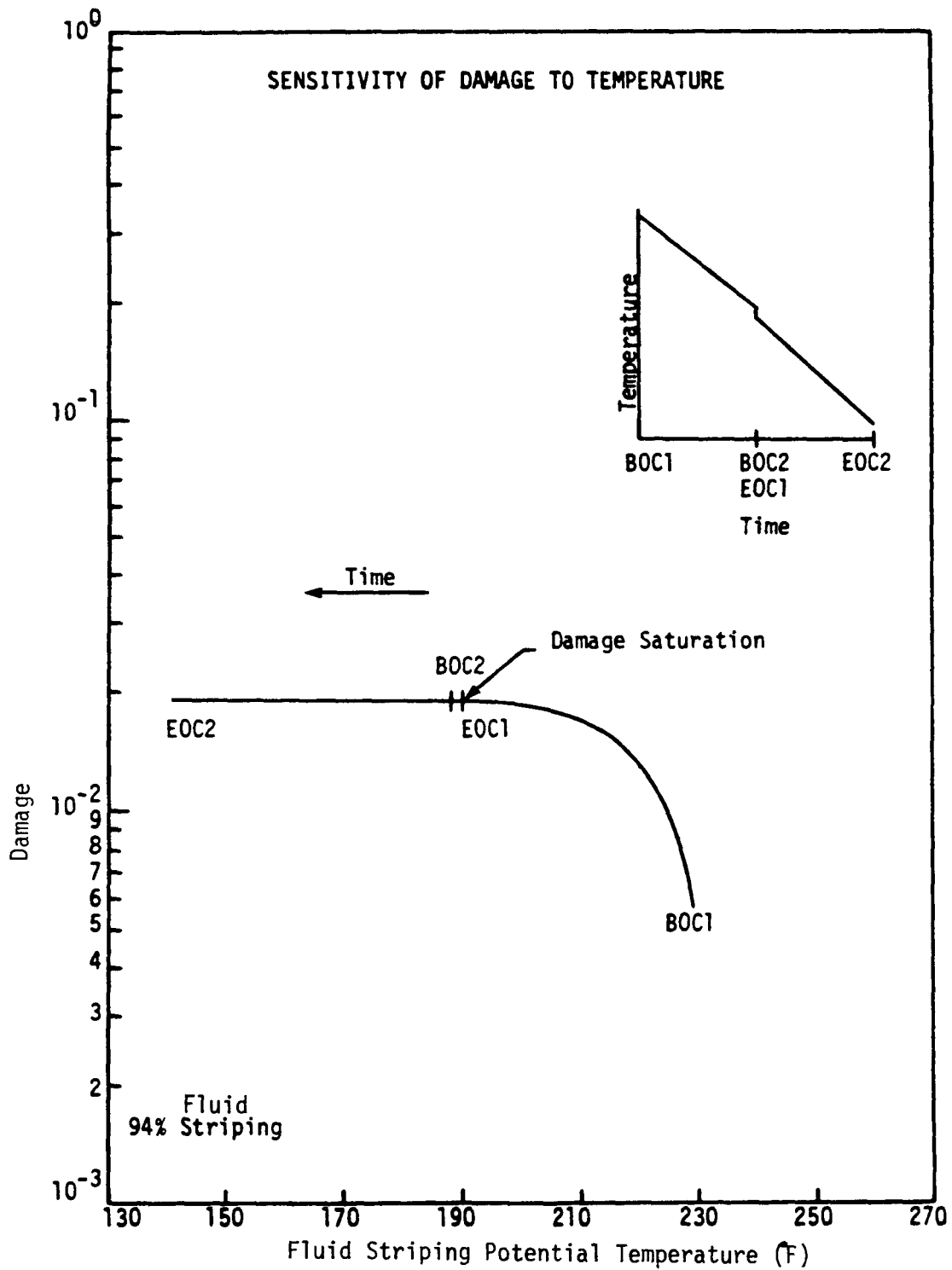


Figure A-11
TYPICAL FUEL ASSEMBLY OUTLET TEMPERATURE STRIPING POTENTIALS

A.3.4 Sensitivity of Damage to Time

Figure A-12 shows the effect of time on damage. For this particular striping case, a typical fuel assembly achieved damage saturation in less than fifty days. Whereas a typical RRS assembly did not achieve damage saturation until the end of the first cycle (128 days). In addition, RRS assemblies and radial blanket assemblies remain in the core longer than fuel assemblies. Therefore, in order to get the same damage accumulation for an RRS or radial blanket assembly, the percent striping or the temperature must be lower.

A.4.0 SELECTION OF DESIGN CASES (STEP 3)

The fuel assembly, the blanket assembly, the removable radial shield assembly, and the primary control assembly fluid striping cases were chosen from the outlet nozzle striping tests (MON and FA test).

A.4.1 Selection of Fuel and Blanket Assembly Design Case

To select a design case for the fuel and blanket assemblies, the highest fluid striping cases in the outlet nozzle feature test and seven assembly test were analyzed. In all, seven cases were chosen. To compare the cases, each case was run at constant temperatures for two cycles. The temperature ΔT 's and the resulting accumulated damage for each ΔT were plotted for each selected case.

Figure A-13 contains the above temperature/damage correlation for each striping case selected. The following observations were made:

- o Some outlet nozzle striping tests yield higher damage than others.
- o Accumulated damage does not necessarily decrease with decreasing fluid striping.

STUDY OF HOW DAMAGE SATURATES WITH TIME

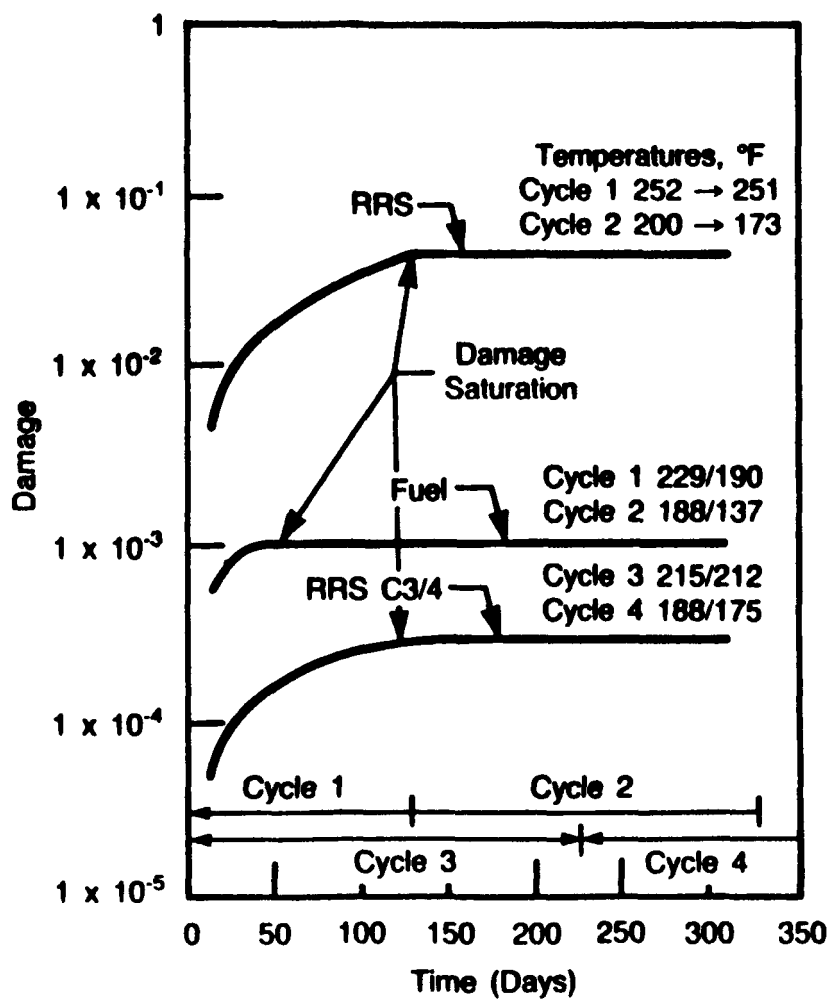


Figure A-12

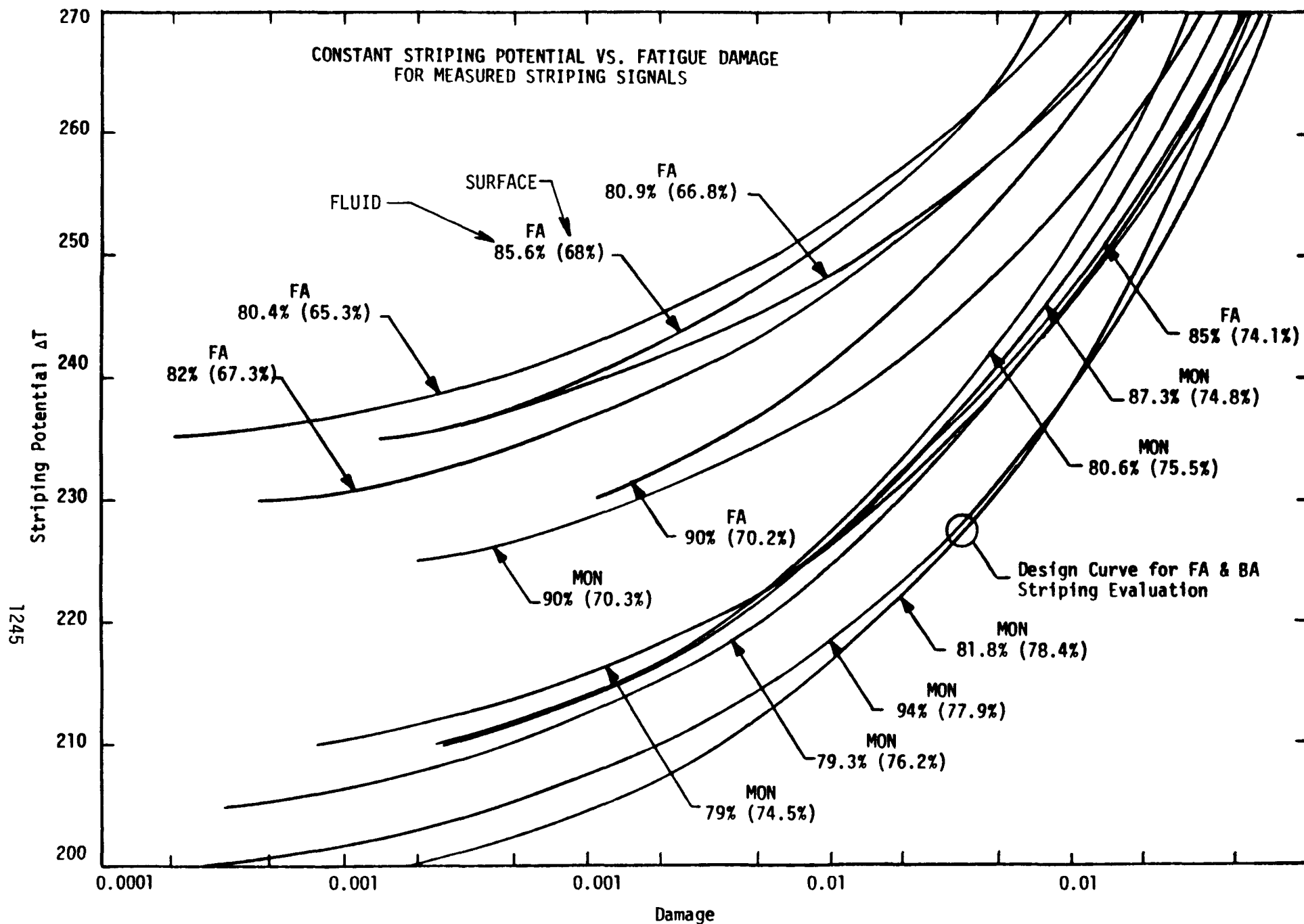


Figure A-13. FLUID (SURFACE) PERCENT STRIPING

To explain the discrepancies, the following analysis was performed:

- o All striping cases were run for a tenth of a cycle (12.8 days) at temperatures of 240°F and 300°F.
- o To compare the two temperature cases, the total damage, the damage due to all other strain ranges and the damage due to maximum strain were plotted in Figure A-14.

Figure A-14 shows that the total damage was affected more by the damage due to other strains than the damage due to the maximum strain at 300°F. However, at 240°F, the total damage is affected more by the damage due to the maximum strain. This happens because more strain cycles are above the endurance strain limit at 300°F than at 240°F.

In the analysis, it was found that fluid striping is a good screening method for selecting worst striping cases, even though it cannot be used as a final selection method. Surface striping ranges from 96% to 78% of fluid striping no matter what the magnitude of the fluid striping is. Therefore, when given a set of striping tests, the following procedure is performed:

- o Find the maximum fluid striping value.
- o Find 78% of the maximum fluid striping value.
- o Find all striping cases in this range.

This procedure assures that the striping case with the most damage for the fuel and blanket assembly is selected.

The final criteria applied to selection of a design case was a design damage limit of 0.1. The 81.8% striping case and the 94% striping case in Figure A-13 encompass all other striping cases at 0.1 damage. In addition, the 81.8% and 94% cases have higher surface striping and higher maximum strain ranges than the other striping cases. The 81.8% striping case was not affected by low cycle strain ranges and the 94% case was only slightly affected by low cycle strain ranges. This made these two cases less sensitive to temperature changes and conservative cases for design.

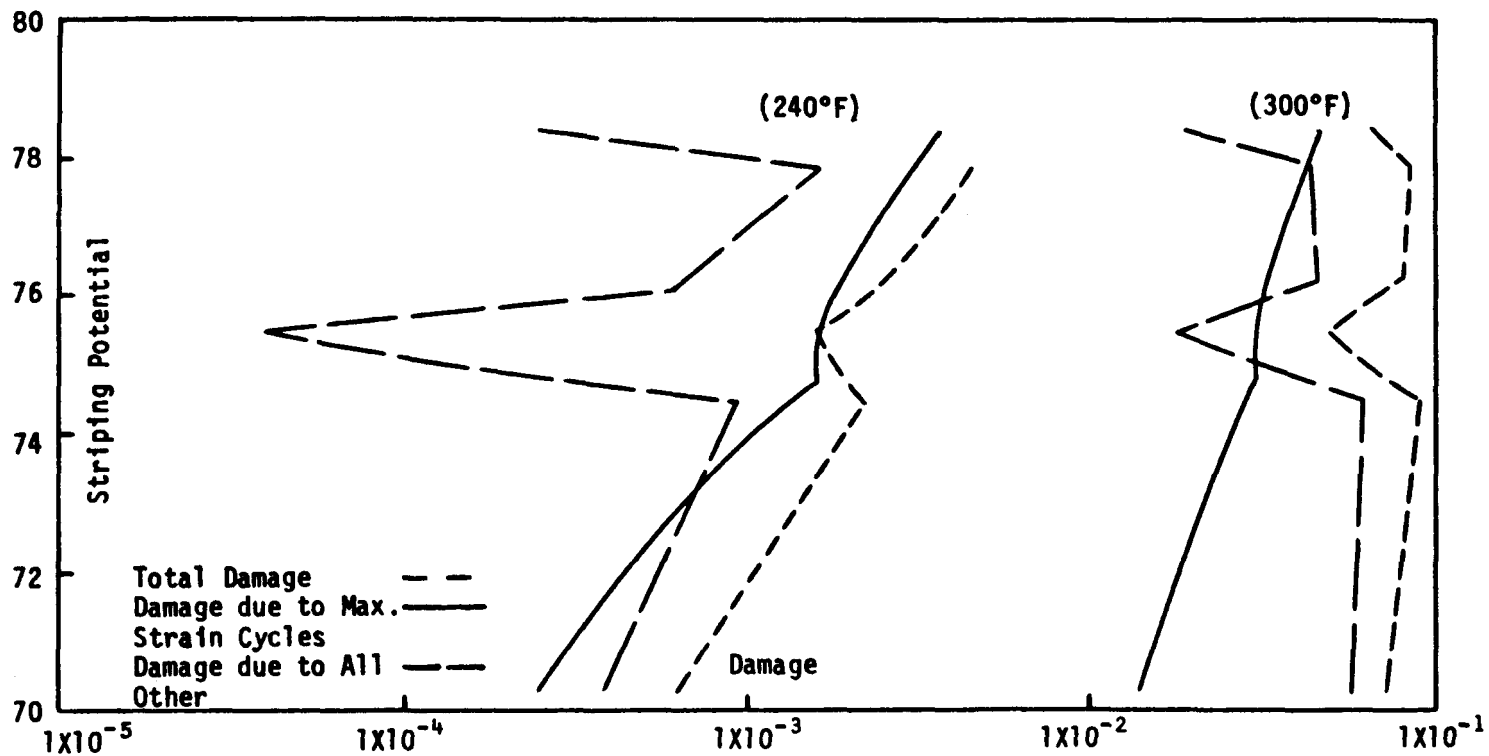


Figure A-14

HOW THE MAXIMUM STRAIN RANGE AND LOW CYCLE STRAIN RANGES
EFFECT TOTAL DAMAGE

A.4.2 Selection of Removable Radial Shield Design Case

The design striping case for the RRS was selected in a different way than the FA and BA design case. Since the maximum fluid striping of the RRS was known from the IRFM test, but data were not in a processable format, the available FA signals were searched to find the highest damage for a given maximum fluid percent and to find the maximum surface striping for each fluid striping case. The RRSA were conservatively designed for 30 years.

Tables A-1 and A-2 show the results of the RRSA analysis. Table A-1 contains the results of the analysis using the same fatigue curve used in the fuel and blanket assembly analysis. Initially, it was expected that the maximum fluid striping would be between sixty and seventy percent for striping temperature potentials of 280°/212° for cycle 1 and 2 and 286°/195° for cycles 3 and 4. At these temperatures, fluid striping of 59.4% would pass the design criteria of 0.1.

Later, the same analysis was performed using a fatigue curve generated by EG&G using strain controlled fatigue data which were failed under load control.[11] This equation (see Table A-2) was used to address some concern about the endurance limit. Table A-2 contains the results of this evaluation for different striping potentials and striping histories. The final analysis for the RRSA was performed using final RRSA temperatures of 292°/223° for cycle 1 and 2 and 312°/217° for cycle 3 and 4. Table A-1 contains the results using the final temperatures and the fatigue curve used for the fuel and blanket assemblies for this case, 52.3% striping would be allowable. However, in Table A-2, the allowable case would be 48.7%. It should be noted that this was the maximum so the actual striping should be somewhat less.

A.4.3 Selection of the Design Case for Primary Control Assembly

Selection of the design case for the primary control assembly was performed in the same way as selection for the fuel and blanket assembly. However, in this case, only tests which were prototypic were analyzed. Figure A-15 contains

TABLE A-1
RRS EVALUATION
 $\Delta \Sigma/2 = .056558 (2N)^{-.328428} + .001215$

<u>% Striping</u>	<u>68%</u>	<u>67.3%</u>	<u>66.8%</u>	<u>65.3%</u>	<u>59%</u>	<u>53.3%</u>
Cycle 1/2	280/212	280/212	280/212	280/212	280/212	280/212
Cycle 3/4	286/195	286/195	286/195	286/195	286/195	286/195
No Mean Stress						
Damage 878 Days	.22	.089	.237	.139	.0021	0
Damage 30 Years	2.2	.89	2.37	1.39	.021	0
Mean Stress						
Damage 878 Days	.245	.516	.65	.057	4.3 E-5	0
Damage 30 Years	2.45	5.16	6.5	.57	4.3 E-4	0

<u>% Striping</u>	<u>59%</u>	<u>53.3%</u>	<u>59%</u>	<u>53.3%</u>	<u>53.3%</u>	<u>48.1%</u>
Cycle 1/2	285/217	285/217	300/232	300/232	292/223*	292/223*
Cycle 3/4	291/200	291/200	306/215	306/215	312/217*	312/217*
No Mean Stress						
Damage 878 Days	.0043	5.1 E-9	.0222	2.15 E-4	-	0
Damage 30 Years	.043	5.1 E-8	.222	2.15 E-3	-	0
Mean Stress						
Damage 878 Days	4.3 E-5	6.2 E-4	-	-	1.23 E-2	
Damage 30 Years	4.3 E-4	6.2 E-3	-	-	1.23 E-1	

*FINAL TEMPERATURES

TABLE A-2
RRS EVALUATION

$$\Delta \Sigma/2 = .337850 (\text{LOG } N)^{-3.63} + .001030$$

<u>% Striping</u>	<u>59%</u>	<u>53.3%</u>	<u>48.1%</u>	<u>42.8%</u>
Cycle 1/2	300/232	300/232	300/232	300/232
Cycle 3/4	306/215	306/215	306/215	306/215
No Mean Stress				
Damage 878 Days	.1862	1.39 E-2	2.05 E-4	0
Damage 30 Years	1.862	1.39 E-1	2.05 E-3	0
Mean Stress				
Damage 878 Days	.0329	7.21 E-2	2.2 E-6	0
Damage 30 Years	.329	7.21 E-1	2.2 E-5	0

<u>% Striping</u>	<u>59%</u>	<u>53.3%</u>	<u>53.3%</u>	<u>48.1%</u>
Cycle 1/2	285/217	285/217	292/223*	292/223*
Cycle 3/4	291/200	291/200	312/217	312/217
No Mean Stress				
Damage 878 Days	6.2 E-2	-	-	4.6 E-4
Damage 30 Years	6.2 E-1	-	-	4.6 E-3
Mean Stress				
Damage 878 Days	-	2.1 E-2	8.3 E-2	-
Damage 30 Years	-	2.1 E-1	8.3 E-1	-

*These temperatures are the final temperatures. The 48.1% case is below the design limit of .1. By interpolation between the 53.3% case and the 48.1% case find that the maximum striping at the design limit would be 48.7%.

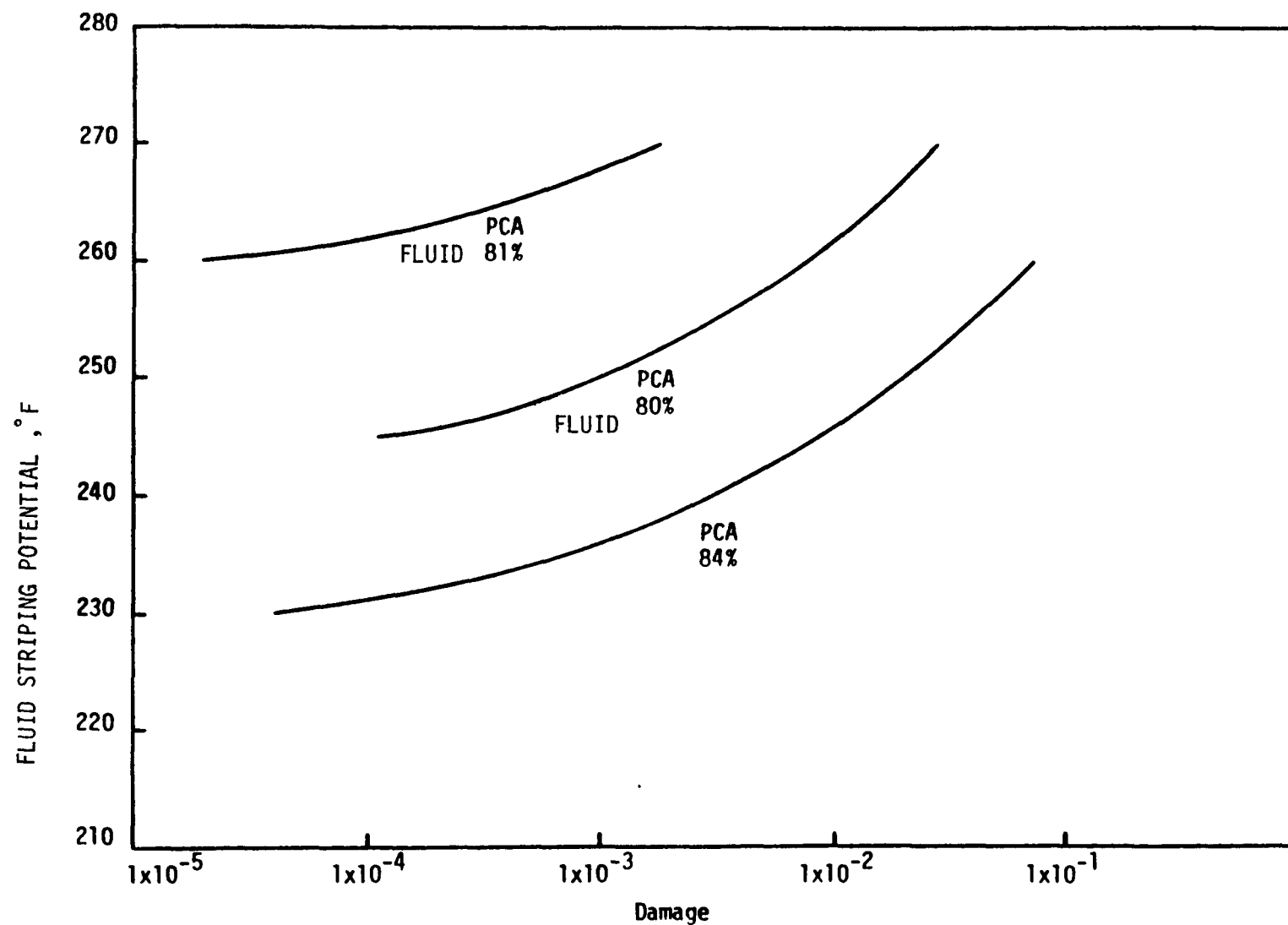


Figure A-15
CONSTANT STRIPING POTENTIAL VS. FATIGUE DAMAGE
FOR MEASURED FLUID STRIPING SIGNALS

the curves for the striping cases which had damage accumulation. Since the 84% case encompasses the other two cases at 0.1, it was chosen as the design case.

A.5.0 RESULTS OF OUTLET NOZZLE STRIPING DAMAGE

The final results for the fuel and blanket assembly are contained in Table 3.12.7-1. This table tabulates the fluid temperature potentials, the ΔT s, and the damage with assembly combinations. The cases listed are representative of the worst cases in the core. It was found that the damage to all assembly combinations passed the design limit of 0.1. Table 3.12.7-2 contains the results of the primary control assembly analysis. This table shows that the selected design case damage passes the design limit of 0.1 using the design fatigue curve and the EG&G fatigue curve. Tables A-1 and A-2 show that the RRS assemblies pass the design damage criteria of 0.1 for 30 years in the core.

A.6.0 REFERENCES

1. TID 2666, Nuclear Systems Materials Handbook.
2. C. E. Jaske, J. S. Perrin, H. Mindlin, Low Cycle Fatigue of Reactor Structural Materials, Reactor Technology, Fall, 1972, V-1. 15, No. 3.
3. C. E. Jaske, J. S. Perrin, H. Mindlin, Cycle Stress-Strain Behavior of Two Alloys at High Temperature, Reactor Technology.
4. B. L. Pierce, R. J. Agosti, K. J. Wells, TAP-A Program for Computing Transient or Steady State Temperature Distribution, May 1970.
5. J. J. Huba, Thermal Striping Statistical Analysis, Programs IELIM and STRIPE, Draft Report.
6. M. D. Carelli, P. W. Rosenthal, W. R. Rymer, Final Predicted Steady State Thermal-Hydraulic Performance of the CRBR Fuel and Blanket Assemblies, April 1983, WARD-D-0324.
7. S. J. Layman, Verification of "RAINFAT" Computer Code, June 1983, LRA-FR-83-4.
8. W. R. Brose, Input to Westinghouse R&D RAINFLOW Computer Program.
9. S. J. Layman, Thermal Striping Damage Sensitivity Study, June 1983.
10. Dr. D. W. Keefer, Idaho National Engineering Laboratory Mechanical Properties Design Data Problem, July 1982, ORNL/MSP/1.3-82/3.

APPENDIX B - THERMAL STRIPING DAMAGE CHARACTERIZATION

The existence of thermal striping fatigue damage is being postulated based on in water thermal striping tests and conservative analysis. Although some thermal striping fatigue failures or cracks in elbows, pipes and test specimens have been induced under controlled and extreme conditions no major thermal striping induced failures have been reported on sodium cooled reactor components. The existence of thermal striping of the postulated severity is uncertain.

The effect of thermal striping fatigue cycles on a structure may be measurable without detailed temperature measurements. Following the thermal striping fatigue tests in the TSTF tests, the specimens were destructively examined. The test specimen and test procedure are described in Reference 8 of Section 3.12.10. After sectioning the cylindrical test specimen micro-hardness measurements through the thickness of the specimen were performed. Figure B-1 shows the micro hardness (DPH) as a function of the distance below the surface for a specimen cycled at 500°F temperature range to 5×10^5 cycles. This specimen cracked at the surface. Also plotted on this figure is the relative metal temperature as a function of distance below the surface. Both curves have similar shapes; therefore, the local micro hardness increase appears to be proportional to the local temperature amplitude. Near the midwall of the specimen in the striping unaffected zone the hardness is 150 DPH. At the surface it saturates at a maximum between 250 and 300 DPH. Hardness profiles for six different specimens are shown in Figure B-2. This figure shows that the hardness penetration:

- o Increases with higher striping temperature
- o Increases at a lower striping frequency

The profile for the dunk specimen, which was dunked in up to 800°F sodium, indicates that the hardness saturates through significant depth of the specimen.

HARDNESS PROFILES THROUGH THE THICKNESS FOLLOW TEMPERATURE PROFILE

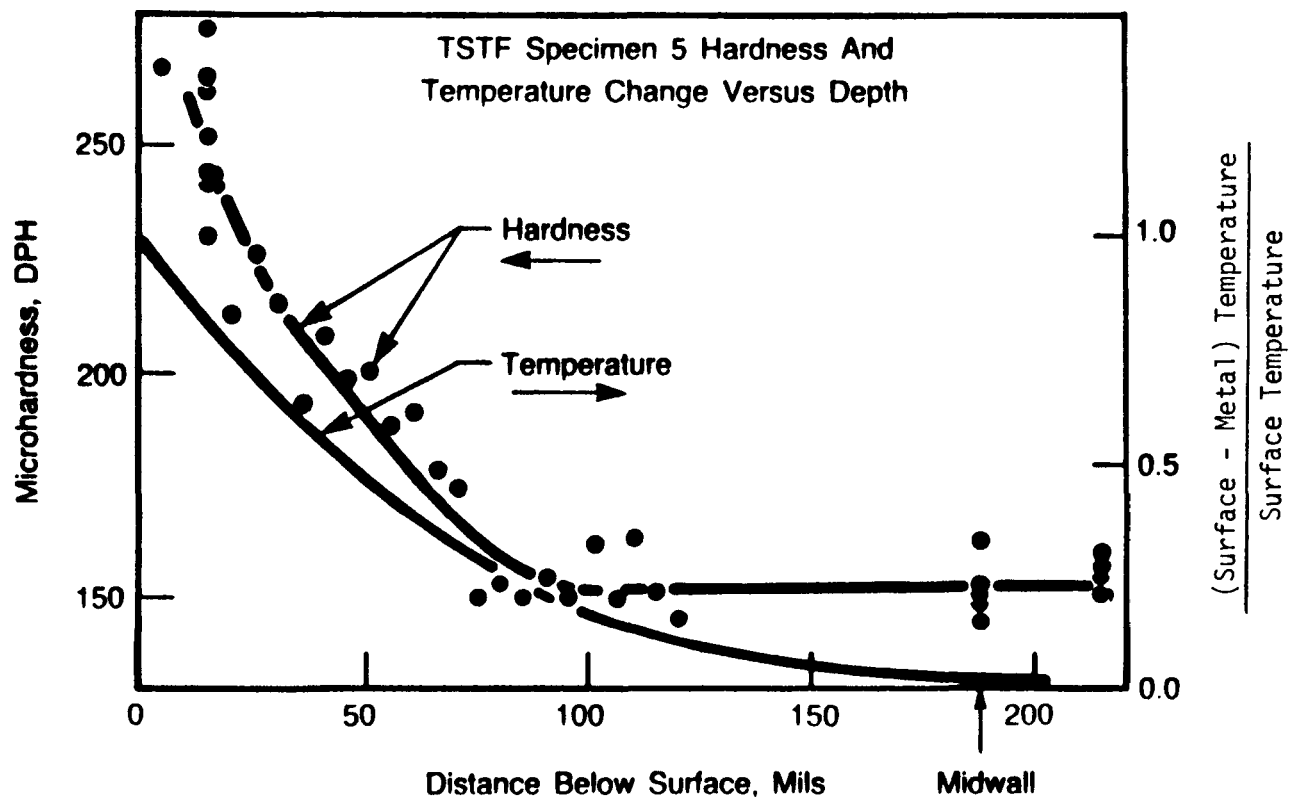


Figure B-1

TSTF SPECIMEN HARDNESS CHANGE VERSUS DEPTH BELOW SURFACE DUE TO STRIPING

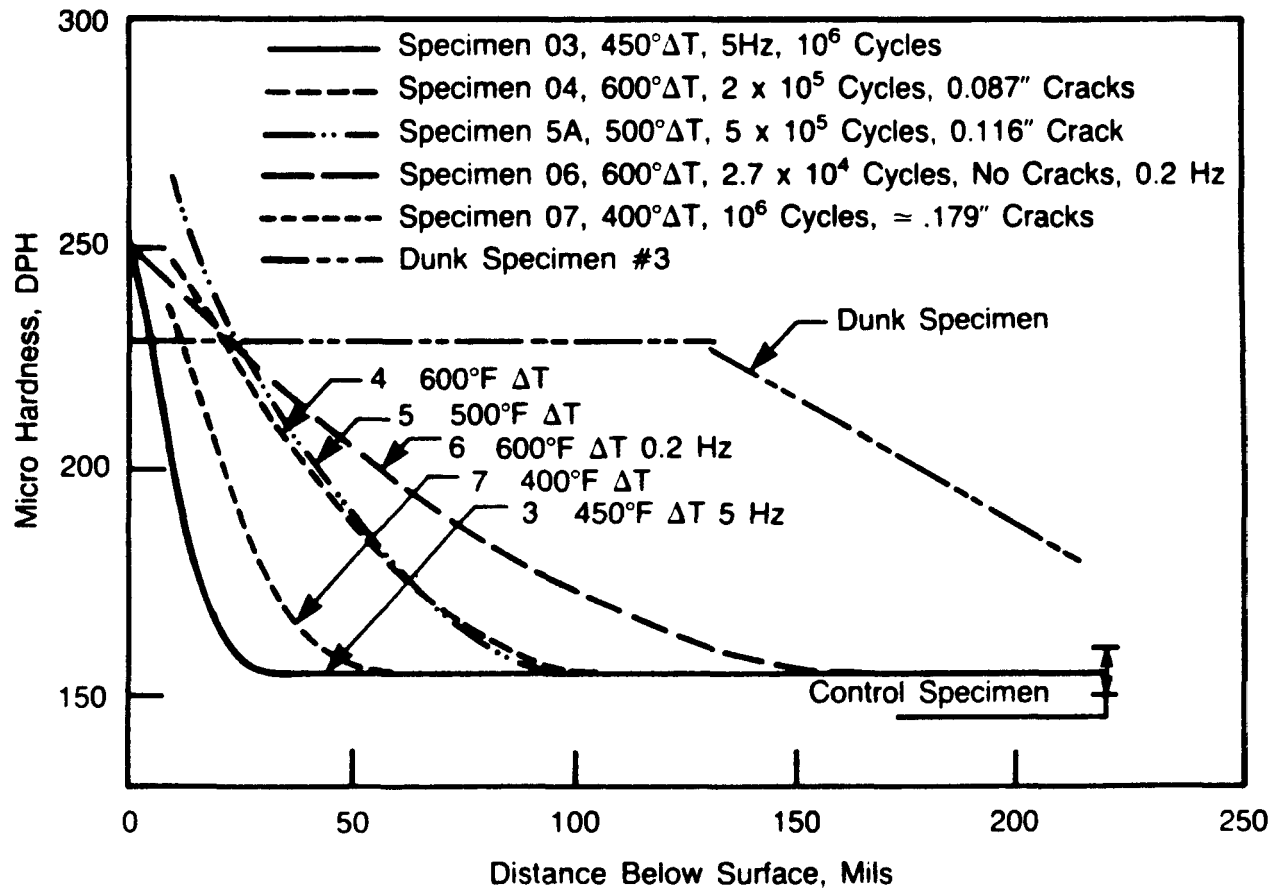


Figure B-2

The results from the past test examination indicates crack propagation in the specimen to a depth beyond a zone in which temperature changes are calculated to cause local yielding of the specimen. The magnitude of the local yielding was indicated by a zone of cold worked microstructure in a 0.05 mm band on both sides of the cracks, from which it was inferred that the cracks established their own stress field near the tip of the crack. This, in turn, determined the direction of further crack propagation and resulted in curved crack growth (or crack kinking) and crack bifurcation.

The significance of these observations is that:

- o A potential correlation was found between striping fatigue damage and the penetration of the striping temperature oscillations into the wall thickness.
- o A method to determine if structures with surface cracks have been subjected to thermal striping fatigue without detail knowledge of the fluid and temperature field.

This method could be developed into a useful tool for failure and local characterization of reactor structures.

APPENDIX C

GENERAL HCDA PROGRESSION PATHS*

The progression of a Hypothetical Core Disruptive Accident (HCDA) can be divided into four phases representing different physical conditions of the core: initiating, meltout, large-scale pool and hydrodynamic disassembly, as shown in Figure C-1. The accident may terminate (i.e., permanent neutronic subcriticality) during any of these four phases, and the hydrodynamic disassembly phase may, in general, be entered from any of the other phases. Best-estimate paths are denoted by thick lines, whereas thin lines represent accident paths predicted with pessimistic assumptions for the CRBRP.

The initiating phase covers the first phase of the accident starting from normal operating conditions where the assembly structure is basically intact. During this phase an imbalance between reactor energy generation and heat removal would cause fuel to melt and move within the assembly. As fuel is removed from the core, the core power decreases to permanent shutdown, or some stable power level with in-place cooling when the sodium flow is maintained as in the transient overpower accident. If the sodium flow continues to decrease as in the loss of flow accident, further damage to the core would occur, and the accident would progress into the meltout phase.

The initiation of the meltout phase is defined as the loss of radial restraint on material relocation provided by the hexcan walls. At this time, part of the reactor core may still satisfy the initiating phase definition, but some of the fuel assemblies will have disrupted and their materials will be radially merging. As the hexcan walls are melted the interstitial spaces between the core assemblies become flow paths for the molten core materials. Downward and radially outward movement of the core materials into the interstitial gaps and possibly into empty control assembly ducts allows the core to become permanently subcritical.

* Reference 4.6.2.

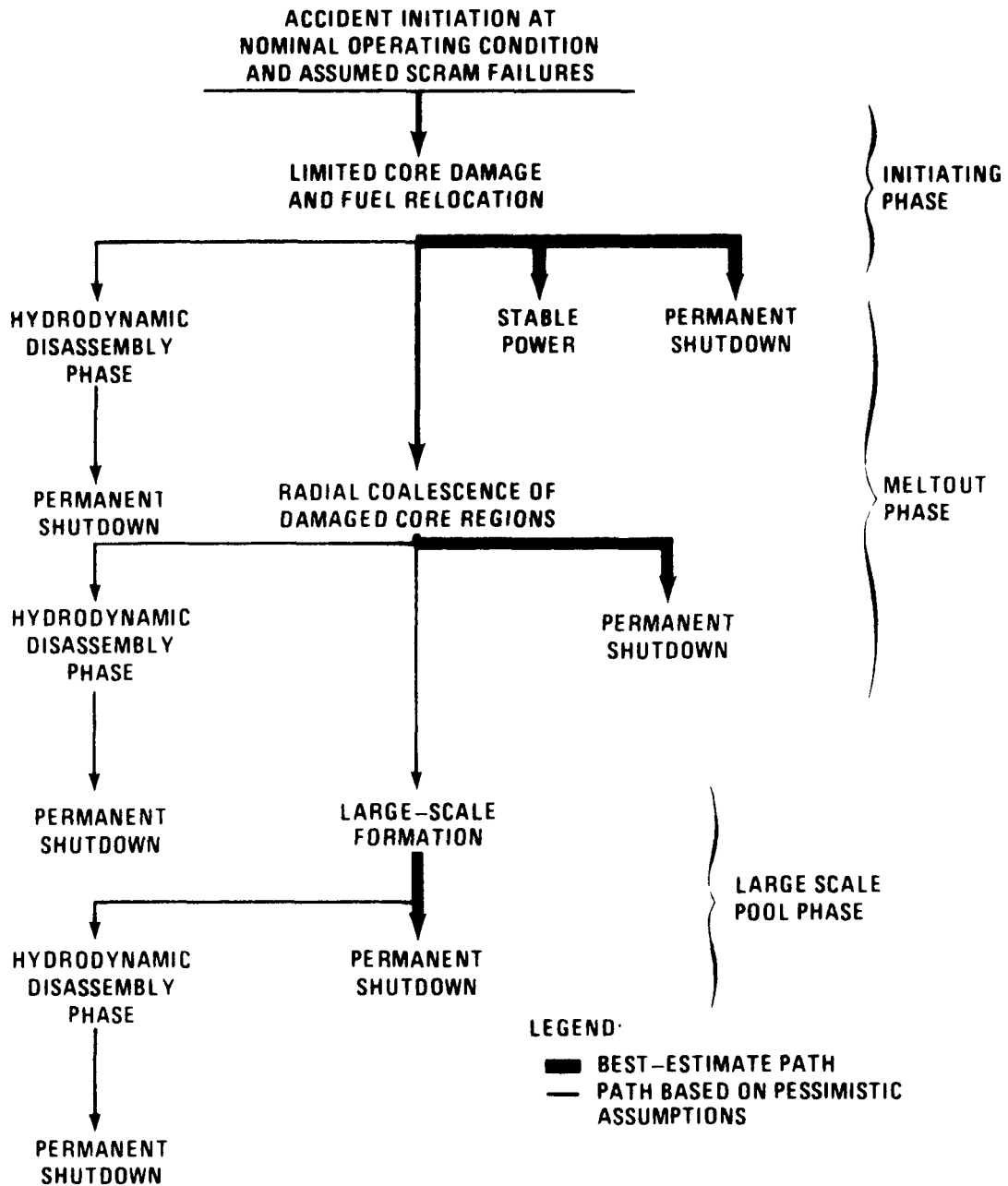


Figure C-1 CRBRP HCDA Progression Paths

If pessimistic assumptions are made relative to the timing and magnitude of fuel removal during the meltout phase, a large-scale pool phase would occur. During this phase, a contiguous molten fuel region is assumed to form whose behavior will neutronically dominate the energetics potential. Continued fuel meltout accompanied by fuel pool dilution by liquid steel, and blowdown is predicted to eventually lead to permanent subcritical conditions.

The final phase, hydrodynamic disassembly, is used to describe the core response to a sustained, superprompt critical excursion that might be predicted using pessimistic assumptions during the three preceding phases. When this phase is entered, the accident is considered energetic and the resulting work energy is calculated and compared with the SMBOB loads provided in the CRBRP structures and components. The primary assumption is that rapidly increasing fuel temperatures result in pressures which far exceed the mechanical restraint capabilities of the core components. Hence, the motion of materials is modeled as purely hydrodynamic in nature. This hydrodynamic disassembly phase is not predicted in the present evaluations unless some particular pessimistic assumptions are made in two scenarios. Even in the two pessimistic scenarios, the energetics potential is well within the structural margin.